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PG&E Letter DCL-94-133

U.S. Nuclear Regulatory Commission
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Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities"

Gentlemen:

In response to Generic Letter 88-20, Supplement 4, enclosed is the Individual Plant Examination of External Events (IPEEE) Report documenting PG&E's completion of the IPEEE for Diablo Canyon Power Plant. The IPEEE Program was conducted consistent with the PG&E program plan described in PG&E Letter DCL-91-308, dated December 23, 1991. The enclosed IPEEE Report concludes that no vulnerabilities related to core damage sequences or containment performance exist at the Diablo Canyon Power Plant.

This submittal completes PG&E's response to Generic Letter 88-20, Supplement 4 for the IPEEE.

Sincerely,

Greg Rueger

Gregory M. Rueger

Subscribed and sworn to before me this 27th day of June 1994.

Bianca E. Zelnik

Notary Public

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6490S/DDS/2020

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS REPORT

FOR DIABLO CANYON POWER PLANT UNITS 1 AND 2

IN RESPONSE TO GENERIC LETTER 88-20 SUPPLEMENT 4

PACIFIC GAS AND ELECTRIC COMPANY

JUNE 1994

Table of Contents

Table of Contents

Page

Table of Contents

Table of Contents

List of Tables

List of Tables

List of Figures

1. EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

This report documents Pacific Gas and Electric Company's (PG&E's) work in response to NRC Generic Letter 88-20, Supplement 4 (Reference 1-1) which requested each utility to perform an Individual Plant Examination of External Events (IPEEE) (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant under full power operating conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage and radioactive material release, and (4) if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents."

The analyses summarized in this report represent an update and enhancement of the external events portion of the original Diablo Canyon Probabilistic Risk Assessment (DCPRA-1988) (Reference 1-2) performed as part of the Long Term Seismic Program (LTSP) (Reference 1-3). The internal events portion of the DCPRA-1988 was updated as part of the Diablo Canyon Power Plant (DCPP) Individual Plant Examination (IPE) (Reference 1-4), which was submitted to the NRC in April 1992. The NRC issued its staff evaluation of the Diablo Canyon IPE in June 1993 (Reference 1-5).

The LTSP reevaluated the seismic design basis 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 Element 4 of this 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 by a team of consultants and utility personnel led by PLG, Inc. The DCPRA-1988 is a full-scope Level 1 probabilistic risk assessment (PRA) that evaluated the probable frequency of experiencing reactor and plant damage as the result of both internal and external initiating events. While the PRA was performed only 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 review of the LTSP is summarized in Supplemental Safety Evaluation Report No. 34 of NUREG-0675 (Reference 1-6), in which the DCPRA-1988 was accepted. Much of the review of the DCPRA-1988 was performed by Brookhaven National Laboratory (BNL); the BNL review of the DCPRA-1988 is documented in NUREG/CR-5726 (Reference 1-7). Finally, the Advisory Committee on Reactor Safeguards accepted the NRC's review of the LTSP and DCPRA-1988 and concluded that DCPP "can be operated without undue risk to the health and safety of the public" (Reference 1-8).

To fulfill the requirements of NRC Generic Letter 88-20, Supplement 4, and as part of PG&E's living PRA program, PG&E updated the external events portion of the DCPRA-1988 to:

- Further enhance PG&E's familiarity with all aspects of the external events portion of the PRA. As part of the update of the external events portion of DCPRA-1988,

PG&E assumed complete responsibility for the external events PRA. The DCPRA models are updated and maintained at PG&E using PLG software.

- Reflect current plant design and operation. This includes the use of updated design and operational data through March, 1993 and December, 1991 respectively with human action failures and internal initiating events updated through June, 1993.

- Address the Sandia fire risk scoping study issues (NUREG/CR-5088) as part of the fire risk assessment (Section 4).

- Perform a containment performance assessment for the seismic, fire, and "other" external events PRA.

- Evaluate and present the results of the external events PRA in a manner consistent with the reporting requirements of Generic Letter 88-20, Supplement 4 and NUREG 1407 (Reference 1-9).

- Maintain a living PRA program. As discussed in Sections 2 and 6, PG&E intends to maintain a "living" internal and external events PRA with periodic updates, to incorporate changes to the operation and design of the plant. Using the insight and knowledge gained from the development of the DCPP internal and external events PRA, PG&E is committed to addressing the impact on risk resulting from certain maintenance activities, design modifications, regulatory changes, and operator actions, as appropriate.

The latest update of the DCPRA is referred to as the DCPRA-1993, and it includes the effects of external initiating events. This report largely follows the format developed in NUREG-1407.

1.2 PLANT FAMILIARIZATION

DCPP is located on the central California coast in San Luis Obispo County, approximately 12 miles west-southwest of the city of San Luis Obispo. The plant consists of two separate, but essentially identical units (Unit 1 and Unit 2).

Each unit was built with a four-loop pressurized water reactor nuclear steam supply system furnished by Westinghouse Electric Corporation. The NSSS for each unit is contained within a steel-lined reinforced concrete structure that is capable of withstanding the pressure that might be expected as a result of the most severe design basis loss of coolant accident.

Although the reactors, structures, and all auxiliary equipment are essentially identical for the two units, there is a difference in the thermal power capacities of the reactors. The licensed reactor rating for Unit 1 is 3338 MWt and for Unit 2 is 3411 MWt, with a corresponding net maximum dependable capacity rating of 1073 MWe and 1087 MWe, respectively.

Unit 1 received its full power operating license from the NRC on November 2, 1984, and began commercial operation on May 7, 1985. Unit 2 received its full power operating license on August 26, 1985, and began commercial operation on March 13, 1986.

A detailed description of the plant site, facilities, and safety criteria is documented in the FSAR Update as well as in the DCPRA-1988. Additional discussion of the plant information considered in the external events portion of DCPRA-1993, along with a discussion of the effort to update the pertinent information and a brief description of the plant walk-throughs for the update, are provided in Section 2.4 and other sections of the report. As part of the update process, design changes installed at DCPP since the completion of the LTSP (such as addition of a sixth dedicated diesel generator and reinforcing of block walls) were reviewed and appropriate changes were incorporated in PRA models.

As part of performing the original DCPRA-1988, extensive reviews of plant documentation and numerous plant walkdowns (for seismic fragilities, fire propagation, and other external events) were performed by PG&E personnel and consultants. Additional walkdowns and documentation reviews, as described in this report, were performed as part of the IPEEE effort.

1.3 OVERALL METHODOLOGY

The PRA methods used to assess the external events closely follow the series of analytical tasks and methods developed by PLG and implemented by PLG in performing numerous full-scope PRAs of U.S. and foreign nuclear power plants. The PLG methodology is described in Reference 1-10.

Separate methodologies were used to address the impacts of each external event (such as seismic, fire, and "other") on the plant response to these events. The general methodology for evaluating external events is summarized below. More detailed methodology for each type of hazard is described in Sections 3, 4, and 5. In the PRA, the impacts of the external hazards are integrated into the plant response model, so both external and internal events are considered in the PRA models.

The risk models are developed using PLG's RISKMAN software (Reference 1-11). RISKMAN combines the plant response model event trees to create a single large event tree from initiating event to plant damage state.

The methods for performing the external events assessment for the DCPP LTSP and the IPEEE are consistent with the methods presented in NUREG/CR-2300 (Reference 1-12).

The approach used to assess the external events is as follows:

- The hazard assessment predicts the various hazard levels and estimates the probable frequency of occurrence.

- The fragility assessment predicts the damage to plant equipment and structures that might occur given the hazard level postulated.

- The plant response logic determines the impact on the plant from the combinations of damaged equipment as a result of the initiator and estimates the frequency of these damage states.

- The screening process compares the frequencies of plant states to other external and internal initiated sequences that have similar impact. If the frequency of the plant state is small enough, or significantly less than other initiators, then the hazard can be screened out from further consideration.

Hazards not screened out from further consideration are provided as inputs into the plant model. In the plant model, external initiators or hazards are treated the same way as internal events. The methodology employed in evaluating internal events is documented in Reference 1-4.

The plant model consists of sequences including:

- Initiating events (including external hazards)

- Support system availability given the specific initiating event

- Frontline system response to the specific initiating event

- Operator response to the specific initiating event, including operator recovery actions

This method is referred to as "large event tree linking approach."

1.4 SUMMARY OF MAJOR FINDINGS

This section summarizes the major findings from the external events evaluation of the DCPRA-1993 Update. Fire and seismic events are the only important contributors to core damage risk from external events. "Other" external events contribute a negligible amount to the overall DCPRA core damage frequency. There were no vulnerabilities identified due to external events and there were no cost-effective design changes identified that could significantly reduce overall plant risk. One procedure change (to trip the RCPs in the event a fire in the control room is likely in the judgement of the shift foreman to disable ASW or CCW, and control room evacuation is necessary) is being evaluated as a result of the IPEEE process. It should be noted that the large uncertainty in the results reflect the uncertainty in hazard determination and fragility assessment.

As part of the LTSP, plant changes were made to DCPP that reduced the overall risks from external events. These included: modification of the diesel fuel oil transfer system, addition of a portable engine-driven fuel oil transfer pump, backup fire water cooling to the centrifugal charging pumps (for RCP seal cooling), and addition of spare parts in the

230 kV switchyard. Reinforcement of block walls at Diablo Canyon has decreased the probability that the block walls would be impacted by seismic events.

1.4.1 Core Damage Frequency Results

Seismic and fire initiating events are the only important external events contributing to core damage frequency at DCPP. The DCPRA-1993 external events update estimated a mean core damage frequency due to seismic events of 4.0E-5 per year; the mean core damage frequency due to fire events is estimated to be 2.7E-5 per year. These results do not differ significantly from those previously determined during the LTSP.

The most important seismic sequences are seismic station blackout with the following characteristics:

- Large seismic event that fails 500kV and 230kV power, as well as a primary turbine building shear wall, causing a loss of all vital AC power.

- Large seismic event that fails 500kV and 230kV power, with independent failure of diesel generators.

The fire risks are dominated by fires in the control room and cable spreading rooms:

- Control room fire that disables the CCW system with no tripping of RCPs and no credit for recovery from the hot shutdown panel.

- Control room fire that disables the CCW system with reactor coolant pumps tripped, and failure of recovery from the hot shutdown panel.

- Cable spreading room fire that disables all 4kV vital AC, ASW, or CCW.

1.4.2 Containment Performance Results

The external events impact on containment performance was assessed, although a complete Level 2 evaluation was not performed as part of the IPEEE. In particular, containment performance was evaluated from the following perspectives:

- The containment structure, including the containment itself, penetrations, hatches, piping, and isolation valves all have high seismic capacities. Three failure modes were conservatively assumed to lead to large containment bypasses: steam generator seismic failure leading to containment failure, excessive LOCA leading to containment failure, and containment structural failure.

- The containment heat removal capability is not vulnerable to seismic events. Containment fan cooler units and containment spray systems have relatively high seismic fragilities with HCLPF (high confidence of low probability of failure) values of approximately 2.5g.

- The containment isolation/containment bypass capability was examined. The only containment isolation valve adversely impacted by external events is the RCP seal return line isolation valves, which are motor-operated valves requiring AC power to close. Operator action was credited to close them following station blackout events.

Containment performance for fire initiators was conservatively evaluated and it was determined that the sequences are similar to the internal events. The conclusion is that external events do not pose any unique threat to containment performance, and containment performance is not significantly different than that identified in the IPE report plant damage states (Reference 1-4).

1.4.3 Vulnerability Screening

Vulnerability screening for the IPEEE external events is consistent with the DCPP IPE vulnerability screening process (Reference 1-4). Based on the guidelines presented in Reference 1-13, a vulnerability refers to any component, system, operator action, or accident sequence that contributes more than 50 percent to the core damage frequency or has a frequency that exceeds 1×10^{-4} per year. No vulnerabilities due to external events were identified based on these screening guidelines.

For containment performance, any containment bypass or large early release that exceeds 1×10^{-5} per year is considered a containment performance vulnerability. No containment performance vulnerabilities were identified due to external events using these screening guidelines.

1.5 REFERENCES

1-1. U.S. Nuclear Regulatory Commission, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities," Generic Letter 88-20, Supplement 4, June 28, 1991.

1-2. PLG, Inc., "Diablo Canyon Probabilistic Risk Assessment," prepared for Pacific Gas and Electric Company, PLG-0637, July 1988.

1-3. Pacific Gas and Electric Company, "Long Term Seismic Program Final Report," PG&E Letter No. DCL-88-192, July 31, 1988.

1-4. Pacific Gas and Electric Company, "Response to Generic Letter 88-20, Individual Plant Examination," PG&E Letter No. DCL-92-087, April 14, 1992.

1-5. U.S. Nuclear Regulatory Commission, "Staff Evaluation of the Diablo Canyon Power Plant (DCPP) Units 1 and 2, Individual Plant Examination (IPE) - Internal Events Submittal," June 30, 1993.

1-6. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report," NUREG-0675, Supplement No. 34, Docket Nos. 50-275 and 50-323, June 1991.

1-7. Bozoki, G. et al., "Review of the Diablo Canyon Probabilistic Risk Assessment," NUREG/CR-5726 (DRAFT), June 1991.

1-8. Pacific Gas and Electric Company, "Results of the Advisory Committee on Reactor Safeguards Meeting on the Diablo Canyon Long Term Seismic Program," PG&E Letter No. LSTP 1.3.1, 1.3.2, Log 91-413, Chron No. 179295, October 21, 1991.

1-9. U.S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

1-10. Kaplan, S., G. Apostolakis, B.J. Garrick, D.C. Bley, and K. Woodward, "Methodology for Probabilistic Risk Assessment of Nuclear Power Plants," PLG-0239, June 1981.

1-11. PLG, Inc., "RISKMAN PRA Workstation Software."

1-12. U.S. Nuclear Regulatory Commission, "PRA Procedures Guide," NUREG/CR-2300, 1983.

1-13. NUMARC, "Transmittal of NUMARC Severe Accident Issue Closure Guidelines," NUMARC Report 91-04, February 21, 1992.

2. EXAMINATION DESCRIPTION

2.1 INTRODUCTION

The objectives for the Individual Plant Examination of External Events (IPEEE) listed in Section 1.1 were accomplished by the completion of a Level 1 (including containment performance assessment), external events probabilistic risk assessment (PRA) for Diablo Canyon Power Plant (DCPP). A Level 1 PRA, as defined in Reference 2-1, considers the performance of the plant systems to the extent needed to resolve scenarios to the point of success or core damage. The Level 1 PRA for the Individual Plant Examination of External Events (IPEEE) includes the performance of containment, containment isolation, containment bypass, and containment heat removal systems.

This report documents Pacific Gas and Electric Company's (PG&E's) work in accordance with Nuclear Regulatory Commission (NRC) Generic Letter 88-20, Supplement 4 (Reference 2-2) and NUREG-1407 (Reference 2-3), that requested each utility to perform an IPEEE. The analyses summarized in this report represent an update of the external events portion of the Diablo Canyon Probabilistic Risk Assessment (DCPRA-1988) (Reference 2-4) performed as part of the DCPP Long Term Seismic Program (LTSP) (Reference 2-5).

The LTSP reevaluated the seismic bases for the 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 Element 4 of 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 by a team of consultants and utility personnel lead by PLG, Inc. The DCPRA-1988 is a full-scope, Level 1 PRA that evaluated the probable frequency of experiencing reactor and plant damage as the result of both internal, as well as external initiating events. While it was performed for only 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 SSER No. 34 for NUREG-0675 (Reference 2-6) in June 1991, accepting the DCPRA-1988. The DCPRA-1988 was reviewed for the NRC primarily by Brookhaven National Laboratory. This review is documented in NUREG/CR-5726 (Reference 2-7). In addition, the Advisory Committee on Reactor Safeguards accepted the NRC's review of the LTSP and DCPRA-1988 and concluded that DCPP "can be operated without undue risk to the health and safety of the public" (Reference 2-8).

As documented in this report, the DCPRA-1988 has been updated to represent the as-built operating condition of DCPP through March 1993, and includes a containment performance assessment. The latest update of the external events portion of the DCPRA is referred to as DCPRA-1993. This report largely follows the format suggested in Table C.1 of NUREG-1407.

2.2 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

The NRC Generic Letter 88-20, Supplement 4, issued on June 28, 1991, requested that each utility perform an IPEEE for severe accident vulnerabilities and that the results of the examination be submitted to the NRC. NUREG-1407 provides further guidance on the performance of the IPEEE. PG&E committed in its program plan submitted to the NRC (Reference 2-9) to complete and submit the DCPP IPEEE report to the NRC by June 28, 1994.

Generic Letter 88-20, Supplement 4, requests "that each licensee perform an individual plant examination of external events to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee-determined improvements and corrective actions to the Commission." Specifically, the purpose of the Generic Letter is for each licensee (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant under full power operating conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage and radioactive material release, and (4) if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

To assure the technical adequacy of the IPEEE and the validation of the results, as outlined in other portions of the report, PG&E updated the DCPRA-1988 to:

- Ensure the plant reflects current plant design and operation. Updated design and operational data were used through March, 1993 and December, 1991, respectively.

- Ensure technical adequacy, the IPEEE report was reviewed by a PG&E peer group, including personnel from Operations, Maintenance, Plant Engineering, Reliability Engineering, Design Engineering, Safety Analysis, Licensing, Training, and Emergency Services. The IPEEE report was also independently reviewed by PLG, the consultant responsible for the original LTSP PRA. Results of the IPEEE evaluations were compared with other external events PRAs, including Seabrook's, to assure some consistency and validity of results.

- Further ensure technical adequacy of the results, uncertainty calculations were performed for the seismic and fire PRAs. Importance calculations were also performed. Finally, some sensitivity analysis was done to further understand and validate the reasonableness of the results.

2.3 GENERAL METHODOLOGY

This section summarizes the technical approach and methods used in the development of the external events portion of the DCPRA-1993.

The overall PRA method closely follows the series of analytical tasks and methods PLG has developed and implemented in performing more than 20 full-scope and phased PRAs

of U.S. and foreign nuclear power plants. A description of the theoretical and mathematical bases for the approach is given in the PLG methodology document (Reference 2-10). The method of analysis used to complete the external events portion of the DCPRA-1993 is virtually identical to the method used in the DCPRA-1988, which was used for the LTSP. Since the completion of the LTSP, PLG's PRA software, RISKMAN (Reference 2-11) has been enhanced significantly, and has been installed on PCs at PG&E. However, the PLG methodology as described in Reference 2-10 remains largely unchanged.

2.3.1 Seismic PRA Methodology

For the LTSP, as well as for the IPEEE, a seismic PRA was developed and evaluated. A detailed discussion of the seismic PRA methodology is presented in Section 3.0. A brief overview is provided in this section.

The main elements of a seismic PRA are the seismic hazard evaluation, structure and component fragility analysis, plant logic analysis, and event tree quantification. A summary of each of these elements of the risk assessment is now provided.

The seismic hazard evaluation provides DCPP-specific seismic hazard levels and the probable frequency of occurrence. These are reduced to six seismic "initiating events", each with a unique probable frequency of occurrence and a corresponding uncertainty distribution.

The structure and component fragility analysis provides unique fragility curves, defined by the median ground spectral acceleration capacities times the product of randomness and uncertainty variables.

The seismic plant logic analysis determines the consequence of various structural and component failures. This logic is added to the event trees used in the general transient event trees developed for the internal events PRA, as used in the Individual Plant Examination (IPE) report (Reference 2-9). The event trees used for general transients were expanded and modified to account for seismic events. For example, a seismic component and structure event tree was added to the general transient event trees to provide a means to evaluate and map seismic failures.

Quantification of results provides both point estimate core damage frequencies and uncertainty distributions for the six discrete seismic acceleration levels evaluated as part of the DCPP seismic PRA. The quantification identifies core damage sequences from seismic failures, as well as core damage sequences following seismic events that are a result of combinations of seismic and non-seismic failures. Included in the quantification are the identification of the important seismic core damage sequences, core damage frequency calculations at the various seismic levels, importance calculations, and plant damage state binning.

2.3.2 Fire PRA Methodology

The general methodology for the fire PRA is the spatial interactions analysis. The purpose of the spatial interaction analysis is to identify those physical interactions involving plant environmental hazards that can cause an initiating event and intersystem dependencies that would contribute significantly to risk by using information on spatial commonalities.

Evaluation of fire events follows the scenario approach in which a large list of possible scenarios is postulated. Scenario refers to a chain of events starting with the initiation of a fire due to ignition of a combustible, the fire growth, the ignition of other combustibles, fire detection, fire suppression, and the fire impact on plant equipment.

Final results of the spatial interaction portion of the fire PRA include an estimation of the scenario frequency, the extent of impact to plant systems of each scenario, and a screening out of all scenarios judged to be of insignificant importance.

Fire scenarios not screened out were grouped into similar scenarios, with their impacts mapped in the general transient event trees used in the IPE. Subsequently, the event trees were quantified to provide the core damage frequencies from each of the scenarios. For control room fires and cable room fires, separate event trees were developed because of the unique impacts and operator actions from these types of fires. As with the seismic PRA, the quantification includes core damage sequences resulting from fire induced failures, and some core damage sequences following fire events are a result of combinations of fire and non-fire failures. Included in the quantification are all of the important fire core damage sequences, core damage frequency calculations for the various fire scenarios or initiating events, importance calculations, and some plant damage state binning.

2.3.3 Other External Events Analyses Methodology

There is a long list of other potential sources of hazards external to the plant systems that were considered as candidates for external initiating event scenarios. Among these scenarios, there were several that could not be screened out solely based on a high level quantitative and qualitative screening. For these external hazards, a more detailed analysis was performed to determine the likelihood of the event initiator and the potential plant impacts and consequences. The events not screened out in the LTSP that had detailed analysis are:

- Aircraft crash and other falling objects
- External fire
- Turbine missiles
- Ship impact
- Hazardous chemicals
- Hurricane winds and tornadoes
- External flooding

The basic approach for all these events is to first perform a conservative screening in the area of initiator frequency, as well as for the conditional likelihood of core damage. If such conservative screening results in core damage frequencies smaller than 10^{-7} per year, no further detailed analysis was performed. If the initiator could not be screened out, then a more realistic estimate of initiator frequency and/or conditional core damage probability was made based on a more detailed analysis of the hazard source and the event scenario involved.

2.4 INFORMATION ASSEMBLY

Much of the plant layout and containment building information used in the external events portion of the DCPRA-1993 is contained in the FSAR Update (Reference 2-12). The model for the DCPRA-1988, developed for the LTSP, was based on information gathered from the following sources:

- DCPP FSAR Update
- Operator Information Manuals
- DCPP System Description Documents
- Licensee Event Reports
- Surveillance Test Procedures
- Maintenance Procedures
- Operating Procedures
- Emergency Operating Procedures
- Information from Operator Surveys (Human Action Analysis)
- Plant Walk-throughs
- Appendix R report

Component failure rates, initiating event frequencies, and maintenance unavailability data used for the DCPRA were based on actual DCPP operating experience, supplemented with applicable industry experience.

To update the external events portion of the DCPRA, and to confirm that the PRA represents the as-built, as-operated plant, PG&E personnel reviewed the above sources of information and also reviewed applicable new documentation to identify any changes in the design and operation of DCPP since 1988 that could impact the external events PRA. The new documents reviewed included:

- DCPP FSAR Update
- Licensee event reports
- Significant operating experience reports
- Design change packages
- Procedure changes
- Technical specification changes
- Maintenance records
- Component clearance records
- Design criteria memorandum
- Miscellaneous plant layout drawings

- Appendix R Update

The review of existing documentation was supplemented with discussions with DCPP operations, design engineering, and plant engineering personnel to further ensure that the PRA reflects the current as-built, as-operated plant.

Plant walkdowns were conducted to further verify the as-built, as-operated plant and to familiarize analysts with specific plant details needed for the PRA model. Seismic, fire, and other external events walkdowns were conducted. These walkdowns are summarized below:

- Date: 3/16/94 - 3/17/94
  Scope: Fire walkdown, including Sandia Fire Risk Scoping Study issues

  Participants: Lead Fire PRA Analyst
  PRA Group Supervisor
  Fire Protection Group Engineers
  On-site Fire Protection Engineer
  PG&E Lead Seismic PRA analyst
  Seismic Interaction Specialist
  DCPP Fire Marshall

- Date: 5/3/94

  Scope: Seismic walkdown

  Participants: Lead Seismic PRA Analyst
  Lead Fire PRA Analyst
  Equipment Qualification Engineer
  Civil Engineering Engineer
  On-site Equipment Qualification Engineer
  PRA Group Supervisor

- Date: 3/10/94
  Scope: External events walkdown

  Participants: PRA Analyst
  PRA Group Supervisor
  Independent Safety Engineering Group Engineer
  PLG PRA Consultant

2.5 REFERENCES

2-1. American Nuclear Society and Institute of Electrical and Electronics Engineers, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," sponsored by the U.S. Nuclear Regulatory

Commission and the Electric Power Research Institute, NUREG/CR-300, April 1983.

2-2. U.S. Nuclear Regulatory Commission, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities," Generic Letter 88-20, Supplement 4, June 28, 1991.

2-3. U.S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

2-4. PLG, Inc., "Diablo Canyon Probabilistic Risk Assessment," prepared for Pacific Gas and Electric Company, PLG-0637, July 1988.

2-5. Pacific Gas and Electric Company, "Long Term Seismic Program Final Report," PG&E Letter No. DCL-88-192, July 31, 1988.

2-6. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report," NUREG-0675, Supplement No. 34, Docket Nos. 50-275 and 50-323, June 1991.

2-7. Bozoki, G. et al., "Review of the Diablo Canyon Probabilistic Risk Assessment," NUREG/CR-5726 (DRAFT), June 1991.

2-8. Pacific Gas and Electric Company, "Results of the Advisory Committee on Reactor Safeguards Meeting on the Diablo Canyon Long Term Seismic Program," PG&E Letter No. LTSP 1.3.1, 1.3.2, Log 91-413, Chron. No. 179295, October 21, 1991.

2-9. Pacific Gas and Electric Company, "Response to Generic Letter 88-20, Individual Plant Examination," PG&E Letter No. DCL-92-087, April 14, 1992.

2-10. Kaplan, S., G. Apostolakis, B.J. Garrick, D.C. Bley, and K. Woodward, "Methodology for Probabilistic Risk Assessment of Nuclear Power Plants," PLG-0239, June 1981.

2-11. PLG, Inc., "RISKMAN PRA Workstation Software."

2-12. Diablo Canyon Power Plant FSAR Update.

3. SEISMIC ANALYSIS

3.0 METHODOLOGY SELECTION

Seismic events can initiate potential accident scenarios, such as causing the reactor to trip and causing the plant structures or plant components to fail or fail to function. Thus, it is important to evaluate the frequency with which various levels of ground accelerations might occur, the likelihood that important plant components could fail at these accelerations, and the plant damage state consequences of component and/or structure failure combinations. This evaluation must be done in an orderly process; one method for accomplishing this is a seismic probabilistic risk assessment (PRA), which was used in the Diablo Canyon IPEEE seismic analysis. A detailed description of the methodology for performing the seismic PRA for the Diablo Canyon Power Plant (DCPP) is contained in References 3-1 and 3-2. The seismic PRA consists of four main steps.

1. **Seismic Hazard Analysis**. The seismic hazard analysis provides the frequencies of ground motions of various sizes at the site.

2. **Fragility Analysis**. The fragility analysis provides the seismically initiated ground accelerations at which plant structures and components are predicted to fail.

3. **Plant Logic Analysis**. The plant logic analysis provides the models that include the seismically induced events that may cause one or more different classes of initiating events and one or more failures of components or systems needed to respond to the initiating event. The plant logic analysis also considers nonseismic failures that can combine with seismically induced failures to produce an accident sequence.

4. **Quantification**. Quantification and assembly of the seismicity, fragilities, and plant logic first provides point estimates of the frequencies of core damage and of various plant damage states (PDS) that might result from seismic initiating events. A comparison is made of various plant damage state point estimate frequencies as contributors to total core damage frequency. Finally, a calculation is made of the probability distribution of core damage frequencies from seismic initiators. The uncertainty or probability distribution is made in lieu of a sensitivity study.

Sections 3.0.1 through 3.0.5 provide more detail on the methodologies used in the DCPP seismic PRA.

3.0.1 Seismic Hazard Methodology

The seismic hazards analysis includes consideration of all seismic sources that can affect ground motions at the DCPP site. The EPRI and LLNL mean hazard curves described in References 3-3 and 3-4 do not apply to western plants such as Diablo Canyon. Plant-specific seismic hazard curves were developed as part of the Long Term Seismic

Program (LTSP). The methods and procedures used to calculate seismic hazards for the DCPP site are documented in the LTSP Final Report (Reference 3-2).

Many factors are considered in the performance of the seismic hazard analysis. Factors considered in the DCPP seismic hazard evaluation include:

- Identification of the seismic sources in the region of interest around the DCPP site on the basis of observed seismic activity and the known tectonic regime.

- Evaluation of the earthquake history of the region to assess the recurrence frequency of earthquakes of different magnitudes to establish frequency-magnitude relationships.

- Specification of upper bound magnitudes of earthquakes for the region of interest.

- Development of attenuation relationships to estimate the intensity of earthquake-induced ground motions. The attenuation effects are a function of source distance and media characteristics between the sources and the site.

- Integration of these elements into a seismic hazard analysis. Considering alternative interpretations of historical information, geology, and empirical relationships, a number of possible combinations of source locations, maximum magnitudes, frequency-magnitude relationships, and attenuation relationships are evaluated, each combination with a different degree of certainty.

The product of the seismic hazard analysis is a set of seismic hazard curves, each having a probability of representing the true curve. The parameter of the ordinate is the annual exceedance frequency and the parameter of the abscissa is the average 5 percent damped spectral acceleration over the 3-8.5 Hz frequency range. The spectral acceleration is approximately two times the peak ground acceleration. The hazard curves for Diablo Canyon are presented in Section 3.1.1 of this report.

It is necessary for the hazard curves used in the seismic analysis to be anchored to the same parameter as the fragilities; in the case of the DCPP seismic PRA, the average 5 percent damped spectral acceleration over the 3-8.5 Hz frequency range was chosen. The seismic hazard curves obtained from the hazard analysis reflect the annual exceedance frequency versus average 5 percent damped spectral acceleration over the 3-8.5 Hz frequency range.

3.0.2 Fragility Methodology

The seismic fragility analysis is performed to predict the fraction of earthquakes of each given acceleration level that would cause specific plant buildings and equipment to fail. The purpose of the fragility analysis was to carefully evaluate each of the structures and components that are included in the risk model to define those failure modes that have the lowest seismic capacities and which, therefore, may constitute the most important or dominant contributors to DCPP seismic risk. The methods for determining fragilities are

described in detail in References 3-1 and 3-2. In the DCPP seismic PRA, this failure fraction (or fragility) is anchored to the same ground motion parameter, i.e. the average 5 percent damped spectral acceleration over the 3-8.5 Hz range, to be consistent with the basis of the seismic hazard analysis.

The fragility evaluations incorporated appropriate aspects of the various DCPP LTSP studies, including the site-specific geotechnical and soil/structure interaction investigations, the median in-structure spectra evaluation, and the structural response variability investigation.

The definition of failure is vitally important to the development of fragilities for structures and equipment. The fragility description of structures consisted of the identification and evaluation of controlling failure modes associated with the important structures. Similarly, the fragility description of mechanical and electrical equipment consisted of the identification and evaluation of controlling failure modes related to elements of the major safe shutdown plant systems. In every case, the fragility analyses were based upon plant-specific structure or component seismic qualification analyses directly related to elements in place at the DCPP.

Design Class I structure failures were defined in terms of inelastic lateral drifts generally corresponding to the onset of significant strength degradation of major structural elements. The exception is the containment building where lateral drifts were limited to lower levels consistent with the need of the containment building to remain pressure-tight. Equipment housed in the important structures was assumed to fail when the structure reached lateral drifts corresponding to the onset of significant strength degradation or severe distress. The fragility estimates for structures correspond to distress levels short of partial or total collapse, but are treated as total collapse in the probabilistic risk assessment. The degree of margin between the onset of significant strength degradation and total collapse is uncertain and difficult to estimate. However, the benefits of this margin, which in most cases is likely to be large, have been conservatively ignored.

Piping, electrical, mechanical, and electro-mechanical equipment vital to safe shutdown of the plant or mitigation of an accident were considered to fail when it was judged they were no longer able to perform their designated function. For mechanical equipment, the fragility definition represents failure to function, loss of anchorage, or rupture of the pressure boundary. For electrical equipment, the fragility represents loss of function due to acceleration-sensitive failure (for example, relay chatter), or loss of function due to structural failure of the cabinet, anchorage, or internals. For ductile systems such as piping, HVAC ducting, and electrical conduits, fragility represents crimping, choking of flow, or rupture due to failure of the supports.

The fragility of a structure or a component is defined as the conditional frequency of its failure for a given value of the ground-motion parameter (for example, spectral acceleration). Thus, the fragility evaluation is based on the estimation of the median ground spectral acceleration value for which the seismic response of a given structure or component exceeds its capacity, resulting in failure. Because there are many sources of variability in the estimation of the median ground spectral acceleration capacity, the

component fragility is described by a family of fragility curves. Figure 3-1 depicts a set of fragility curves. The spectral acceleration at any point within the family of fragility curves is computed as:

$$\bar{S}_a = \overset{V}{\bar{S}_a}\, e^{(C_1\beta_R + C_2\beta_U)}$$

where C_1 and C_2 are the statistical constants associated with the failure fraction and confidence level of interest (Figure 3-1); where

$\bar{S}_a$ is the spectral acceleration capacity,

$\overset{V}{\bar{S}_a}$ is the median spectral acceleration capacity,

β_R represents the randomness variability; and

β_U represents the uncertainty variability.

3.0.3 Plant Logic Analysis

The plant logic analysis determines the consequences of various structural and plant component failures. The approach used in the risk assessment relies on the logic expressed by the event trees used in the internal initiating events analyses. In these trees, the event tree top events are system functions or recovery actions. The likelihood of success or failure in moving along the various pathways through the top events leads to the frequencies of the various PDS. The sum of the plant damage state frequencies is the frequency of core damage.

The plant logic analysis process is represented in graphical form in Figure 3-2. The first step is to identify the components whose seismic failure could initiate an accident scenario. This identifies the event tree, or trees (initially developed in an internal events analysis), that can be used in the seismic analysis. For example, if failure of a number of components will cause the reactor and/or turbine to trip, a general transient event tree would contain all the possible scenarios of interest. Other component failures could generate other initiating events and also require the use of additional event trees. Once the appropriate event tree(s) are identified, it is sometimes necessary to expand it for use in a seismic analysis by adding the passive components (such as buildings, cable trays, conduits, and HVAC ducts) typically not included in the trees developed for the internal events analysis. These passive components have been added to a seismic component and structure event tree, which is put at the beginning of the linked event tree model.

Seismic failures of active components are also included in this seismic component and structure event tree.

It is necessary to develop a table that relates the seismic failure of each component to the unavailability of one or more systems reflected in the top events in each event tree. Also, the seismic failure of more than one component could fail the same top event. Table 3-7 identifies these failure impacts, which serve as the basis for the seismic event tree logic.

3.0.4 Point Estimate Quantification

The initial quantification is one in which point estimates are made of the frequency of each seismically initiated scenario using the mean values from the seismic fragilities and hazard curves. This enables the identification of significant scenarios that dominate the frequency of seismic initiated core damage. This process also enables a comparison between seismic and nonseismic initiated scenario frequencies to determine the significance of seismic events to the total risk.

The initial point estimate process is begun by selecting discrete accelerations in selected ranges, and then determining the mean annual frequency at each of these acceleration levels. Six ranges with discrete values for each were selected (these are the same as in Reference 3-2).

As seen in Figure 3-2, a table is developed to indicate the calculated mean conditional failure fraction of each top event involving seismic failure modes at each of the discrete accelerations. These failure fractions represent either a single component or multiple components. In addition to the seismic failures, components might also be unavailable because of nonseismic causes (random failures, maintenance, testing, etc.). The mean conditional nonseismic unavailabilities for top events that do not contain seismic failure modes (which are constant over all accelerations) are considered separately in the system models. The top events without seismic failure modes may be impacted, however, by the failure of top events that do maintain seismic failure modes. These seismic failure impacts are accounted for in the split fraction assignment rules. For example, if offsite power is lost seismically, top event OG for nonseismic failures of offsite power would be guaranteed failed. The top events containing seismic failure modes are nearly all included in a separate seismic component and structure event tree. In this way, both the seismic and nonseismic failure modes are accounted for when quantifying the seismic event trees.

As the final step, the seismic event trees are quantified at the six discrete seismic acceleration levels.

3.0.5 Uncertainty Quantification

The results of quantifying the linked event tree seismic model are the mean frequencies of the damage states and lists of the highest frequency scenarios contributing to these damage states. To generate an uncertainty distribution (rather than a point estimate) for a plant damage state frequency, the sequences that contribute most significantly to that

damage state were requantified using distributions for the input variables (i.e., initiating events and split fractions) in Monte Carlo simulation. These distributions represent uncertainty resulting from seismic hazard, component fragility, and component failure rates. Correlation of the initiating event frequencies and the split fractions at the different initiator acceleration ranges was considered.

3.1 SEISMIC PRA ANALYSIS

3.1.1 Hazard Analysis

As mentioned in the methodology section (Section 3.0.1), the seismic hazard analysis provides the probabilistic representation of the seismic ground motion at the DCPP site for use in the PRA. The EPRI and LLNL mean hazard curves described in References 3-3 and 3-4 do not apply to Diablo Canyon. Plant-specific hazard curves, were developed for the LTSP (Reference 3-7). These seismic hazard curves considered the seismic sources that could affect the site. The seismic sources included the Hosgri, Los Osos, San Luis Bay, Santa Lucia Bank, West Huasna, Lompoc, Rinconada, Nacimiento, and San Andreas faults. The characteristics of the seismic sources, maximum magnitude distribution, rate of earthquake occurrence, and attenuation of ground motion were used to develop a probabilistic representation of earthquake ground motions expected at the site. A detailed explanation of the seismic hazard analysis is contained in Reference 3-2.

It is necessary for the hazard curves used in the seismic analysis to be anchored to the same parameter as the fragilities; in this case, average spectral acceleration. Table 3-1 and Figure 3-3 present the hazard curves used in the seismic analysis for the PRA quantification process. They reflect the annual exceedance frequency versus peak spectral acceleration. Table 3-2 and Figure 3-4 reflect mean values of frequencies at discrete accelerations.

The NRC review of the seismic hazard analysis is summarized in Reference 3-8. The NRC review of the DCPP seismic hazard analysis concluded that:

> "The seismic hazard analysis provided a reasonable probabilistic representation of the earthquake ground motions at the site. The Hosgri fault zone was found to dominate the seismic hazard at the site. The Los Osos and San Luis Bay faults each contribute only a few percent to the total hazard. Relative contributions to the total hazard from the other faults are insignificant. Sensitivity studies showed the important parameters are slip rate, maximum magnitude, and ground-motion attenuation." (Reference 3-8, Section 23.4.3.4)

As part of the IPEEE effort, it was verified that the seismic hazard curves developed as part of the LTSP were appropriate for use in the IPEEE. Reference 3-9 documents the validity of the LTSP seismic hazard curves for use in the IPEEE.

Soil liquefaction is not an issue at DCPP; DCPP is a rock site.

Appendix C, Section C.2.1.2 of NUREG-1407 (Reference 3-4) requests documentation of the following information: "Also, if an upper bound cutoff to ground motion of less than 1.5g peak ground acceleration is assumed, the results of sensitivity studies to determine whether the cutoff affected the overall results and the delineation and ranking of seismic sequences." As shown in Table 3-1, the maximum average 5 percent damped spectral acceleration over the 3-8.5 Hz frequency range considered in the hazards analysis is 4.0g. Using a conversion factor of 2.34, a 4.0g spectral acceleration converts to a peak ground acceleration of 1.7g. Since the peak ground acceleration exceeds 1.5g, no sensitivity studies are required to determine whether the cutoff affected the overall results and delineation and ranking of seismic sequences. Additionally, spectral accelerations greater than 4.0g contribute less than one percent to seismic core damage, and can be ignored.

The seismic hazard curves were used to develop seismic initiating events. Seismic initiating events for six discrete acceleration levels were defined. The six discrete spectral acceleration levels are consistent with the LTSP, and are defined on Page 6-204 of Reference 3-2; the seismic initiating events frequencies are obtained from Table J-35 of Reference 3-1. The seismic initiating event frequencies (point estimate) are listed in Table 3-5.

The six seismic initiating events and their frequencies are used in the point estimate quantification of the seismic PRA. The uncertainty analysis utilizes the full set of 8 hazard curves presented in Figure 3-3.

3.1.2 Review of Plant Information and Walkdown

As part of the plant design and construction, extensive plant walkdowns were performed to determine structural and equipment seismic capability and detailed documentation of the walkdowns was developed. Additionally, as part of the LTSP, a seismic fragility plant walkdown was conducted by fragility and PRA analysts. The walkdown included an examination of Design Class II items that could lead to failure of Design Class I items (systems interaction program). No Design Class II items were found that could fail and put a safety-related component out of service.

An additional plant walkdown was conducted by NRC Staff and consultants as part of the LTSP in March 1988. The walkdown emphasized the seismic risk-important components and structures, and primarily focused on identifying potential failure modes.

A confirmatory IPEEE seismic plant walkdown was performed on May 3, 1994 (Reference 3-10). The primary purposes of the walkdown were:

- Understand failure modes and fragilities of lowest capacity structures and components
- Walkdown components/structures that have been significantly modified since completion of LTSP (for example, safety-related block walls, sixth diesel generator, steam generator blowdown modifications)

- Review potential for seismic/fire interactions
- Review potential for seismically induced floods and possible impact
- Review containment performance/containment integrity issues
- Provide confirmation of the as-built, as-operated plant

The following personnel were involved in the seismic walkdown:

PRA Senior Engineer

PRA - IPEEE Seismic Lead Engineer

PRA - IPEEE Fire Lead Engineer

Civil Engineer

Equipment Qualification Engineer

A walkdown checklist was developed, partly based on the criteria identified in the EPRI seismic margin document (Reference 3-11). The walkdown confirmed the reasonableness of the identified failure modes, as well as the consequences of failure.

To confirm the as-built, as-operated plant for the IPEEE, design changes incorporated at DCPP since the completion of the LTSP were reviewed for their impact on the seismic PRA model. The conclusions of the design change review are documented in Reference 3-12. Changes identified since the LTSP that could impact the seismic PRA include the following:

Safety-related block wall modifications (Reference 3-15)

Sixth diesel generator addition

Steam generator blowdown modification impact on turbine building structural response

To further assure the seismic PRA reflects the as-built, as-operated plant, a design change procedure (Reference 3-13) has been implemented that requires additions and modifications, as listed in Table 3-3 and which are described in the NRC's SSER No. 34 (Reference 3-8), to be reviewed to verify that the plant high-confidence-of-low-probability-of-failure (HCLPF) values remain acceptable.

3.1.3 Analysis of Plant System and Structure Response

The seismic PRA provides a means to quantify the seismic risk of operating the plant. The main elements of a seismic risk analysis are the seismic hazard evaluation, structure

and component fragility analysis, plant logic analysis, and event tree quantification. In this section, the plant logic analysis, as well as the quantification process are described.

The seismic plant logic analysis determines the consequences of various structure and component failures. The logic developed for the DCPP PRA general transient event trees (Reference 3-16) provides the basis for the development of the seismic event trees.

The plant logic analysis is represented in graphical form in Figure 3-2. First, the components whose seismic failure could initiate an accident scenario are defined. This defines the event trees that were initially developed in the internal events analysis that can be modified for the seismic analysis. For example, if a seismic event causes loss of the 500 kV and 230 Kv lines, a seismically induced loss of offsite power event would result, and the general transient event trees would contain the appropriate logic. Once the appropriate event trees are identified, it is necessary to add seismic failure modes to the split fractions, or to add them as separate top events. Also, some passive components, typically not modeled in the internal events analysis (such as buildings, cable trays, conduits, and HVAC ducts), need to be added to the seismic event trees.

3.1.3.1 Seismic Event Trees

The following seismic event trees were used in the seismic analysis. All the event trees, except the seismic component and structure event tree, were modified from the general transient event trees.

These event trees are shown in Figures 3-5 to 3-9.

Seismic Component and Structure Event Tree
Seismic Electric Power (Reference 3-14)
Seismic Mechanical Support (Reference 3-14)
Seismic General Transient (Reference 3-16)
Seismic Late Tree (Reference 3-16 and 3-17)

3.1.3.2 Seismic Top Events

Most of the top events of these trees are the same as those described in the internal event analysis of the Individual Plant Examination (Reference 3-20). However, all the top events in the seismic component and structure event tree are new and several other top events in the other event trees have been changed or were added. The new top events are described below. The relationship of the new seismic top events to component failure is provided in Table 3.7.

The new top events modeled in the seismic component and structure event tree are as follows:

SOP - Seismic Loss of Offsite Power. This top event represents the loss of all offsite power and is based on the 230 Kv switchyard seismic fragility, which is significantly

stronger than the 500 Kv switchyard seismic fragility. Following any plant trip, the emergency AC buses are switched from the 500 Kv to the 230 Kv source of offsite power.

SDC - Seismic Loss of 125V DC Power. This top event represents the seismic failure of 125V DC power, which has similar impacts to those of the internal events analysis.

STRUT - Turbine Building Strut. This top event represents the seismic failure of a turbine building structural component that can impact the failure fragilities for other components within the turbine building.

SACSS - Seismic All 4 Kv Vital AC Power/Strut Success. This top event represents the seismic failure of all 4kV vital power conditional on the turbine building strut not seismically failing.

SACSF - Seismic All 4 Kv Vital AC Power/Strut Failure. This top event represents the seismic failure of all 4 kV vital power conditional on seismic failure of the turbine building strut.

SDG - Seismic All Six Diesel Generators. This top event represents the seismic failure of all six diesel generators. There are three dedicated diesel generators for each unit.

SFO - Seismic Fuel Oil Transfer. This top event represents the seismic failure of the fuel oil transfer system, which has similar impacts to those of the internal events analysis.

SVI - Seismic All Four Vital Instrument Channels. This top event represents the seismic failure of all four vital instrument channels, which has similar impacts to those of the internal events analysis.

SRT - Seismic Reactor Trip. This top event represents the portion of the seismic failure of reactor trip due to the seismic damage of reactor internals preventing rod insertion.

SPT - Seismic Partial Reactor Trip. This top event represents the portion of the seismic failure of reactor trip due to switchgear seismic failure. If offsite power is also lost seismically, this fragility has no impact on the reactor trip success probability.

SCV - Seismic Control Room Ventilation. This top event represents the seismic failure of control room ventilation, which has similar impacts to those of the internal events analysis.

SCB - Seismic Component Cooling Water (CCW) Bypass. This top event represents the failure of containment integrity due to failure of the CCW connection to the containment fan cooler units inside containment and permits a potential release path from the containment through the CCW system.

SCC - Seismic Component Cooling Water. This top event represents the seismic failure of CCW (except for failure modes modeled in top event SCB), which has similar impacts to those of the internal events analysis.

SAS - Seismic Auxiliary Saltwater. This top event represents the seismic failure of auxiliary saltwater which has similar impacts to those of the internal events analysis.

SSV - Seismic 480V Switchgear Ventilation. This top event represents the seismic failure of 480V switchgear ventilation, which has similar impacts to those of the internal events analysis.

SSG - Seismic Steam Generators. This top event represents the seismic failure of the steam generator supports and postulated failure of the reactor coolant system and steam connecting piping. Failure of this top event is modeled as leading to core damage. The top event failure also is modeled as failing containment because it results in high containment internal pressure.

SRW - Seismic Refueling Water Storage Tank. This top event represents the seismic failure of the refueling water storage tank, which has similar impacts to those of the internal events analysis.

SPR - Seismic Pressurized Relief/Small LOCA. This top event represents the seismic failure of the power-operated relief valves (PORVs), which has similar impacts to those of the internal events analysis, i.e., the PORVs are assumed to lift and failure to reseat resulting in a small LOCA.

SSE - Seismic RCP Seal Integrity. This top event represents the seismic failure of the reactor coolant pump (RCP) seal integrity which has similar impacts to those of the internal events analysis. The RCPs may fail seismically in a manner that results in a loss of seal integrity.

SCH - Seismic Centrifugal Charging Pumps. This top event represents the seismic failure of the centrifugal charging pumps, which has similar impacts to those of the internal events analysis.

SAW - Seismic Auxiliary Feedwater. This top event represents the seismic failure of the auxiliary feedwater (AFW) steam driven pump and piping, which is conservatively assumed to have a similar impact to auxiliary feedwater failure in the internal event analysis.

SFC - Seismic Containment Fan Cooler Units. This top event represents the seismic failure of the containment fan cooler units, which has similar impacts to those of the internal events analysis.

SCS - Seismic Partial Containment Spray. This top event represents the seismic failure of one containment spray pump resulting from failure of the spray additive tank.

SCP - Seismic Large Containment Failure. This top event represents the seismic failure of the containment structure creating a large opening (greater than 3-inch diameter) that has similar impacts to those of the internal events analysis.

SSH - Seismic Containment Spray Header Failure. This top event represents the seismic failure of the containment spray headers that fails the containment spray function.

The new top events added to the internal events general transient event tree to create the seismic event trees are as follows:

AT - Reactor Trip. This top event is a switch that reflects the state of reactor trip from both seismic failure modes (top events SRT, SPT) and nonseismic failures; i.e., top event RT in the support model. For seismic events, failure of reactor trip is conservatively assumed to result in core damage.

SCT - Seismic Relay Chatter. This top event represents the potential occurrence of relay chatter resulting in the interruption of all emergency AC power.

TD - Turbine-Driven AFW Pump. This top event is guaranteed failed if top event SAW is failed. If seismic failure has not occurred, this top event represents the random failure of the turbine-driven pumps. This top event is only asked if relay chatter has occurred (i.e., top event SCT fails). The initial impact of relay chatter is failure of all AC power which precludes the operation of the motor-driven auxiliary feedwater pump. This top event helps determine the time available for recovery of relay chatter.

OC - Operator Resets Seismic Relay Chatter. This top event represents the operator action to reset seismic relay chatter, given failure of top event SCT, in order to restore emergency AC power.

SEL - Seismic Excessive LOCA. This top event represents a large seismic failure of the reactor coolant system that results in core damage since the resulting loss of coolant exceeds the emergency core cooling system makeup capability.

SID - Seismic Control Rocm and Hot Shutdown Panel Indication. This top event represents seismic failure of control room and hot shutdown panel indications that results in failure to control core cooling functions and, therefore, is assumed to lead to core damage.

3.1.3.3 Seismic Top Event Split Fraction Values

The point estimate split fraction values for the top events used in the event trees are summarized in Table 3-4. The seismic top event split fractions are derived from the component and structural fragility values, which are defined in Section 3.1.4. Nonseismic equipment failures are also included in the event trees and the nonseismic split fractions for these top events are included in Table 3-4.

Unless otherwise noted, the nonseismic component failure rates used in the Seismic PRA are the same as the nonseismic failure rate used for nonseismic initiating events. This is because the seismic failure modes are considered separately, as separate top events.

3.1.3.4 Seismic Human Actions Analysis

The human actions that must be performed following a seismic event were analyzed using the results of the nonseismic estimates made for the internal events analysis. The values for the nonseismic human action failure rates were multiplied by a factor greater than one to account for lower success rates that may follow a seismic event. Seismic events may produce psychological stresses different than those following other initiating events. The human action multiplication factors only account for the operator response. The fragility of the actuation equipment and of the equipment to be actuated is accounted for separately in the system analysis.

Consistent with the LTSP, three different multiplication factors were defined: one for seismic events with spectral accelerations less than 1.75g, one for spectral accelerations between 1.75 and 2.5g, and one for spectral accelerations greater than 2.5g. The multiplication factor for spectral accelerations less than 1.75g is typically 1.0. This means that the seismic event may initiate a transient (i.e., cause reactor and turbine trip and affect the performance of some hardware), but it will not significantly affect operator performance; this is treated like any other initiating event. For spectral accelerations between 1.75 and 2.5g, the operator may be disconcerted and confused by equipment and structure movement taking place around him, but he is unlikely to be physically affected. A multiplication factor of 5 typically was assigned to error rates for seismic events within this range. For spectral accelerations greater than 2.5g, the operator may be even more anxious and may be physically affected. He may be knocked down or knocked against something; things may fall on him, or the atmosphere may be clouded by dust limiting visibility. It is not expected that operators will be trapped or otherwise disabled by falling objects. A multiplication factor of 30 was used for these cases. These three multiplication factors were used for all significant human actions. For less significant human actions, the largest multiplication factor, 30, was applied at all acceleration levels to simplify the model in a conservative manner. All of the operator routes to remotely actuated equipment were checked for potential blockage resulting from a seismic event. No operator routes were judged as likely to be blocked. The seismic human action values used in the IPEEE are summarized in Table 3-6.

3.1.3.5 Seismic Dependencies

A table was developed that defines the impact of seismic failure of a component to system or top event failures. Some seismic failures between similar redundant components (due to proximity or other factors) are conservatively considered completely dependent; for example, the residual heat removal (RHR) pumps. Other seismic failures among similar components, specifically balance of plant piping and supports, which function in series are conservatively considered completely independent. The seismic PRA component grouping table is shown in Table 3-7.

3.1.3.6 Relay Chatter Analysis

A detailed relay chatter analysis was performed as part of the LTSP, and it was utilized in this analysis. The objectives of the relay chatter analysis were to:

- Identify relay contacts that affect components required for safe shutdown following a seismic event

- Determine which of the contacts are susceptible to relay chatter

- Determine the consequences of relay chatter

- Determine how the operator can diagnose the problem

- Determine the means available for the operator to correct the problem, such as resetting the control circuit in the control room

As a result of analyzing many systems and components during the LTSP, two components were determined to have seismic relay chatter fragilities that make a significant contribution to seismic core damage. The two components that were modeled are the diesel generator control panel and the 4 Kv Switchgear. The analysis also concluded that relay chatter did not impact containment isolation. Also, human action failure rates for recovery of relay chatter were developed and modeled.

3.1.3.7 Seismically Induced Very Small LOCAs

It was postulated (Reference 3-22) that a seismic event may cause very small reactor coolant system (RCS) leaks. These leaks are postulated to result from seismic failure or degradation of small piping or pump seals. Although the larger seismic failures of RCS piping are explicitly modeled as Excessive, Large, Medium, and Small (those that exceed the charging pump makeup capability of approximately 150 gpm) loss of coolant accidents (LOCAs), these probabilities of seismic RCS piping failure do not include the probability of very small RCS piping failures (which result in leaks which would not exceed charging pump makeup capability). The internal events model, which was used as a basis for the seismic event tree model, did not require charging pump function for most sequences in which a LOCA had not occurred. To conservatively model these seismically caused very small leaks, all seismic initiating events require charging pump function to maintain RCS level.

3.1.3.8 Seismically Induced Fires

Seismically induced fires were assessed as part of the LTSP. The potential for seismically induced fires was also considered in the fire risk scoping study evaluation (Section 4.8) following the approach outlined in the EPRI Fire-Induced Vulnerability Evaluation final report (Reference 3-19).

3.1.3.9 Seismic Events Success Criteria

System success criteria and mission times were generally left the same as those used in the internal events analysis. Two exceptions are the mission times for the diesel generator system and the fuel oil transfer system. These mission times were changed from 6 to 24 hours. This conservatively models the longer recovery time that may be required to restore offsite power following a seismic initiator.

3.1.3.10 Seismically Induced Floods

The internal flooding scenarios previously analyzed (Reference 3-20) were reviewed and none was determined to present unique seismic problems. Additionally, a number of the seismic top events include contributing causes of piping failure or other component failures which considers potential seismic flooding scenarios. Seismically induced fires, seismic actuation of fire suppression systems, and seismic degradation of fire suppression systems are addressed in Sections 4.8.1.1 to 4.8.1.3.

3.1.4 Evaluation of Component Fragilities and Failure Modes

The fragilities used in the seismic PRA quantification are listed in Table 3-8. These are the same as the values used in the LTSP or have minor refinements with the following exceptions:

- Safety-related block walls were added
- 230 Kv fragility values were revised
- Diesel generator control panel relay chatter revised
- Containment spray pumps revised

The HCLPF values for components and structures are included in Table 3-8.

3.1.5 Analysis of Plant Systems and Sequences

The seismic initiating events have the potential to cause seismic failures of components or structures and are assumed to lead to a reactor trip. Nonseismic failures may also occur in seismic initiated sequences. The seismic initiating event frequencies for the various magnitude seismic events, along with the conditional top event split fraction probabilities (both seismic and nonseismic), are included in the seismic plant model. This model is quantified to determine the probability of seismically induced core damage and the frequencies of PDS.

3.1.5.1 Seismic Quantification

The six seismic initiating events were quantified using the event trees presented in Section 3.1.3.1. The resulting point estimate core damage frequency for the six initiating events

is presented in Figure 3-10. The point estimate core damage frequency due to all seismic events is calculated to be 4.0E-5. Plant fragilities were also calculated that consider both seismic and nonseismic failures. The plant failure fragility (initiator contribution to core damage, including seismic failure and random failure modes, divided by each seismic initiator frequency) for core damage is presented in Figure 3-11.

3.1.5.2 Seismic Uncertainty Analysis

To account for uncertainties in the seismic initiating event frequencies, the component failure rates, and the equipment maintenance unavailabilities, the uncertainty in the seismic core damage frequency was calculated. The seismic PRA core damage frequency has the following statistical characteristics:

Point Estimate	= 4.0E-5
5^{th} Percentile	= 6.8E-7
Median	= 5.2E-6
95^{th} Percentile	= 2.4E-4

The results of the seismic uncertainty analysis are also presented in Figure 3-13. The fifth and ninety-fifth percentile values indicate the uncertainty in the calculation of the core damage frequency. The uncertainty in the seismic risk is greater than that from internal events. This is primarily due to the large uncertainties of seismic hazard curves and the component fragilities.

3.1.5.3 Seismic Sequences

The IPEEE reporting guidelines provided in NUREG-1407 (Reference 3-4) suggest using the core damage sequence selection criteria provided in NUREG-1335 (Reference 3-18). Some of the NUREG-1335 selection criteria also deal with containment performance. This analysis provides the results in terms of systemic sequences, as opposed to functional sequences. The reporting guidelines for systemic sequences are as follows:

1. Any systemic sequence that contributes 1E-7 or more per reactor year to core damage

2. Any systemic sequence within the upper 95 percent of the core damage frequency

3. Any other systemic sequences that the utility determines to be important to core damage frequency

The NRC sequence reporting guidance states that the total number of most significant sequences to be reported need not exceed 100.

Table 3-9 contains the top 100 sequences from the seismic PRA quantification. The highest frequency class of sequences is a seismically caused loss of all emergency power

(station blackout). Station blackout sequences constitute approximately 40 percent of the seismic core damage frequency. These station blackout events that result in core damage often include loss of offsite power and then, include seismic failure of one or more electrical components: 4 kV Vital AC Power (top event SACSS), Diesel Generators (top event SDG), 125 Vital DC Power (top event SDC), or Relay Chatter (top event SCT) (See Section 3.1.3.2 for top event description). Figure 3-12 presents the fragility of this class of station blackout sequence.

Many of the station blackout core damage sequences are those in which the turbine driven AFW pump initially operates and are commonly termed "slow station blackout" sequences . This refers to the timing of the sequence and means that if no recovery actions occur, core damage and vessel failure do not occur until later than 10 hours after sequence initiation (Reference 3-20). If no recovery occurs, containment failure would not occur until more than one day after sequence initiation.

3.1.5.4 Seismic PRA Top Event Importance

Table 3-10 contains the top event importance for seismic initiators, which includes seismic and nonseismic top events. Table 3-11 contains the seismic component importance. The probabilistic or fractional importance is the fraction of the core damage frequency involving failure of this top event or component. The Fussel-Vesely importance is approximately the fraction of the core damage frequency that the top events's or component's failure directly contributed to core damage. To contrast these two importance types, there are component failures that do not contribute to core damage, specifically containment heat removal systems, that will have a positive probabilistic or fractional importance, but will have a zero Fussel-Vesely importance to core damage. The key top event Fussel-Vesely importances in Table 3-10 are described below:

- The highest Fussel-Vesely importance is for top event SOP, seismic loss of offsite power. Although this failure does not lead directly to core damage, it significantly increases the likelihood of core damage resulting due to a loss of power in one form or another. Failure of SOP makes the plant more vulnerable to seismic or nonseismic failure of the emergency diesel generators which would then lead to core damage if no recovery is credited. The Seismic Component Importance Report (Table 3-11) shows the significance of the single component in top event SOP (ZOSPWR-230 kV switchyard) to core damage.

- The second highest Fussel-Vesely importance is top event SACSS, failure of 4 kV vital power. This leads directly to core damage, if no recovery occurs, since all power to pumps that can resupply the reactor coolant is lost and it will eventually lead to loss of DC power, which results in secondary heat removal failure. Table 3-11 shows that the second highest Fussel-Vesely component is the turbine building shear wall, which is one of several components that result in SACSS failure.

- The third, fourth, sixth, and ninth ranked top events are for nonseismic and seismic failure of the diesel generators, which, along with loss of offsite power (top event SOP), lead to core damage.

- The fifth and seventh highest Fussel-Vesely seismic top event failures are OC and SCT, recovery from relay chatter and relay chatter, respectively. Failure of both of these top events leads to core damage for similar reasons as SACSS failure discussed above. Table 3-11 shows the third highest ranked component is 4 Kv switchgear relay chatter component failure, which results in SCT top event failure.

3.1.5.5 Seismic Basic Event Importance

Table 3-12 presents the basic event importance ranking for seismic initiators and lists the most significant nonseismic failures and human actions by Fussel-Vesely importance. The key basic event importances are described below:

- The top two basic event contributing to core damage are human actions to reduce CCW heat loads and to crosstie the ASW to the other unit, respectively. These human actions both occur at the highest seismic acceleration and have the largest multiplication factor applied to the error rate.

- The next basic event is the human action to fail to switch to RHR recirculation cooling, which is conservatively analyzed at the highest seismic level failure rate, for all seismic initiators.

- The next three basic events are equipment failures: failure of the pressurizer safety valves to reclose.

3.1.5.6 Seismic Core Damage Frequency Vulnerability Evaluation

Based on the results presented in this study and the previous findings of the DCPRA-1988, no fundamental vulnerabilities with regard to seismic induced core damage exist at DCPP. The NUMARC Report 91-04, "Severe Accident Issue Closure Guideline," (Reference 3-23) provides vulnerability screening guidelines for core damage sequences. These guidelines are summarized as follows:

CORE DAMAGE FREQUENCY PER GROUP (PER REACTOR YEAR)	RECOMMENDED ACTION
Less than 10^{-6}	No action required
10^{-5} to 10^{-6}	Establish Severe Accident Management Guideline, with emphasis on preventing core damage, vessel failure, and containment failure.
10^{-4} to 10^{-5} or 20% to 50% of CDF	Make change in EOPs, other plant procedures, or make minor hardware change, with emphasis; or establish Severe Accident Management Guideline.
Greater than 10^{-4} or Greater than 50% of CDF	**VULNERABILITY** - Make plant administrative, procedural or hardware modification, with emphasis on reducing the likelihood of the sequence initiator; or make change in plant procedures with emphasis on prevention of core damage; or establish Severe Accident Management Guideline.

Based on the above guidelines, a vulnerability refers to any component, system, operator action, or accident sequence that contributes more than 50% to the core damage frequency or has a frequency that exceeds 1E-4 per year. The seismic PRA core damage frequency of 4.0E-5 per year calculated in this study is sufficiently low so as to preclude any vulnerabilities based solely on frequency. Additionally, no one type of sequence contributes more than 50 percent of the core damage frequency since the most frequent component failure sequence is seismic loss of offsite power and seismic failure of the turbine building shear wall which contributes 18 percent (Table 3-11) of the seismic core damage frequency.

3.1.6 Analysis of Containment Performance

3.1.6.1 Seismic Plant Damage States

The seismic event tree quantification binned the seismically induced core damage sequences into various PDS. The PDS defined are identical to those defined in the DCPP IPE (Reference 3-20). The PDS define the entry conditions into the Level 2 analysis, and define the thermodynamic conditions of the RCS at the time of core damage, the status of containment, and the availability of passive and active plant features that can terminate the accident or mitigate the release of radioactive materials to the environment. Table 3-13 contains the PDSs with the ten highest frequencies for the seismic event tree quantification of the six seismic initiating events.

3.1.6.2 Seismic Containment Vulnerability Screening

The seismic PRA results were reviewed to determine whether containment performance is deemed adequate. Containment vulnerabilities are defined as they were in the IPE

(Reference 3-20) as large (greater than 3-inch diameter equivalent opening), early releases or bypasses which exceed 1E-5 per year in frequency, or are greater than 20 percent of the total core damage frequency.

The large, early releases (defined by PDS LNNNL, LNYAL, LNYCL, SXNNL, LNYGL, etc.), constitute approximately 3 percent of the seismic core damage frequency and have a total frequency of only approximately 1E-6 per year. Thus, there is no containment vulnerability due to seismic initiating events. The aggregate core damage plus large containment failure fragility, conditional failure fraction of core damage combined with a large containment isolation failure or of loss of containment integrity, is shown in Figure 3-14.

Much of the seismically initiated core damage frequency resulting in large containment failures (approximately one third of all large containment failures) are a consequence of steam generator failure. The model of steam generator failure that results in containment failure has several layers of conservatism:

- Steam generator failure is modeled as occurring when stresses exceed the ultimate strength of the upper lateral support; the ultimate strength of the upper lateral support was conservatively estimated. Even if the support fails, the amount of steam generator movement would be limited by nearby structures.

- The postulated movement of the steam generator for such stress is assumed to rupture the RCS piping or the steam piping. This assumption is conservative because the piping is ductile and may not rupture even if steam generator movement caused strain in the piping.

- Rupture of one or more of the RCS pipes due to steam generator movement is assumed to fail containment due to the blowdown pressure load. This assumption is conservative because the containment has significant capacity beyond the likely pressure that would result from a single steam generator failure.

Structure failure of the containment as a result of seismic loading contributes only a small amount (less than a tenth of all seismically caused large containment failures) to the total core damage plus containment failure frequency. This is because the containment has a relatively high seismic capability; the HCLPF value for containment is 3.58g.

Excessive LOCA, top event SEL, is also conservatively modeled as resulting in large containment failure.

Small containment failures, less than an equivalent 3-inch diameter hole, are predicted for approximately 16 percent of the seismic core damage sequences. There are more small containment failures than large containment failures because power is required to isolate the RCP seal return line and many seismic sequences result in loss of AC power due to failure of several different components. Loss of AC power leaves the seal return line unisolated and, if operator actions to manually isolate this line fail, a small containment isolation failure results. Both relay chatter and block wall failure are modeled as failing

electric power and, consequentially failing small containment isolation. The aggregate small containment failure fragility, i.e., the conditional failure fraction of core damage coupled with a small containment isolation failure, is shown in Figure 3-15.

Another failure mode was identified and modeled that may result in small containment isolation failure. Failure of the containment fan cooler units connection to the component cooling water (CCW) piping may create a communication path between the containment atmosphere and the CCW system. Bypass of the containment boundary can then occur when the water in the CCW system has been forced out of the system by the containment pressure. At this time, containment atmosphere can be released by entering the failed CCW pipe inside containment, passing through the CCW piping to the surge tank, and out the unisolated surge tank vent. If the vent is isolated, release from the CCW surge tank could occur through the relief valve (which discharges to the auxiliary building) after containment pressure exceeds the 30 psig relief valve setpoint. This is modeled as top event SCB. This is a relatively small contributor to small containment isolation failure since this failure occurs in less than 2 percent of the core damage sequences compared to 16 percent for total small containment isolation failure.

Table 3-13 contains the frequencies of the top ten PDS that result from quantifying the six seismic initiating events. These PDS are similar to those from the internal events analysis (Reference 3-20) in that six of the top eight PDS also appear in the top eight internal event PDS. The seismic PDS are different from those of internal events in that the proportion that have total loss of AC power is higher due to the importance of the offsite power, turbine building shear wall, 4 kV switchgear, and diesel generator components. Also, PDS that result in large containment failure or large isolation failure (fifth letter "L" in the PDS descriptor) have a higher frequency in the seismic analysis because of the unique seismic top events that lead to core damage and containment failure (top events SEL and SSG).

3.2 USI A-45, GI-131 AND OTHER SEISMIC SAFETY ISSUES

As part of the DCPP IPE report (Reference 3-20), an evaluation of applicable unresolved safety issues (USIs), generic safety issues (GSIs), and generic issues (GIs) was made to determine the ability of the internal events PRA to identify vulnerabilities and resolve the issues. The IPE report (Reference 3-20) addressed USI A-45, Shutdown Decay Heat Removal Requirements, for internal events. This section provides an additional evaluation of these issues based on the seismic analysis.

Section 3.2.1 of this report addresses resolution of USI A-45, for seismically initiated events. Section 3.2.2 of this report addresses resolution of GI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants." Section 3.2.3 addresses the status of other seismically related GSIs and USIs at DCPP.

3.2.1 USI A-45, "Shutdown Decay Heat Removal Requirements"

The IPE report addressed USI A-45 for internal events. This section evaluates the same decay heat removal functions that were evaluated in the IPE: AFW, bleed and feed, and RHR. These decay heat removal functions were evaluated to identify unique seismic impacts by identifying the key components for each function and their seismic capability and contribution to core damage.

Table 3-8 contains the DCPP component and structure fragilities considered in the seismic PRA. Other components of these systems that were not listed have even higher seismic capabilities. The HCLPF capacities and the seismic importance of the secondary heat removal function (steam-driven AFW pump and steam generators) are given in the table below. The motor-driven AFW pumps have much greater seismic capacity and, therefore have a much smaller importance. To properly evaluate the steam generator values, the conservatism of their modeling (see Section 3.1.6.2) should also be considered. The importance of seismic failure of these components to seismically initiated core damage sequences is small.

The HCLPF seismic capacities and the seismic importance of the feed and bleed related systems (power-operated relief valves and centrifugal charging pumps) are given in the table below. The importance of these feed and bleed related components to seismically initiated core damage sequences is negligible.

The HCLPF seismic capacities and the seismic importance of the RHR function (refueling water storage tank, RHR pumps, and RHR heat exchangers) are given in the table below. The importance of these RHR components to seismically initiated core damage sequences is negligible.

No seismic-related weaknesses or vulnerabilities (i.e., greater than 1E-4 or greater than 50 percent of core damage frequency (Reference 3-23)) related to decay heat removal were identified in the seismic PRA evaluation. As a result, USI A-45, as it relates to seismic events at DCPP, is considered closed.

Decay Heat Removal Component	HCLPF Capacity* (g)	Seismic Importance (% of Core Damage Frequency)
Steam Driven Auxiliary Feedwater Pump	3.38	< 0.1
Steam Generator	2.63	1.2
Refueling Water Storage Tank	3.37	< 0.1
RHR pumps	3.35	< 0.1
RHR Heat Exchanger	3.49	< 0.1

Power Operated Relief Valves	2.32	< 0.1
Centrifugal Charging Pumps	4.45	< 0.1

* Average 5 percent damped spectral acceleration in the 3-8.5 Hz range

3.2.2 GI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants"

GI-131 was identified at certain Westinghouse plants because portions of the in-core flux mapping system that have not been seismically analyzed are located directly above the seal table. Failure of this equipment during a seismic event could be postulated to cause multiple failures at the seal table and could be postulated to produce an equivalent small-break LOCA.

As documented in a 1985 letter to the NRC (Reference 3-21), in March 1982, during the normal course of the Diablo Canyon Unit 1 Seismically Induced Systems Interaction Program (SISIP), PG&E postulated an interaction between the nonsafety-related portions of the movable incore flux mapping system (interaction source) and the tubing/seal table (interaction target). A similar interaction was postulated for Unit 2 in April 1983 during the Unit 2 SISIP. Subsequently, PG&E requested Westinghouse to perform an analysis to evaluate the ability of the fixed and movable frame assemblies of the flux mapping system to withstand a Hosgri earthquake and maintain structural integrity. Westinghouse recommended the following modifications; these modifications were completed on Unit 1 in April 1984 and on Unit 2 in June 1985.

- Weld the fixed frame baseplates to the trolley beam
- Replace the 0.375-inch diameter cap screws, which connect the wheel assemblies and the movable frame, with ASTM A325 bolts (or equivalent) of the same size.
- Add 0.25-inch plate stiffeners to the movable frame anchors
- Modify the existing movable frame seismic anchor brackets in accordance with a new Westinghouse design; provide additional brackets to the free ends of the movable frame wheel assemblies
- Add restraint for the isolation valve support structure

After completion of these modifications, the SISIP walkdown team inspected the installation.

It is concluded that the modifications to the DCPP flux mapping equipment precludes any potential seismic interaction problems associated with the flux mapping system. Thus, G-131 is considered closed.

3.2.3 Other Seismic Issues

The following NRC programs, identified in NUREG-1407, are related to seismic events. The status of these programs is as follows:

- USI A-17, "System Interactions in Nuclear Power Plants"

  USI A-17 addresses NRC concerns regarding the interaction of various systems with regard to whether actions or consequences could adversely affect the redundancy or consequences of safety systems. Per NUREG-1407, the evaluation of spatial system interaction under seismic conditions is included in USI A-46. A discussion of USI A-46 is provided below; USI A-17, with regard to seismic risk at DCPP, is considered closed or not applicable since it is considered as part of USI A-46.

- USI A-40, "Seismic Design Criteria"

  USI A-40 investigates selected areas of the seismic design process. The NRC Staff identified alternative approaches to the NRC criteria in the Standard Review Plan to reflect the current state-of-the-art and industry practice. The concern for the seismic capacity of the safety-related, above-ground tanks is included in USI A-46. USI A-40 is not applicable to DCPP; DCPP's seismic design criteria address the issues identified by USI A-40. Thus, USI A-40 is considered closed for DCPP.

- USI A-46, "Verification of Seismic Adequacy of Equipment in Operating Plants"

  USI A-46 developed an alternative method and acceptance criteria to verify the seismic adequacy of equipment in some operating plants with construction permits docketed before 1972. This issues is not applicable to DCPP; DCPP seismic Category I (Design Class I) equipment has been designed according to NRC accepted seismic design criteria and methods.

- The "Eastern U.S. Seismicity Issue (or Charleston Earthquake Issue)"

  The "Eastern U.S. Seismicity Issue" identifies the possibility of large earthquakes that could occur in the central or eastern United States. This issue is not applicable to DCPP, since the plant is located in the western United States, and the issue is only applicable to central and eastern U.S. plants.

None of the issues listed in this section is applicable to DCPP; as such, all of these issues are considered closed for Diablo Canyon.

3.3 REFERENCES

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3-16. Pacific Gas and Electric Company, Calculation File No. C.4.2, "DCPP General Transient Event Trees," Revision 5, May 9, 1994.

3-17. Pacific Gas and Electric Company, Calculation File No. N.1, "DCPP PRA Plant Damage States," Revision 1, April 25, 1994.

3-18. U.S. Nuclear Regulatory Commission, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August 1989.

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3-21. Pacific Gas and Electric Company, "IE Information Notice 85-45: Potential Seismic Interaction Involving the Movable Incore Flux Mapping System Used in Westinghouse-Designed Plants," DCL-85-242, July 18, 1985.

3-22. NUREG/CR-4840, "Procedure for the External Event Core Damage Frequency Analysis for NUREG-1150," November 1990.

3-23. NUMARC, "Severe Accident Issue Closure Guideline," NUMARC Report 91-04, January 1992.

Table 3-1. Seismic Hazard Table

Exceedance Frequencies for MODEL Name: DC93SEIS

Fragility Case: SEIS

Accelerations

	Curve Weights	.200	.500	.800	1.000	1.200	1.500	2.000	2.500	3.000	4.000
1.	3.42E-01	1.28E-02	4.36E-03	1.57E-03	6.69E-04	2.93E-04	7.81E-05	9.76E-06	1.36E-06	1.84E-07	1.74E-09
2.	1.96E-01	1.65E-02	6.25E-03	2.82E-03	1.55E-03	7.69E-04	2.35E-04	2.71E-05	2.97E-06	3.20E-07	2.77E-09
3.	2.17E-01	1.86E-02	7.49E-03	3.71E-03	2.31E-03	1.35E-03	5.30E-04	9.01E-05	1.31E-05	1.72E-06	1.67E-08
4.	1.11E-01	1.96E-02	8.37E-03	4.40E-03	2.97E-03	1.95E-03	9.58E-04	2.40E-04	5.02E-05	9.20E-06	1.46E-07
5.	3.60E-02	1.49E-02	6.82E-03	3.81E-03	2.75E-03	2.04E-03	1.32E-03	5.81E-04	2.04E-04	5.94E-05	2.96E-06
6.	4.30E-02	3.64E-02	1.58E-02	8.50E-03	5.79E-03	3.82E-03	1.82E-03	3.97E-04	6.95E-05	1.08E-05	1.33E-07
7.	3.20E-02	4.15E-02	1.88E-02	1.05E-02	7.45E-03	5.27E-03	2.91E-03	8.14E-04	1.68E-04	2.85E-05	2.85E-07
8.	2.30E-02	5.26E-02	2.80E-02	1.45E-02	1.06E-02	7.89E-03	4.90E-03	1.83E-03	5.26E-04	1.24E-04	4.32E-06

Table 3-2. Mean Seismic Hazard Frequency Table

Exceedance Frequencies for MODEL Name: DC93SEIS

Fragility Case: SEIS

Accelerations

.200	.500	.800	1.000	1.200	1.500	2.000	2.500	3.000	4.000
1.85E-02	7.44E-03	3.56E-03	2.19E-03	1.35E-03	6.26E-04	1.61E-04	3.73E-05	7.89E-06	2.42E-07

Table 3-3. List of Plant Modifications to be Checked Against the Long Term Seismic Program HCLPF Values

PLANT STRUCTURES

- Major structural modifications to
  - Containment building
  - Concrete internal structures of the Containment building
  - Auxiliary building
  - Turbine building
  - Intake structure
- New Design Class I structures that significantly impact Plant seismic margins
- New major non-category I structures that are located in the proximity of Design Class I structures and that significantly impact Plant seismic margins
- Masonry walls (all new construction and significant modification beyond the modification program described in Reference 24)

EQUIPMENT AND COMPONENTS

- Nuclear Steam Supply System
  - Steam Generators
  - Power Operated Relief Valves
  - Reactor Coolant Pumps
- Component Cooling Water System
  - Heat Exchangers
  - Surge Tanks
- Emergency Diesel Generators
  - Excitation Cubicles
  - Control Panels
- Containment Fan Coolers
- 4.16kV Vital Electric Power System,
  - Switchgear
- 125V DC Electric Power System
  - Batteries
  - Battery Chargers
  - Switchgear Breaker Panels (vital)
- 120V AC Electric Power System
  - Inverters
- 480 V Vital Electric Power System
  - 4160V/480V Transformer
- Control Room
  - Main Control Board Anchorage
- 230kV Switchyard (Non-Class I)
  - Circuit Breakers
  - Switches
  - Transformers
- Miscellaneous
  - Instrument Impulse Lines (which affect loss-of-coolant accident detection/mitigation)
  - Relays (review for chatter potential)
  - New major safety-related equipment that significantly impacts seismic margins

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
AA1F	AA	6.1169E-04	VITAL AC TRAIN F FAILS (BKR RECOVERY)	
AA1G	AA	6.1169E-04	VITAL AC TRAIN G FAILS (BKR RECOVERY)	
AA1H	AA	6.1169E-04	VITAL AC TRAIN H FAILS (BKR RECOVERY)	
AA2FG	AA	1.9582E-06	VITAL AC TRAIN F&G FAIL (BKR RECOVERY)	
AA2FH	AA	1.9582E-06	VITAL AC TRAIN F&H FAIL (BKR RECOVERY)	
AA2GH	AA	1.9582E-06	VITAL AC TRAIN G&H FAIL (BKR RECOVERY)	
AA3FGH	AA	3.8781E-07	VITAL AC TRAIN F&G&H FAIL (BKR RECOVERY)	
AA4F	AA	6.4931E-04	VITAL AC TRAIN F FAILS (NO BKR RCVRY)	
AA4G	AA	6.4931E-04	VITAL AC TRAIN G FAILS (NO BKR RCVRY)	
AA4H	AA	6.4931E-04	VITAL AC TRAIN H FAILS (NO BKR RCVRY)	
AA5FG	AA	2.3386E-05	VITAL AC TRAIN F&G FAIL (NO BKR RCVRY)	
AA5FH	AA	2.3386E-05	VITAL AC TRAIN F&H FAIL (NO BKR RCVRY)	
AA5GH	AA	2.3386E-05	VITAL AC TRAIN G&H FAIL (NO BKR RCVRY)	
AA6FGH	AA	5.6163E-06	VITAL AC TRAIN F&G&H FAIL (NO BKR RCVRY)	
AB1F	AB	1.3511E-03	SU FEEDER BKR F FAILS (W/RCVRY)	
AB1G	AB	1.3511E-03	SU FEEDER BKR G FAILS (W/RCVRY)	
AB1H	AB	1.3511E-03	SU FEEDER BKR H FAILS (W/RCVRY)	
AB2FG	AB	5.4650E-06	SU FEEDER BKR F&G FAIL (W/RCVRY)	
AB2FH	AB	5.4650E-06	SU FEEDER BKR F&H FAIL (W/RCVRY)	
AB2GH	AB	5.4650E-06	SU FEEDER BKR G&H FAIL (W/RCVRY)	
AB3FGH	AB	9.3238E-07	SU FEEDER BKR F&G&H FAIL (W/RCVRY)	
AB4F	AB	1.4375E-03	SU FEEDER BKR F FAILS (NO RCVRY)	
AB4G	AB	1.4375E-03	SU FEEDER BKR G FAILS (NO RCVRY)	
AB4H	AB	1.4375E-03	SU FEEDER BKR H FAILS (NO RCVRY)	
AB5FG	AB	5.4972E-05	SU FEEDER BKR F&G FAIL (NO RCVRY)	
AB5FH	AB	5.4972E-05	SU FEEDER BKR F&H FAIL (NO RCVRY)	
AB5GH	AB	5.4972E-05	SU FEEDER BKR G&H FAIL (NO RCVRY)	
AB6FGH	AB	1.3482E-05	SU FEEDER BKR F&G&H FAIL (NO RCVRY)	
AC1	AC	4.3822E-03	ALL CONDITIONS (NO SUPPORT REQUIRED)	
AF1	AF	6.1170E-04	All support available with recovery	AA1F
AFA	AF	6.4930E-04	All support available no recovery	AA4F
AFF	AF	1.0000E+00	Guaranteed Failure	
AG1	AG	6.1010E-04	DF-S, AF-S, with recovery	AA1G
AG2	AG	3.2020E-03	DF-S, DG-F, with recovery	AA2FG
AG3	AG	6.1170E-04	DF-F, with recovery	AA1G
AGA	AG	6.2630E-04	DF-S, AF-S, with recovery	AA4G
AGB	AG	3.6020E-02	DF-S, AF-S, no recovery	AA5FG
AGC	AG	6.4930E-04	DF-F, no recovery	AA4G
AGF	AG	1.0000E+00	Guarnanteed Failure	
AH1	AH	6.0890E-04	DF-S, DG-S, AF-S, AG-S (BKR RCVRY)	AA1H
AH2A	AH	2.5760E-03	DF-S, DG-S, AF-S, AG-F (BKR RCVRY)	AA2GH
AH2B	AH	2.5760E-03	DF-S, DG-S, AF-F, AG-S (BKR RCVRY)	AA2FH
AH3	AH	1.9800E-01	DF-S, DG-S, AF-F,AG-F (BKR RCVRY)	AA3FGH
AH4A	AH	6.1010E-04	DF-S, DG-F, AF-S (BKR RCVRY)	AA1H
AH4B	AH	6.1010E-04	DF-F, DG-S, AG-S (BKR RCVRY)	AA1H
AH5A	AH	3.2020E-03	DF-S, DG-F, AF-F (BKR RCVRY)	AA2FH
AH5B	AH	3.2020E-03	DF-F, DG-S, AG-F (BKR RCVRY)	AA2GH
AH6	AH	6.1170E-04	DF-F, DG-F (BKR RCVRY)	AA1H
AHA	AH	6.0890E-04	DF-S, DG-S, AF-S, AG-S (NO RCVRY)	AA4H
AHBA	AH	2.8390E-02	DF-S, DG-S, AF-S, AG-F (NO RCVRY)	AA5GH
AHBB	AH	2.8390E-02	DF-S, DG-S, AF-F, AG-S (NO RCVRY)	AA5FH
AHC	AH	2.4020E-01	DF-S, DG-S, AF-F, AG-F (NO RCVRY)	AA6FGH
AHDA	AH	6.2630E-04	DF-S, DG-F, AF-S (NO RCVRY)	AA4H
AHDB	AH	6.2630E-04	DF-F, DG-S, AG-S (NO RCVRY)	AA4H
AHEA	AH	3.6020E-02	DF-S, DG-F, AF-F (NO RCVRY)	AA5FH
AHEB	AH	3.6020E-02	DF-F, DG-S, AG-F (NO RCVRY)	AA5GH
AHF	AH	1.0000E+00	Guaranteed Failure	
AHG	AH	6.4930E-04	DF-F, DG-F (NO RCVRY)	AA4H

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
AI1	AI	1.7749E-04	ALL SUPPORT AVAILABLE	
AS1	AS	1.5125E-05	NO LOSP; POWER AVAILABLE AT UNIT 1 4160V BUSES F AND G AND AT 125V DC BUS 12 (OP1)	
AS2	AS	4.1080E-03	NO LOSP; UNIT 1 4160V BUS F FAILED (OP2)	
AS3	AS	4.1332E-03	NO LOSP; UNIT 1 4160V BUS G FAILED OR 125V DC BUS 12 FAILED (OP1)	
AS4A	AS	1.5444E-02	NO LSOP; UNIT 1 4160V BUS F FAILED AND UNIT 1 4160V BUS G OR 125V DC BUS 12 FAILED (OP2)	
AS4B	AS	3.8932E-02	AS4 W ESAM 5	
AS4C	AS	1.8572E-01	AS4 W ESAM 30	
AS5	AS	4.1543E-03	LOSP; UNIT 1 4160V BUS F FAILED (OP2)	
AS6	AS	4.8968E-05	LSOP; UNIT 2 4160V BUS F FAILED (OP1)	
AS7	AS	1.8584E-01	LOSP; UNIT 1 4160V BUSES F AND G FAILED (OP20	
AS8	AS	4.6688E-03	LOSP; UNIT 1 4160V BUS F FAILED AND UNIT 2 4160V BUS F OR G FAILED (OP2)	
AS9	AS	5.2556E-03	LOSP; UNIT 1 4160V BUS G FAILED AND UNIT 2 4160V BUS F FAILED (OP1)	
ASA	AS	1.9157E-04	LOSP; UNIT 2 4160V BUSES F AND G FAILED	
ASB	AS	2.0357E-01	LOSP; UNIT 1 4160V BUSES F AND G FAILED AND UNIT 2 4160V BUS F OR G FAILED (OP2)	
ASC	AS	2.1633E-02	LOSP; UNIT 1 4160V BUS F OR G FAILED AND UNIT 2 4160V BUSES F AND G FAILED, OR NLOSP UNIT 1 BUS F FAILED AND NO CREDIT FOR UNIT 2 PUMPS	
ASD	AS	6.6833E-05	NLOSP; NO CREDIT FOR UNIT 2 PUMPS	
ASE	AS	1.8671E-02	NLOSP; UNIT 1 BUS G FAILED AND NO CREDIT FOR UNIT 3 PUMPS	
ASF	AS	1.0000E+00	GURANANTEED FAILURE	
ATO	AT	0.0000E+00	GUARANTEED SUCCESS	
ATF	AT	1.0000E+00	GUARANTEED FAILURE	
AW1	AW	1.9430E-05	ALL SUPPORT SYSTEMS AVAILABLE	
AW2	AW	9.2430E-02	ALL SUPPORT SYSTEMS AVAILABLE	AWH2
AW3A	AW	6.5788E-04	MDP 1-3 FAILS (NO SUPPORT FOR MDP 1-2)	
AW3B	AW	6.5788E-04	MDP 1-2 FAILS (NO SUPPORT FOR MDP 1-3)	
AW4	AW	5.8660E-02	SUPPORT FOR BOTH MDP'S UNAVAILABLE	
AW5	AW	4.4630E-03	SUPPORT FOR ALL 10% STM DMPS UNAVAILABLE	
AW6	AW	1.8110E-01	SUPPORT FOR 10% STM DMP UNAVAILABLE	AWH6
AW7	AW	6.2091E-03	SUPPORT FOR ALL 10% STM DMPS AND THE TDP UNAVAILABLE	
AW8A	AW	2.0439E-02	NO SUPPORT FOR 10% STM DUMPS AND MDP 1-2	
AW8B	AW	2.0439E-02	NO SUPPORT FOR 10% STM DUMPS AND MDP 1-3	
AW9	AW	1.1975E-01	NO SUPPORT FOR 10% STM DMPS & BOTH MDP'S	
AWAA	AW	7.9411E-02	NO SUPPORT FOR 10% STM DMPS/TDP/MDP 1-2	
AWAB	AW	7.9411E-02	NO SUPPORT FOR 10% STM DMPS/TDP/MDP 1-3	
AWB	AW	1.7586E-02	SUPPORT FOR THE TDP AND ONE MDP UNAVAILABLE (DUE TO 1 SG DEPRESSURIZING)	
AWC	AW	1.3010E-03	REACTOR TRIP FAILURE WITH TT SUCCESSFUL AND ALL OTHER SUPPORT SYSTEMS AVAILABLE	AWTC
AWD	AW	1.9955E-04	SUPPORT FOR THE TDP UNAVAILABLE	
AWF	AW	1.0000E+00	AFW SYSTEM GUARANTEED FAILURE	
AWH2	AWH	9.2433E-02	ALL SUPPORT SYSTEMS AVAILABLE	
AWH6	AWH	1.8111E-01	SUPPORT FOR 10% STM DMP UNAVAILABLE	
AWTC	AWT	1.3012E-03	REACTOR TRIP FAILURE WITH TURBINE TRIP SUCCESS AND ALL OTHER SUPPORT SYSTEMS AVAILABLE	
BB1F	BB	6.1593E-02	Train 2F fails with Recovery - TF=S	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
BB1G	BB	6.3410E-02	Train 2G fails with Recovery - TS=S	
BB1H	BB	5.9765E-02	Train 2H fails with Recovery - TH=S	
BB2FG	BB	2.2030E-02	Trains 2F & 2G fail with Recovery - TF=S	
BB2FH	BB	2.2025E-02	Trains 2F & 2H fail with Recovery - TF=S	
BB2GH	BB	2.2028E-02	Trains 2G & 2H fail with Recovery - TS=S	
BB3FGH	BB	8.9939E-05	Trains 2F, 2G, and 2H fail with Recovery - TS=S	
BF1	BF	6.1590E-02	TRAIN 2F FAILS WITH RECOVERY	BB1F
BFF	BF	1.0000E+00	Guaranteed failure	
BG1	BG	4.4100E-02	TRAIN 2G FAILS WITH RECOVERY (BF-S)	BB1G
BG2	BG	3.5770E-01	TRAIN 2G FAILS WITH RECOVERY (BF-F)	BB2FG
BGF	BG	1.0000E+00	Guaranteed failure	
BH1	BH	1.7620E-02	TRAIN 2H FAILS WITH RECOVERY (BF-S, BG-S)	BB1H
BH2	BH	5.3010E-01	TRAIN 2H FAILS WITH RECOVERY (BF-S, BG-F)	BB2GH
BH3	BH	5.5440E-01	TRAIN 2H FAILS WITH RECOVERY (BF-F, BG-S)	BB2FH
BH4	BH	4.0840E-03	TRAIN 2H FAILS WITH RECOVERY (BF-F, BG-F)	BB3FGH
BHF	BH	1.0000E+00	Guaranteed failure	
BI0	BI	0.0000E+00	GUARANTEED SUCCESS	
CC1	CC	2.5914E-05	ALL SUPPORT AVAILABLE	
CC2	CC	4.3318E-03	LOSS OF 4KV BUS H	
CC3F	CC	4.8364E-03	LOSS OF 4KV BUS F	
CC3G	CC	4.8359E-03	LOSS OF 4KV BUS G	
CC4FH	CC	2.5284E-01	LOSS OF 4KV BUS F & H	
CC4GH	CC	2.5284E-01	LOSS OF 4KV BUSES G AND H	
CC5A	CC	1.8939E-02	LOSS OF 4KV BUSES F AND G	
CC5B	CC	5.1493E-02	CC5 W ESAM = 5	
CC5C	CC	2.5489E-01	CC5 W ESAM = 30	
CC6	CC	6.4700E-05	LOSP, ALL OTHER SUPPORT AVAILABLE	
CC7F	CC	5.3915E-03	LOSP, LOSS OF 4KV BUS F	
CC7G	CC	5.3910E-03	LOSP, LOSS OF 4 KV BUS G	
CC7H	CC	5.3915E-03	LOSP, LOSS OF 4 KV BUS H	
CCF	CC	1.0000E+00	GUARANTEED FAILURE	
CDF	CD	1.0000E+00	no description entered	
CH1	CH	8.2063E-04	ALL SUPPORT AVAILABLE	
CH2	CH	2.3756E-02	ONE STANDBY PUMP TRAIN AVAILABLE ONLY	
CH3	CH	2.3756E-02	NORMALLY RUNNING PUMP TRAIN AVAILABLE ONLY	
CH4	CH	8.2063E-04	LOSP ; ALL SUPPORT AVAILABLE	
CHF	CH	1.0000E+00	GUARANTEED FAILURE	
CI1	CI	4.0819E-03	EITHER INBOARD OR OUTBOARD ISOL. VALVES MUST CLOSE	
CI2	CI	6.1302E-03	INBOARD VLVES (PEN 45) AND 1/2 VLVES (PEN 50, 51, 52) CLOSE	
CI3	CI	8.2200E-03	INBOARD ISOLATION VALVES (PEN 45, 50, 51, 52) MUST CLOSE	
CI4	CI	4.0819E-03	INBD. OR OUTBD. ISOLATION VLVS CLOSE - EXCESSIVE LOCA	
CI5	CI	6.1302E-03	INBD. PEN.45 & 1/2 VLVS PEN. 50,51,52 CLOSE -ELOCA	
CI6	CI	8.2200E-03	INBD. ISOL. VLVS. PEN. 45, 50, 51, 52 CLOSE - ELOCA	
CIF	CI	1.0000E+00	GUARANTEED FAILURE	
CP1	CP	1.3060E-06	EITHER INBOARD OR OUTBOARD ISOLATION VALVE(S) MUST CLOSE	
CP2	CP	1.6017E-05	OUTBOARD ISOLATION VALVES MUST CLOSE	
CP3	CP	8.4162E-03	FRACTION OF TIME PENETRATION 61, 62 OR 63 IS OPEN	
CP4	CP	1.3060E-06	SAME AS CP1 WITH VI FAILED SEISMICLY	
CP5	CP	1.6017E-05	SAME AS CP2 WITH VI FAILED SEISMICLY	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description.................	ISF......
CP6	CP	8.4162E-03	SAME AS CP3 WITH VI FAILED SEISMICLY	
CPF	CP	1.0000E+00	SCP=F	
CS0	CS	0.0000E+00	GUARANTEED SUCCESS	
CS1	CS	5.3730E-04	1/2 TRAINS OPERATES (ALL SUPPORT AVAILABLE)	
CS2	CS	1.6415E-02	1/1 TRAIN OPERATES (LOSS OF ONE VITAL BUS OR SSPS TRAIN)	
CSF	CS	1.0000E+00	GUARANTEED FAILURE	
CT1	CT	0.0000E+00	4KV SWGR &MCB CHATT	
CT2	CT	0.0000E+00	DG CTRL BD, 4KV SWGR	
CTF	CT	1.0000E+00	no description entered	
CV1	CV	1.0875E-03	1/2 SUBTRNS:ALL SUPRT (OSP,2F,1G,1H,2H)	
CV2	CV	2.0956E-01	1/2 SUBTRNS:NORM PWR SUBTRN F UNAVL(2F)	
CV3	CV	2.4102E-01	1/1 SUBTRNS:NO SUPPRT SUBTRN F (2F,1G)	
CV4	CV	2.6613E-02	1/2 SUBTRNS:NO SUPPRT SUBTRN H (1H,2H)	
CV5	CV	6.7733E-03	1/2 SUBTRNS:LOSP-VITAL AVLB(2F,1G,1H,2H)	
CV6	CV	4.4221E-02	1/1 SUBTRNS:LOSP-NO SUPPRT SUBTRN H	
CVF	CV	1.0000E+00	GUARANTEED FAIL-480V 2F,1G,1H,2H UNAVAIL	
CVI1	CVI	4.7629E-02	INITIATING EVENT FREQ FOR 1 YEAR	
CX1	CX	7.8007E-05	ALL SUPPORT AVAILABLE	
DA1	DA	7.3024E-04	VITAL DC TRAIN F (DC TRAIN 11) IS UNAVAILABLE	
DA2	DA	7.3024E-04	VITAL DC TRAIN G (DC TRAIN 12) IS UNAVAILABLE	
DA3	DA	7.2560E-04	VITAL DC TRAIN H (DC TRAIN 13) IS UNAVAILABLE	
DA4	DA	5.3297E-07	VITAL DC TRAINS F AND G ARE UNAVAILABLE	
DA5	DA	5.2962E-07	VITAL DC TRAINS F AND H ARE UNAVAILABLE	
DA6	DA	5.2962E-07	VITAL DC TRAINS G AND H ARE UNAVAILABLE	
DA7	DA	3.8643E-10	VITAL DC TRAINS F, G AND H ARE UNAVAILABLE	
DF1	DF	7.3020E-04	CSF FOR DF GIVEN: ALL SUPPORT AVAILABLE	DA1
DFF	DF	1.0000E+00	GUARANTEED FAILURE	
DG1	DG	7.3020E-04	CSF for DG given: DF-S	DA2
DG2	DG	7.2990E-04	CSF for DG given: DF-F	DA4
DGC1	DGC	7.5000E-01	DIESEL GENERATOR COUPLING FACTOR	
DGCF	DGC	1.0000E+00	SEISMIC FAILURE OF DGS IS CORRELATED, I.	
DGF	DG	1.0000E+00	Guaranteed Failure.	
DH1	DH	7.2560E-04	CSF for DH given: DF-S, DG-S	DA3
DH2	DH	7.2530E-04	CSF for DH given: DF-S, DG-F	DA6
DH3	DH	7.2530E-04	CSF for DH given: DF-F, DG-S	DA5
DH4	DH	7.2510E-04	CSF for DH given: DF-F, DG-F	DA7
DHF	DH	1.0000E+00	Guaranteed Failure.	
EL1	EL	1.0000E+00	EXCESSIVE LOCA	
ELF	EL	1.0000E+00	no description entered	
FC1	FC	6.7607E-07	2 OF 5 CFCUS START AND OPERATE 24 HOURS	
FC2	FC	5.3602E-06	2 OF 4 CFCUS START AND OPERATE 24 HOURS	
FC3	FC	1.5291E-04	2 OF 3 CFCUS START AND OPERATE 24 HOURS	
FC4	FC	1.7422E-02	2 OF 2 CFCUS START AND OPERATE 24 HOURS	
FCF	FC	1.0000E+00	GUARANTEED FAILURE	
FO0	FO	0.0000E+00	GUARANTEED SUCCESS	
FO1A	FO	9.9367E-05	ALL SUPPORT AVAILABLE	
FO1B	FO	9.9367E-05	FO1 W ESAM = 5	
FO1C	FO	9.9367E-05	FO1 W ESAM = 30	
FO2A	FO	6.9153E-03	SUPPORT FOR ONE TRAIN AVAILABLE; NO SUPPORT FOR OTHER TRAIN	
FO2B	FO	6.9153E-03	FO2 W ESAM = 5	
FO2C	FO	6.9153E-03	FO2 W ESAM = 30	
FO3A	FO	1.5840E-04	ONE TRAIN HAS NORMAL SUPPORT POWER; OTHER TRAIN HAS BACKUP POWER ONLY	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
F03B	FO	5.7340E-04	F03 W ESAM = 5	
F03C	FO	3.1671E-03	F03 W ESAM = 30	
F04A	FO	1.0071E-02	BOTH NORMAL SUPPORT POWER UNAVAILABLE; BOTH BACKUP POWER SUPPLIES AVAILABLE	
F04B	FO	5.0075E-02	F04 W ESAM = 5	
F04C	FO	3.0010E-01	F04 W ESAM = 30	
F05A	FO	1.4841E-02	BOTH NORMAL SUPPORT POWER AND ONE BACKUP POWER SUPPLIES ARE UNAVAILABLE	
F05B	FO	5.4845E-02	F05 W ESAM = 5	
F05C	FO	3.0487E-01	F05 W ESAM = 30	
FOF	FO	1.0000E+00	GUARANTEED FAILURE	
FWF	FW	1.0000E+00	no description entered	
GF0	GF	0.0000E+00	GUARANTEED SUCCESS	
GF1	GF	9.0870E-02	DG 1-3 (BUS F) STARTS & RUNS FOR 6 HR	
GFF	GF	1.0000E+00	GUARANTEED FAILURE	
GG0	GG	0.0000E+00	GUARANTEED SUCCESS	
GG1	GG	8.9700E-02	DG 1-2 (BUS G) : DF-S	
GG2	GG	1.0250E-01	DG 1-2 (BUS G) : GF-F	
GG3	GG	9.0870E-02	DG 1-2 (BUS G) : GF-B	
GGF	GG	1.0000E+00	GUARANTEED FAILURE	
GH0	GH	0.0000E+00	GUARANTEED SUCCESS	
GH1	GH	8.8810E-02	DG 1-1 (BUS H) : GF-S,GG-S	
GH2	GH	9.8730E-02	DG 1-1 (BUS H) : GF-S/F,GG-F/S	
GH3	GH	1.3540E-01	DG 1-1 (BUS H) : GF-F,GG-F	
GH4	GH	8.9700E-02	DG 1-1 (BUS H) : GF-S/B,GG-B/S	
GH5	GH	1.0250E-01	DG 1-1 (BUS H) : GF-F/B,GG-B/F	
GH6	GH	9.0870E-02	DG 1-1 (BUS H) : GF-B,GG-B	
GHF	GH	1.0000E+00	GUARANTEED FAILURE	
GX1		9.0856E-02	1/3 DIESELS UNAVAILABLE	
GX2		9.3126E-03	2/3 DIESELS UNAVAILABLE	
GX3		1.2610E-03	3/3 DIESELS UNAVAILABLE	
HR1	HR	1.1431E-04	ALL SUPPORT AVAILABLE	
HR2	HR	2.0434E-03	TOP EVENT CH OR SI FAILED	
HR3	HR	4.7204E-03	TOP EVENT LA OR LB FAILED	
HR4	HR	4.9760E-03	TOP EVENT CH OR SI AND TOP EVENTS LA OR LB FAILED	
HR5	HR	1.1431E-04	4KV BUS F FAILED	
HR6	HR	2.0434E-03	4KV BUS F FAILED, TOP EVENT CH OR SI FAILED	
HR7	HR	4.7204E-03	4KV BUS F FAILED, TOP EVENT LA OR LB FAILED	
HR8	HR	4.9760E-03	4KV BUS F FAILED, TOP EVENT CH OR SI & LA OR LB FAILED	
HR9	HR	4.7200E-03	4KV BUS F AND 4KV BUS G FAILED	
HRA	HR	2.9698E-03	4KV BUS F AND 4KV BUSH FAILED	
HRB	HR	4.7204E-03	4KV BUS G FAILED	
HRC	HR	7.6505E-03	4KV BUS G FAILED, TOP EVENT CH OR SI FAILED	
HRD	HR	2.9698E-03	4KV BUS H FAILED	
HRE	HR	7.8286E-03	4KV BUS H FAILED, TOP EVENT CH OR SI FAILED	
HRF	HR	9.9998E-01	GUARANTEED FAILURE	
HS0	HS	0.0000E+00	no description entered	
HSF	HS	1.0000E+00	no description entered	
I11	I1	8.1524E-04	GIVEN DF-S, AF-S, AG-S OR DF-S, AF-F, AG-S	
I12	I1	1.2611E-03	GIVEN DF-S, AF-S, AG-F OR DF-S, AF-F, AG-F	
I1F	I1	1.0000E+00	GIVEN DF-F (GUARANTEED) FAILURE	
I21	I2	4.0767E-04	GIVEN DF-F (GUARANTEED FAILURE)	
I22	I2	6.3060E-04	GIVEN DG-S, AG-F	
I23	I2	6.3967E-04	GIVEN AG-S, I1-F	
I24	I2	6.3967E-04	GIVEN DG-S, AG-F, I1-F	
I2F	I2	1.0000E+00	GIVEN: DG-F (GUARANTEED FAILURE)	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description...............	ISF......
I31	I3	8.1524E-04	GIVEN DH-S, AH-S, AG-S OR DH-S, AH-F, AG-S	
I32	I3	1.2611E-03	GIVEN DH-S, AH-S, AG-F OR DH-S, AH-F, AG-F	
I3F	I3	1.0000E+00	GIVEN DH-F (GUARANTEED FAILURE)	
I41	I4	4.0767E-04	GIVEN: DG-S, AH-S, AG-S, OR DG-S, AH-F, AG-S	
I42	I4	6.3060E-04	GIVEN: DG-F, AH-S OR AG-F, DG-S, (AH-S OR AH-F)	
I4F	I4	1.0000E+00	GIVEN DG-F, AH-F (GUARANTEED FAILURE)	
IAF	IA	1.0000E+00	GUARANTEED FAILURE	
ID1	ID	0.0000E+00	no description entered	
IDF	ID	1.0000E+00	no description entered	
II1	II	1.6462E-05	INITIATING EVENT GIVEN ALL SUPPORT AVAILABLE	
ITF	IT	1.0000E+00	RHR SYSTEM GUARANTEED RUPTURE	
LA1A	LA	1.7440E-02	ALL SUPPORT AVAILABLE (SBLOCA)	LP4BA
LA1B	LA	2.4980E-02	LA1 W ESAM = 5	LP4BB
LA1C	LA	7.2100E-02	LA2 W ESAM = 30	LP4BC
LA2	LA	7.2100E-02	ALL SUPPORT AVAILABLE (BLEED AND FEED)	LP5B
LA3	LA	1.5550E-02	ALL SUPPORT AVAILABLE (LLOCA/MLOCA)	LP6B
LAF	LA	1.0000E+00	GUARANTEED FAILURE	
LB1A	LB	1.5350E-02	ALL SUPPORT AVAILABLE (LA SUCCESS - SLOCA)	LP4AA
LB1B	LB	1.5410E-02	LB1 W ESAM = 5	LP4AB
LB1C	LB	1.5780E-02	LB1 W ESAM = 30	LP4AC
LB2A	LB	1.3490E-01	ALL SUPPORT AVAILABLE (LA FAILED - SLOCA)	LP1A
LB2B	LB	3.9850E-01	LB2 W ESAM = 5	LP1B
LB2C	LB	7.9690E-01	LB2 W ESAM = 30	LP1C
LB3A	LB	1.7440E-02	LA GUARANTEED FAILURE (SLOCA)	LP4AA
LB3B	LB	2.4980E-02	LB3 W ESAM = 5	LP4AB
LB3C	LB	7.2100E-02	LB3 W ESAM = 30	LP4AC
LB4	LB	1.5780E-02	ALL SUPPORT AVAILABLE (LA SUCCESSFUL - B&F)	LP5A
LB5	LB	7.9690E-01	ALL SUPPORT AVAILABLE (LA FAILED - B&F)	LP2
LB6	LB	7.2100E-02	LA GUARANTEED FAILED (B&F)	LP5A
LB7	LB	1.5340E-02	ALL SUPPORT AVAILABLE (LA SUCCESSFUL - LLOCA)	LP6A
LB8	LB	2.9040E-02	ALL SUPPORT AVAILABLE (LA FAILED - LLOCA)	LP3
LB9	LB	1.5550E-02	LA GUARANTEED FAILURE (LLOCA)	LP6A
LBF	LB	1.0000E+00	GUARANTEED FAILURE	
LI1	LI	2.9664E-05	ALL CONDITIONS EXCEPT LLOCA (NO SUPPORT REQUIRED)	
LI2	LI	6.7710E-03	LLOCA I.E. (AC FAILURE)	LI1
LP1A	LP	2.3522E-03	no description entered	
LP1B	LP	9.9522E-03	LP1A W ESAM = 5	
LP1C	LP	5.7457E-02	LP1A W ESAM =30	
LP2	LP	5.7457E-02	no description entered	
LP3	LP	4.5168E-04	no description entered	
LP4AA	LP	1.7437E-02	RHR TRAIN A FAILS (SLOCA)	
LP4AB	LP	2.4976E-02	LP4A W ESAM = 5	
LP4AC	LP	7.2099E-02	LP4A W ESAM =30	
LP4BA	LP	1.7437E-02	RHR TRAIN B FAILS (SLOCA)	
LP4BB	LP	2.4976E-02	LP4B W ESAM = 5	
LP4BC	LP	7.2099E-02	LP4B W ESAM =30	
LP5A	LP	7.2099E-02	RHR TRAIN A FAILS (BLEED & FEED)	
LP5B	LP	7.2099E-02	RHR TRAIN B FAILS (BLEED & FEED)	
LP6A	LP	1.5551E-02	RHR TRAIN A FAILS (LLOCA/MLOCA)	
LP6B	LP	1.5551E-02	RHR TRAIN B FAILS (LLOCA/MLOCA)	
LPF	LP	1.0000E+00	RHR GUARANTEED FAILURE	
LV1	LV	3.5050E-04	ALL CONDITIONS (NO SUPPORT REQUIRED)	
LW1	LW	4.2680E-04	FLOW INTO RWST	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
LW2	LW	0.0000E+00	GUARANTEED SUCCESS	
LW3	LW	4.2590E-04	MOV SUPPORT NOT AVAILABLE	
MC1	MC	1.0000E-02	FRACTION OF TIME MODERATOR COEFFICIENT <-7	
MS0	MS	0.0000E+00	no description entered	
MS1	MS	1.1690E-02	no description entered	
MS2	MS	9.9990E-01	no description entered	
MS2A	MS	1.3199E-04	no description entered	
MSF	MS	1.0000E+00	no description entered	
MU1	MU	5.3512E-02	POWER AVAILABLE AT AC BUSES G AND H	
MU2	MU	1.8920E-01	POWER AVAILABLE AT AC BUSES G & H (MAKEUP VIA RFW PUMP)	
MUF	MU	1.0000E+00	GUARANTEED FAILURE	
MUV	MU	1.6200E-02	ILOCA - FAILURE TO ESTABLISH MAKEUP TO RWST	
NI0	NI	0.0000E+00	GUARANTEED SUCCESS	
NIF	NI	1.0000E+00	GUARANTEED FAILURE	
NM0	NM	0.0000E+00	GUARANTEED SUCCESS	
NMF	NM	1.0000E+00	GUARANTEED FAILURE	
NR0	NR	0.0000E+00	GUARANTEED SUCCESS	
NRF	NR	1.0000E+00	GUARANTEED FAILURE	
NV1	NV	1.3253E-04	ALL SUPPORT AVAILABLE	
NV2	NV	2.1366E-03	GIVEN DC 13 OR DC12 FAILED AND OG SUCCEEDED	
NVF	NV	1.0000E+00	GIVEN DC 13 AND 12 FAILED OR OG FAILED	
OB1	OB	2.3263E-01	Loss of Instrument Air	
OB2	OB	2.3263E-01	Loss of Instrument Air, Charging failed	
OB3	OB	5.5550E-01	Loss of 1 DC bus initiating event	
OB3A	OB	7.7450E-02	Failure of Bus 1-1	
OB3ABC	OB	2.3263E-01	Loss of 1 DC Bus	
OB3B	OB	1.4500E-01	Failure of Bus 1-2	
OB3C	OB	3.3300E-01	Failure of Bus 1-3	
OBF	OB	1.0000E+00	Guaranteed Failure	
OC1A	OC	1.2000E-02	OP. RECOVERS RELAY C, ESAM 1	
OC1B	OC	6.0010E-02	OC1 W ESAM 5	
OC1C	OC	3.6010E-01	OC1 W ESAM 30	
OC2A	OC	1.3000E-02	OP. RECOVERS RELAY C	
OC2B	OC	6.5000E-02	OC2 W ESAM = 5	
OC2C	OC	3.9000E-01	OC2 W ESAM = 30	
OC3A	OC	1.1000E-02	OP. RECOVERS RELAY C	
OC3B	OC	5.5000E-02	OC3 W ESAM = 5	
OC3C	OC	3.3000E-01	OC3 W ESAM = 30	
OC4A	OC	1.4000E-02	OP. RECOVERS RELAY C.	
OC4B	OC	7.0000E-02	OC4 W ESAM = 5	
OC4C	OC	4.2000E-01	OC4 W ESAM = 30	
OCF	OC	1.0000E+00	GUARANTEED FAILURE	
ODF	OD	1.0000E+00	no description entered	
OE1	OE	8.1010E-04	OPERATOR FAILS TO INITIATE BORATION	
OE2	OE	8.1010E-04	OPERATOR FAILS TO INITIATE BORATION	
OE3	OE	8.1010E-04	OPERATOR FAILS TO INITIATE BORATION	
OG1	OG	8.8444E-04	ALL SUPPORT AVAILABLE	
OGF	OG	1.0000E+00	GUARANTEED FAILURE	
OI1	OI	1.0000E+00	no description entered	
OI2	OI	1.0000E+00	no description entered	
OI3A	OI	9.3010E-03	OI3 W ESAM 1	
OI3B	OI	4.6510E-02	OI3 W ESAM 5	
OI3C	OI	2.7900E-01	OI3 W ESAM 30	
OIF	OI	1.0000E+00	no description entered	
OL1	OL	1.6990E-02	FAILURE TO DEPRESSURIZE RCS	
OLF	OL	1.0000E+00	GUARANTEED FAILURE TO DEPRESSURIZE RCS	
OP1	OP	2.0000E-03	OPERATOR FAILS TO TERMINATE SAFETY	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description.................	ISF......
			INJECTION	
OP2	OP	1.4000E-02	OPERATOR FAILS TO TERMINATE SAFETY INJECTION	
OS0	OS	0.0000E+00	GUARANTEED SUCCESS	
OS1	OS	1.5260E-02	MANUAL SI ACTUATION WITH ONE SSPS TRAIN UNAVAIL	
OSF	OS	1.0000E+00	GUARANTEED FAILURE	
OV1	OV	1.1000E-02	FAILURE TO DIAGNOSE AND ISOLATE ILOCA	
OVF	OV	1.0000E+00	GUARANTEED FAILURE TO ISOLATE ILOCA	
PA1	PA	1.2098E-02	1/2 PORVS AND 3/3 SRVS	
PA2	PA	1.8366E-02	1/1 PORVS AND 3/3 SRVS	
PA3	PA	4.0994E-02	1/2 PORVS AND 3/3 SRVS; BLK VLVS NOT AVAIL.	
PA4	PA	3.3117E-02	1/1 PORVS AND 3/3 SRVS; BLK VLVS NOT AVAIL.	
PB1	PB	2.6710E-02	2/2 PORVS AND 3/3 SRVS	
PB2	PB	5.5549E-02	2/2 PORVS AND 3/3 SRVS; BLK VLVS NOT AVAIL.	
PCC1	PCC	1.0872E-02	2/2 PORVS AND 2/3 SRVS (OR 3/3 SRVS) LOSP	
PCC2	PCC	1.0790E-02	3/3 SRVS LOSS OF IA AND 1 SUPPORT TRAIN	
PCC3	PCC	9.7660E-03	3/3 SRVS LOSS OF IA AND ALL PORV SUPPORT	
PCC4	PCC	4.0345E-02	2/2 PORVS AND 2/3 SRVS (OR 3/3 SRVS) NO BLK VLVS	
PCC5	PCC	2.5526E-02	3/3 SRVS NO BLK VLVS	
PL1	PL	0.0000E+00	POWER LEVEL GREATER THAN 80%	
PO1	PO	2.3318E-03	1/2 PORV'S ATWT, BORATION, ALL SUPPORT, AFW AVAIL.	
PO2	PO	4.5821E-02	2/2 PORV'S ATWT, BORATION, NO BLOCK VALVES, NO AFW	
PO3	PO	3.1228E-02	1/2 PORVS ATWT, BORATION, NO BLOCK VALVES,	
POF	PO	1.0000E+00	GUARANTEED FAILURE	
PR0	PR	0.0000E+00	Guaranteed Success	
PR1	PR	2.0526E-03	1/2 PORV'S or 1/3 SRV'S, LOSP OR SGTR	
PR2	PR	1.2100E-02	1/2 PORV'S AND 3/3 SRV'S	PA1
PR3	PR	2.6710E-02	2/2 PORV'S AND 3/3 SRV'S	PB1
PR4	PR	1.0870E-02	2/2 PORV'S AND 2/2 PORV'S AND 2/3 SRV'S OR (3/3 SRV'S)	PCC1
PR5	PR	1.2860E-02	1/2 PORV'S OR (1/3 SRV'S), HPI OR SLB	PRW5
PR6	PR	1.0919E-03	1/1 PORV OR (1/3 SRV'S), LOSP OR SGTR	
PR7	PR	1.8360E-02	1/1 PORV AND 3/3 SRV'S	PA2
PR8	PR	1.0790E-02	3/3 SRV'S	PCC2
PR9	PR	8.6060E-03	1/1 PORV OR (1/3 SRV'S), HPI OR SLB	PRW9
PRA	PR	8.8045E-03	1/3 SRV'S	
PRB	PR	9.7650E-03	3/3 SRV'S	PCC3
PRC	PR	2.9240E-01	1/3 SRV'S	PRWC
PRD	PR	3.0949E-02	1/2 PORV'S OR (1/3 SRV'S), LOSP/SGTR, NO BLK VLVS	
PRE	PR	4.1000E-02	1/2 PORV'S AND 3/3 SRV'S BLK VLVS NOT AVAIL.	PA3
PRF	PR	1.0000E+00	GUARANTEED FAILURE	
PRG	PR	5.5560E-02	2/2 PORV'S AND 3/3 SRV'S BLK VLVS NOT AVAIL.	PB2
PRH	PR	4.0350E-02	2/2 PORV'S AND 2/3 SRV'S OR (3/3 SRV'S) NO BLK VLVS	PCC4
PRI	PR	1.9280E-01	1/2 PORV'S OR (1/3 SRV'S), HPI OR SLB NO BLK VLVS	PRWI
PRJ	PR	1.5843E-02	1/1 PORV OR (1/3 SRV'S), LOSP/SGTR, NO BLK VLVS	
PRK	PR	3.3120E-02	1/1 PORV AND 3/3 SRV'S NO BLK VLVS	PA4
PRL	PR	2.5530E-02	3/3 SRV'S NO BLK VLVS	PCC5
PRM	PR	1.0050E-01	1/1 PORV OR (1/3 SRV'S), HPI OR SLB NO BLK VLVS	PRWM

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description.................	ISF......
PRP	PR	1.9610E-07	1/2 PORV'S OR (1/3 SRV'S), MANUAL REACTOR TRIP	PRPP
PRPP	PR	2.0526E-03	no description entered	
PRQ	PR	1.0430E-07	1/1 PORV OR (1/3 SRV'S), MANUAL REACTOR TRIP	PRQQ
PRQQ	PR	1.0919E-03	no description entered	
PRR	PR	8.4120E-07	1/3 SRV'S, MANUAL REACTOR TRIP	PRRR
PRRR	PR	8.8045E-03	no description entered	
PRS	PR	2.9580E-06	1/2 PORV'S OR (1/3 SRV'S), MANUAL REACTOR TRIP	PRSS
PRSS	PR	3.0949E-02	no description entered	
PRT	PR	1.5140E-06	1/1 PORV OR (1/3 SRV'S), MANUAL REACTOR TRIP	PRTT
PRTT	PR	1.5843E-02	no description entered	
PRU	PR	8.9040E-02	1/1 BLOCK VALVE CLOSES, ALL SUPPORT AVAILABLE	
PRW5	PRW	1.2856E-02	no description entered	
PRW9	PRW	8.6068E-03	no description entered	
PRWC	PRW	2.9236E-01	no description entered	
PRWI	PRW	1.9276E-01	no description entered	
PRWM	PRW	1.0054E-01	no description entered	
RA1	RA	2.2630E-01	AS1 / ASD	
RA2	RA	1.8990E-01	AS2 / ASC	
RA3	RA	2.2140E-01	AS3 / ASE	
RA4	RA	4.4070E-01	(AS4 + CC5) / ASF	
RA5	RA	1.9200E-01	AS5 / ASC	
RA6	RA	2.5560E-01	AS6 / ASA	
RA7	RA	4.4080E-01	(AS7 + CC5) / ASF	
RA8	RA	2.1580E-01	AS8 / ASC	
RA9	RA	2.4290E-01	AS9 / ASC	
RAA	RA	1.0000E+00	ASA / ASA	
RAB	RA	4.5860E-01	(ASB + CC5) / ASF	
RAC	RA	1.0000E+00	ASC / ASC	
RAD	RA	8.9700E-03	(AS3 + CC3F|CC3G) / ASF	
RAZ	RA	1.0000E+00	ASW NOT RECOVERED	
RC1	RC	7.0526E-05	BOTH RHR PUMP TRAINS OPERABLE	
RC2	RC	1.7974E-03	ONE RHR PUMP TRAIN OPERABLE	
RCF	RC	1.0000E+00	GUARANTEED FAILURE	
RE1	RE	1.8780E-01	RESLC1 * REOB1	
RE2	RE	4.7830E-02	RESLC2	
RE3	RE	2.3460E-03	RESQ8	
RE4	RE	6.8960E-04	ZHERE2 * REAC06	
RE6	RE	3.7000E-03	ZHESV3	
RE7	RE	3.9080E-01	ZHERP2 + SE3	
RE8	RE	3.0300E-04	ZHERE2 * RESLC3	
RE9	RE	1.4000E-02	ZHEOB2	
REA	RE	3.6940E-01	RSEQ24	
REB	RE	2.3460E-03	RSEQ25	
REC	RE	6.4960E-02	ZHEAW3 + AW4	
RED	RE	3.1420E-01	REAC06	
REE	RE	3.0300E-04	RSEQ34	
REF	RE	5.0840E-01	RESLC1	
REG	RE	4.2600E-03	ZHEAW4 + AW3A	
REH	RE	8.6400E-03	ZHEF06 * RESLC1	
REI	RE	9.6820E-04	ZHESV3 * REAC12	
REJ	RE	3.6940E-01	REOB1	
REK	RE	3.8010E-03	ZHEAW4 + AWD	
REL	RE	6.2260E-02	ZHEAW4 + AW4	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
REN	RE	4.7500E-02	2 * CH2	
REP	RE	1.8920E-01	MU2	
REQ	RE	8.5710E-02	ZHEAW3 + AWAA	
REZ	RE	1.0000E+00	NOT RECOVERED	
RF1	RF	3.3169E-02	SWITCHOVER AFTER SLOCA OR B/F WITH CS FAILED	
RF2	RF	3.6169E-02	SWITCHOVER AFTER SLOCA OR B/F WITH CS SUCCESS	
RF3	RF	3.9827E-02	SWITCHOVER AFTER LLOCA OR MLOCA INITIATING EVENT	
RF4	RF	1.8372E-01	SWITCHOVER TO RECIRCULATION AFTER CORE MELT	
RFF	RF	1.0000E+00	GUARANTEED FAILURE	
RP0	RP	0.0000E+00	no description entered	
RP1	RP	1.0000E+00	no description entered	
RP2	RP	9.5500E-01	TRIP RCP'S IF CCW IS LOST	
RPF	RP	1.0000E+00	no description entered	
RS1	RS	1.0000E+00	no description entered	
RSF	RS	1.0000E+00	no description entered	
RT0	RT	0.0000E+00	GUARANTEED SUCCESS	
RT1	RT	8.7679E-06	1/2 TRAINS (BOTH SSPS SIGNALS GENERATED)	
RT2	RT	8.9931E-06	1/2 TRAINS (DC POWER LOST TO ONE SHUNT TRIP COILS)	
RT3	RT	1.5620E-05	1/2 TRAINS (DC POWER LOST TO BOTH SHUNT TRIP COILS)	
RT4	RT	1.1396E-04	1/1 TRAIN (ONLY ONE SSPS SIGNAL GENERATED)	
RT5	RT	1.6618E-04	1/1 TRAIN (ONE SSPS SIGNAL, LOP TO SHUNT TRIP COIL)	
RT6	RT	6.1630E-06	GRAVITY INSERTION (INSUFFICIENT POWER TO PREVENT INSERTION)	
RT7	RT	2.0388E-02	OPERATOR INITIATED (DC POWER LOST TO BOTH SHUNT COILS)	
RTF	RT	1.0000E+00	GUARANTEED FAILURE	
RW1	RW	2.7763E-05	ALL CONDITIONS (NO SUPPORT REQUIRED)	
RWF	RW	1.0000E+00	SRW=F	
S11A	S1	1.0318E-02	SSPS TRAIN A FAILS (GENERAL TRANSIENT)	
S11B	S1	1.0318E-02	SSPS TRAIN B FAILS (GENERAL TRANSIENT)	
S12	S1	1.9850E-04	SSPS TRAIN A&B FAIL (GENERAL TRANSIENT)	
S21A	S2	1.5780E-02	SSPS TRAIN A FAILS - ALL SUPPORT (LLOCA)	
S21B	S2	1.5780E-02	SSPS TRAIN B FAILS - ALL SUPPORT (LLOCA)	
S22A	S2	2.2268E-02	SSPS TRAIN A FAILS - IC2&3 UNAVAI(LLOCA)	
S22B	S2	2.2268E-02	SSPS TRAIN B FAILS - IC2&3 UNAVAI (LLOCA	
S23	S2	8.4435E-04	SSPS TRAIN A&B FAIL -ALL SUPPORT (LLOCA)	
S24	S2	7.3381E-03	SSPS TRAIN A&B FAIL-IC2&3 UNAVAI (LLOCA)	
S31A	S3	1.6638E-02	SSPS TRAIN A FAILS (SGTR)	
S31B	S3	1.6638E-02	SSPS TRAIN B FAILS (SGTR)	
S32	S3	4.5386E-04	SSPS TRAIN A&B FAIL (SGTR)	
S41A	S4	1.9332E-02	SSPS TRAIN A FAILS - ALL SUPPORT (SLBI)	
S41B	S4	1.9332E-02	SSPS TRAIN B FAILS - ALL SUPPORT (SLBI)	
S42A	S4	2.5820E-02	SSPS TRAIN A FAILS -IC2&3 UNAVAIL (SLBI)	
S42B	S4	2.5820E-02	SSPS TRAIN B FAILS -IC2&3 UNAVAIL (SLBI)	
S43	S4	9.0493E-04	SSPS TRAIN A&B FAIL - ALL SUPPORT (SLBI)	
S44	S4	7.3987E-03	SSPS TRAIN A&B FAIL-IC2&3 UNAVAIL (SLBI)	
S51A	S5	1.6635E-02	SSPS TRAIN A FAILS (SLBO)	
S51B	S5	1.6635E-02	SSPS TRAIN B FAILS (SLBO)	
S52	S5	4.5004E-04	SSPS TRAIN A&B FAIL (SLBO)	
S61A	S6	1.6635E-02	SSPS TRAIN A FAILS (SLOCA)	
S61B	S6	1.6635E-02	SSPS TRAIN B FAILS (SLOCA)	
S62	S6	4.5004E-04	SSPS TRAIN A&B FAIL (SLOCA)	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
SA0	SA	0.0000E+00	GUARANTEED SUCCESS	
SA1	SA	1.0320E-02	GENERAL TRANSIENT	S11A
SA2	SA	1.5780E-02	LLOCA - ALL SUPPORT AVAILABLE	S21A
SA3	SA	2.2270E-02	LLOCA - IC II&III UNAVAILABLE	S22A
SA4	SA	1.6630E-02	SGTR	S31A
SA5	SA	1.9330E-02	SLBI - ALL SUPPORT AVAILABLE	S41A
SA6	SA	2.5820E-02	SLBI - IC II&III UNAVAILABLE	S42A
SA7	SA	1.6630E-02	SLBO	S51A
SA8	SA	1.6630E-02	SLOCA	S61A
SACSF1	SACSF	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SACSF2	SACSF	4.9900E-03	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SACSF3	SACSF	2.8700E-02	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SACSF4	SACSF	8.3800E-02	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SACSF5	SACSF	2.4100E-01	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SACSF6	SACSF	4.6900E-01	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SACSS1	SACSS	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SACSS2	SACSS	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SACSS3	SACSS	8.3600E-03	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SACSS4	SACSS	3.9200E-02	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SACSS5	SACSS	1.4900E-01	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SACSS6	SACSS	3.3000E-01	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SAF	SA	1.0000E+00	GUARANTEED FAILURE	
SAS1	SAS	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SAS2	SAS	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SAS3	SAS	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SAS4	SAS	0.0000E+00	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SAS5	SAS	0.0000E+00	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SAS6	SAS	8.0100E-03	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SAW1	SAW	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SAW2	SAW	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SAW3	SAW	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SAW4	SAW	0.0000E+00	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SAW5	SAW	1.5200E-03	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SAW6	SAW	2.8600E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SB0	SB	0.0000E+00	GUARANTEED SUCCESS	
SB1	SB	1.0230E-02	SA-S (GENERAL TRANSIENT)	S11B
SB2	SB	1.9240E-02	SA-F (GENERAL TRANSIENT)	S12
SB3	SB	1.0320E-02	SA-B (IC I UNAVAILABLE) (GENERAL TRANS)	S11B
SB4	SB	1.5180E-02	SA-S ALL SUPPORT (LLOCA)	S21B
SB5	SB	1.5270E-02	SA-S II&III UNAVAILABLE (LLOCA)	S22B
SB6	SB	5.3500E-02	SA-F ALL SUPPORT AVAILABLE (LLOCA)	S23
SB7	SB	3.2950E-01	SA-F II&III UNAVAILABLE (LLOCA)	S24
SB8	SB	2.2270E-02	SA-B II&III UNAVAILABLE (LLOCA)	S22B
SB9	SB	1.6450E-02	SA-S (SGTR)	S31B
SBA	SB	2.7280E-02	SA-F (SGTR)	S32
SBB	SB	1.6630E-02	SA-B (IC I UNAVAILABLE - SGTR)	S31B
SBC	SB	1.8790E-02	SA-S ALL SUPPORT AVAILABLE (SLBI)	S41A
SBD	SB	1.8910E-02	SA-S II&III UNAVAILABLE (SLBI)	S42B
SBE	SB	4.6810E-02	SA-F ALL SUPPORT AVAILABLE (SLBI)	S43
SBF	SB	1.0000E+00	GUARANTEED FAILURE	
SBG	SB	2.8650E-01	SA-F II&III UNAVAILABLE (SLBI)	S44
SBH	SB	2.5820E-02	SA-B II&III UNAVAILABLE (SLBI)	S42B
SBI	SB	1.6450E-02	SA-S (SLBO)	S51B
SBJ	SB	2.7060E-02	SA-F (SLBO)	S52
SBK	SB	1.6630E-02	SA-B (IC I UNAVAILABLE - SLBO)	S51B
SBL	SB	1.6450E-02	SA-S (SLOCA)	S61B
SBM	SB	2.7060E-02	SA-F (SLOCA)	S62
SBN	SB	1.6630E-02	SA-B (IC I UNAVAILABLE - SLOCA)	S61B

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
SCB1	SCB	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SCB2	SCB	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SCB3	SCB	1.2500E-05	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SCB4	SCB	2.0000E-03	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SCB5	SCB	7.4600E-03	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SCB6	SCB	3.5000E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SCC1	SCC	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SCC2	SCC	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SCC3	SCC	3.9400E-04	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SCC4	SCC	3.4200E-03	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SCC5	SCC	1.9800E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SCC6	SCC	1.5800E-01	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SCH1	SCH	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SCH2	SCH	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SCH3	SCH	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SCH4	SCH	0.0000E+00	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SCH5	SCH	0.0000E+00	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SCH6	SCH	1.0900E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SCP1	SCP	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SCP2	SCP	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SCP3	SCP	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SCP4	SCP	0.0000E+00	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SCP5	SCP	0.0000E+00	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SCP6	SCP	8.6300E-04	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SCS1	SCS	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SCS2	SCS	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SCS3	SCS	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SCS4	SCS	3.0500E-04	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SCS5	SCS	4.2200E-03	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SCS6	SCS	1.7700E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SCT1	SCT	2.5300E-04	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SCT2	SCT	2.0100E-02	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SCT3	SCT	6.8300E-02	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SCT4	SCT	1.3400E-01	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SCT5	SCT	2.7200E-01	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SCT6	SCT	4.5100E-01	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SCTO	SCT	0.0000E+00	GUARANTEED SUCCESS WHEN 4KV SG FAILED	
SCV1	SCV	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SCV2	SCV	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SCV3	SCV	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SCV4	SCV	3.4800E-04	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SCV5	SCV	1.5100E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SCV6	SCV	3.4400E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SDC1	SDC	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SDC2	SDC	9.1400E-05	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SDC3	SDC	2.2400E-03	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SDC4	SDC	8.4700E-03	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SDC5	SDC	3.8200E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SDC6	SDC	1.2600E-01	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SDG1	SDG	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SDG2	SDG	1.1700E-04	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SDG3	SDG	4.1500E-03	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SDG4	SDG	1.6900E-02	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SDG5	SDG	6.6700E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SDG6	SDG	1.8300E-01	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SE0	SE	0.0000E+00	GUARANTEED SUCCESS	
SE1A	SE	1.4920E-02	CCW UNAVAIL WITH TWO CCP'S AVAIL	
SE1B	SE	6.2928E-02	SE1 W ESAM 5	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
SE1C	SE	3.6302E-01	SE1 W ESAM 30	
SE2	SE	0.0000E+00	CCW AVAIL AND SEAL COOLING ASSUMED SUCCESSFUL	
SE3A	SE	2.9591E-02	CCW UNAVAIL WITH ONE CCP AVAIL	
SE3B	SE	7.7065E-02	SE3 W ESAM 5	
SE3C	SE	3.7382E-01	SE3 W ESAM 30	
SEF	SE	1.0000E+00	GUARANTEED FAILURE	
SEL1	SEL	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SEL2	SEL	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SEL3	SEL	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SEL4	SEL	7.2200E-05	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SEL5	SEL	2.9200E-03	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SEL6	SEL	2.9900E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SELF	SEL	1.0000E+00	SSG=F	
SF0	SF	0.0000E+00	GUARANTEED SUCCESS	
SF1	SF	1.3510E-03	ALL SUPPORT AVAILABLE (W/RCVRY)	AB1F
SFA	SF	1.4370E-03	ALL SUPPORT AVAILABLE (NO RCVRY)	AB4F
SFC1	SFC	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SFC2	SFC	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SFC3	SFC	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SFC4	SFC	0.0000E+00	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SFC5	SFC	5.5300E-04	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SFC6	SFC	5.9500E-03	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SFF	SF	1.0000E+00	Guaranteed Failure	
SFO1	SFO	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SFO2	SFO	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SFO3	SFO	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SFO4	SFO	0.0000E+00	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SFO5	SFO	3.1200E-04	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SFO6	SFO	1.1000E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SG0	SG	0.0000E+00	GUARANTEED SUCCESS	
SG1	SG	1.3470E-03	SF-S (W/RCVRY)	AB1G
SG2	SG	4.0450E-03	SF-F (W/RCVRY)	AB2FG
SG3	SG	1.3510E-03	SF-B (W/RCVRY)	AB1G
SGA	SG	1.3840E-03	SF-S (NO RCVRY)	AB4G
SGB	SG	3.8240E-02	SF-F (NO RCVRY)	AB5FG
SGC	SG	1.4370E-03	SF-B (NO RCVRY)	AB4G
SGF	SG	1.0000E+00	Guaranteed Failure	
SH0	SH	0.0000E+00	GUARANTEED SUCCESS	
SH1	SH	1.3450E-03	SF-S, SG-S (W/RCVRY)	AB1H
SH2A	SH	3.3680E-03	SF-S, SG-F (W/RCVRY)	AB2GH
SH2B	SH	3.3680E-03	SF-F, SG-S (W/RCVRY)	AB2FH
SH3	SH	1.7060E-01	SF-F, SG-F (W/RCVRY)	AB3FGH
SH4A	SH	1.3470E-03	SF-S, SG-B (W/RCVRY)	AB1H
SH4B	SH	1.3470E-03	SF-B, SG-S (W/RCVRY)	AB1H
SH5A	SH	4.0450E-03	SF-F, SG-B (W/RCVRY)	AB2FH
SH5B	SH	4.0450E-03	SF-B, SG-F (W/RCVRY)	AB2GH
SH6	SH	1.3510E-03	SF-B, SG-B (W/RCVRY)	AB1H
SHA	SH	1.3450E-03	SF-S, SG-S (NO RCVRY)	AB4H
SHBA	SH	3.0010E-02	SF-S, SG-F (NO RCVRY)	AB5GH
SHBB	SH	3.0010E-02	SF-F, SG-S (NO RCVRY)	AB5FH
SHC	SH	2.4520E-01	SF-F, SG-F (NO RCVRY)	AB6FGH
SHDA	SH	1.3840E-03	SF-S, SG-B (NO RCVRY)	AB4H
SHDB	SH	1.3840E-03	SF-B, SG-S (NO RCVRY)	AB4H
SHEA	SH	3.8240E-02	SF-F, SG-B (NO RCVRY)	AB5FH
SHEB	SH	3.8240E-02	SF-B, SG-F (NO RCVRY)	AB5GH
SHF	SH	1.0000E+00	Guaranteed Failure	
SHG	SH	1.4370E-03	SF-B, SG-B (NO RCVRY)	AB4H

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
SI1	SI	2.0074E-03	ALL SUPPORT AVAILABLE (1/2)	
SI2A	SI	1.3563E-02	SI PUMP 1-1 AVAILABLE (PUMP 1-2 UNAVAIL)	
SI2B	SI	1.3563E-02	SI PUMP 1-2 AVAILABLE (PUMP 1-1 UNAVAIL)	
SI3	SI	2.5136E-02	MEDIUM LOCA; ALL SUPPORT AVAILABLE, CH FAILED. (2/2)	
SID1	SID	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SID2	SID	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SID3	SID	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SID4	SID	7.0500E-04	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SID5	SID	6.0500E-03	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SID6	SID	2.4200E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SIF	SI	1.0000E+00	GUARANTEED FAILURE	
SL1	SL	3.8790E-03	ALL SUPPORT AVAILABLE	
SL2	SL	4.0710E-03	LOSS OF SUPPORT TO 10% STEAM DUMP VALVES	
SM1	SM	1.4790E-02	2700 GPM LEAK RATE NORMALIZED TO 150	
SM2	SM	1.3270E-02	2700 GPM LEAK RATE NORMALIZED TO 150	
SOP1	SOP	1.9400E-02	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SOP2	SOP	5.2700E-01	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SOP3	SOP	8.2700E-01	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SOP4	SOP	9.2700E-01	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SOP5	SOP	9.8400E-01	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SOP6	SOP	9.9700E-01	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SPR1	SPR	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SPR2	SPR	3.5100E-04	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SPR3	SPR	3.1800E-03	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SPR4	SPR	7.9200E-03	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SPR5	SPR	2.8900E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SPR6	SPR	8.4200E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SPT1	SPT	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SPT2	SPT	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SPT3	SPT	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SPT4	SPT	2.5400E-04	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SPT5	SPT	3.3800E-03	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SPT6	SPT	1.2500E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SR1	SR	5.1125E-02	1/2 TRAINS OPERATES (ALL SUPPORT AVAILABLE)	
SR2	SR	5.7638E-02	1/1 TRAIN OPERATES (LOSS OF ONE VITAL BUS OR SSPS TRAIN)	
SRF	SR	1.0000E+00	GUARANTEED FAILURE	
SRT1	SRT	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SRT2	SRT	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SRT3	SRT	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SRT4	SRT	0.0000E+00	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SRT5	SRT	0.0000E+00	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SRT6	SRT	6.4200E-05	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SRW1	SRW	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SRW2	SRW	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SRW3	SRW	7.9200E-04	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SRW4	SRW	4.5000E-03	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SRW5	SRW	2.0300E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SRW6	SRW	1.2100E-01	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SSE1	SSE	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SSE2	SSE	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SSE3	SSE	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SSE4	SSE	0.0000E+00	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SSE5	SSE	1.8800E-03	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SSE6	SSE	3.3800E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SSF	SS	1.0000E+00	RCS PRESSURE BELOW 1850# FOLLOWING ATWT	
SSG1	SSG	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
SSG2	SSG	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SSG3	SSG	3.8200E-04	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SSG4	SSG	3.0000E-03	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SSG5	SSG	1.1200E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SSG6	SSG	3.1900E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SSH1	SSH	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SSH2	SSH	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SSH3	SSH	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SSH4	SSH	0.0000E+00	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SSH5	SSH	0.0000E+00	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SSH6	SSH	3.5000E-03	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SSV1	SSV	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SSV2	SSV	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SSV3	SSV	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SSV4	SSV	3.4800E-04	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SSV5	SSV	1.4800E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SSV6	SSV	3.1000E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
STRUT1	STRUT	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
STRUT2	STRUT	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
STRUT3	STRUT	4.0200E-04	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
STRUT4	STRUT	3.3300E-03	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
STRUT5	STRUT	1.3500E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
STRUT6	STRUT	4.0300E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
SV0	SV	0.0000E+00	GUARANTEED SUCCESS (STATION BLACKOUT SCENARIOS)	
SV1A	SV	2.5510E-07	1/2 TRAINS; OSP, 480V 1F, 1H AVAILABLE	
SV1B	SV	1.2755E-06	SV1A W/ ESAM=5	
SV1C	SV	7.6528E-06	SV1A W/ ESAM=30	
SV2A	SV	1.3951E-04	1/1 TRAIN START AND RUN; 480V BUS 1F UNAVAILABLE	
SV2B	SV	6.9773E-04	SV2A W/ ESAM=5	
SV2C	SV	4.1862E-03	SV2A W/ ESAM=30	
SV3A	SV	1.3566E-04	1/1 TRAIN CONTINUE TO RUN; 480 V BUS 1H UNAVAIL.	
SV3B	SV	6.7850E-04	SV3A W/ ESAM=5	
SV3C	SV	4.0708E-03	SV3A W/ ESAM=30	
SV4A	SV	1.0826E-06	1/2 TRAINS START AND RUN; LOSP, 480V BUS 1F, 1H AVAILABLE	
SV4B	SV	5.4134E-06	SV4A W/ ESAM=5	
SV4C	SV	3.2477E-05	SV4A W/ ESAM=30	
SV5A	SV	9.9890E-04	ONLY RECOVERY POSSIBLE, BUS 1F, 1H UNAVAILABLE	
SV5B	SV	4.9954E-03	SV5A W/ ESAM=5	
SV5C	SV	2.9974E-02	SV5A W/ ESAM=30	
SVBI	SVB	3.1985E-04	INITIATING EVENT	
SVF	SV	1.0000E+00	GUARANTEED FAIL	
SVI1	SVI	0.0000E+00	SEIS1, g Levels: 2.01E-01 to 1.25E-00	
SVI2	SVI	0.0000E+00	SEIS2, g Levels: 1.25E-00 to 1.75E+00	
SVI3	SVI	0.0000E+00	SEIS3, g Levels: 1.75E+00 to 2.00E+00	
SVI4	SVI	1.9400E-03	SEIS4, g Levels: 2.00E+00 to 2.500E+00	
SVI5	SVI	1.0800E-02	SEIS5, g Levels: 2.500E+00 to 3.00E+00	
SVI6	SVI	3.5800E-02	SEIS6, g Levels: 3.00E+00 to 3.99E+00	
TD1	TD	5.8660E-02	SCT=F	AW4
TD2	TD	1.1970E-01	SCT=F +	AW9
TDF	TD	1.0000E+00	SCT=F*DG=F	
TF0	TF	0.0000E+00	GUARANTEED SUCCESS	
TF1	TF	9.0870E-02	DG 2-3 (BUS F) STARTS & RUNS FOR 6 HR	
TFF	TF	1.0000E+00	GUARANTEED FAILURE	

Table 3-4. Point Estimate Split Fraction Values

MODEL Name: DC93SEIS

Master Frequency File: SEIS93R7

SF Name...	Top.......	SF Value...	Split Fraction Description................	ISF......
TG0	TG	0.0000E+00	GUARANTEED SUCCESS	
TG1	TG	8.9700E-02	DG 2-1 (BUS G) : TF-S	
TG2	TG	1.0250E-01	DG 2-1 (BUS G) : GF-F	
TG3	TG	9.0870E-02	DG 2-1 (BUS G) : GF-B	
TGF	TG	1.0000E+00	GUARANTEED FAILURE	
TH0	TH	0.0000E+00	GUARANTEED SUCCESS	
TH1	TH	8.8810E-02	DG 2-2 (BUS H) : TF-S,TG-S	
TH2	TH	9.8730E-02	DG 2-2 (BUS H) : TF-S/F,TG-F/S	
TH3	TH	1.3540E-01	DG 2-2 (BUS H) : TF-F,TG-F	
TH4	TH	8.9700E-02	DG 2-2 (BUS H) : TF-S/B,TG-B/S	
TH5	TH	1.0250E-01	DG 2-2 (BUS H) : TF-F/B,TG-B/F	
TH6	TH	9.0870E-02	DG 2-2 (BUS H) : TF-B,TG-B	
THF	TH	1.0000E+00	GUARANTEED FAILURE	
TT0	TT	0.0000E+00	no description entered	
TT1	TT	1.4419E-05	no description entered	
TT2	TT	1.3148E-03	no description entered	
TT3	TT	5.0108E-03	no description entered	
TT4	TT	3.3004E-03	no description entered	
TT5	TT	4.6009E-03	no description entered	
TT6	TT	8.2969E-03	no description entered	
TTF	TT	1.0000E+00	no description entered	
VA1	VA	4.8330E-03	ALL SUPPORT AVAILABLE	VV2A
VAF	VA	1.0000E+00	GUARANTEED FAILURE	
VB1	VB	4.5950E-03	ALL SUPPORT AVAILABLE (VA SUCCESSFUL)	VV2B
VB2	VB	5.3830E-02	ALL SUPPORT AVAILABLE (VA FAILED)	VV1
VB3	VB	4.8330E-03	VA GUARANTEED FAILURE	VV2B
VBF	VB	1.0000E+00	GUARANTEED FAILURE	
VC1	VC	6.8930E-02	560 GPM LEAK RATE NORMALIZED TO 150 GPM	
VCF	VC	1.0000E+00	LEAK RATE > RELIEF CAPACITY	
VDI	VD	1.4480E-06	VD IE FREQUENCY - 150 GPM LEAK (DISCHARGE VLVS)	
VI0	VI	0.0000E+00	VESSEL INTEGRITY	
VI1	VI	1.1000E-04	VESSEL INTEGRITY UNCONTROLLED COOLDOWN (TT&	
VI2	VI	2.2000E-02	VESSEL INTEGRITY LOSS OF SECONDARY HEAT SINK	
VI3	VI	2.0000E-03	VESSEL INTEGRITY MEDIUM LOCA EVENTS	
VI4	VI	1.8000E-06	VESSEL INTEGRITY SGTR WITH SUCCESSFUL ECCS	
VI5	VI	9.0000E-03	VESSEL INTEGRITY SGTR WITH DELAYED ECCS TER	
VO1	VO	7.2600E-05	3/3 RELIEF VLVS OPEN ON DEMAND	
VO2	VO	4.8400E-05	2/2 RELIEF VLVS OPEN ON DEMAND	
VSI	VS	4.1240E-07	VS IE FREQUENCY - 150 GPM LEAK (SUCTION VLVS)	
VV1	VV	2.6017E-04	CONTAINMENT SUMP TRAIN A AND B FAILS	
VV2A	VV	4.8332E-03	CONTAINMENT SUMP TRAIN A FAILS	
VV2B	VV	4.8332E-03	CONTAINMENT SUMP TRAIN B FAILS	
VVF	VV	1.0000E+00	GUARANTEED FAILURE	
WL1	WL	6.9048E-05	ALL SUPPORT SYSTEMS AVAILABLE	
WL2	WL	1.0156E-03	SSPS TRAIN A OR B UNAVAILABLE	
WL3	WL	1.0000E+00	SSPS TRAINS A AND B UNAVAILABLE	
WLF	WL	1.0000E+00	GUARANTEED FAILURE	

Table 3-5. Seismic Initiating Events		
SEISMIC INITIATING EVENT DESIGNATOR	SPECTRAL* ACCELERATION LEVEL (g)	FREQUENCY (PER YEAR)
SEIS1	0.0 to 1.25	1.41E-02
SEIS2	1.25 to 1.75	8.00E-04
SEIS3	1.75 to 2.0	1.47E-04
SEIS4	2.0 to 2.5	1.17E-04
SEIS5	2.5 to 3.0	2.82E-05
SEIS6	3.0 to 4.0	7.43E-06
TOTAL SEISMIC INITIATING EVENT FREQUENCY		1.52E-02

* Average 5% damped spectral acceleration over the 3-8.5 Hz frequency range

Table 3-6. Seismic Human Action Values					
Human Action Identifier	Human Action Description	Point Estimate Values Used for Internal Events Analysis	Multiplication Factors Used for Seismic Initiating Events		
			SEIS1 SEIS2	SEIS3 SEIS4	SEIS5 SEIS6
ZHEAC1	Failure to Recover from common cause Startup Breaker Failure on Demand	2.30E-03	30	30	30
ZHEAS1	Failure to remotely crosstie U1 & U2 ASW-Both U1 Pumps Failure	7.10E-03	30	30	30
ZHEAS2	Failure to Locally crosstie U1 & U2 ASW or Fail to Open	5.90E-03	1	5	30
ZHECC1	Failure to Reduce CCW heat Loads with One CCW Pump	8.20E-03	1	5	30
ZHECT1	Failure to Restore AC Power - Relay Chatter	1.20E-02	1	5	30
ZHECT2	Failure to Restore AC Power - Relay Chatter, SLOCA, No AFW	1.30E-02	1	5	30
ZHECT3	Failure to Restore AC Power - Relay Chatter and No AFW	1.10E-02	1	5	30
ZHECT4	Failure to Restore AC Power-Relay Chatter, SLOCA, No AFW	1.40E-02	1	5	30
ZHEF04	Failure to Realign Fuel transfer pump Power Source to Opposite unit	1.00E-02	1	5	30
ZHEF05	Failure to Realign Fuel Oil transfer pump Given 1 pump loss of Power	1.50E-02	1	5	30
ZHEF06	Failure to Align a Dedicate, Portable Fuel Oil Transfer pump	1.70E-02	30	30	30
ZHELA1	Failure to Trip RHR if RCS Pressure is High, for Feed & Bleed	1.90E-03	1	5	30
ZHELA2	Failure to Trip RHR if RCS Pressure is High, for Small LOCA	1.90E-03	1	5	30

Table 3-6. Seismic Human Action Values					
Human Action Identifier	Human Action Description	Point Estimate Values Used for Internal Events Analysis	Multiplication Factors Used for Seismic Initiating Events		
			SEIS1 SEIS2	SEIS3 SEIS4	SEIS5 SEIS6
ZHEMU1	Failure to Align RHR Pump Suction From Hot Leg	1.60E-03	30	30	30
ZHEMU2	Failure to Reduce Injection Flow to RCS & Provide Makeup	6.20E-03	30	30	30
ZHEOB1	Failure to Initiate Bleed and Feed Cooling	7.28E-03	30	30	30
ZHEOB2	Failure to Establish Instrument Air to Containment for Third PORV	1.40E-02	30	30	30
ZHEOI3	Failure to Close Seal Return Valve 8100 -SBO	9.30E-03	1	5	30
ZHEOS1	Failure to Manually Actuate SI Equipment one or Both SSPS trains Failed	5.09E-04	30	30	30
ZHEPR3	Failure to Isolate an Isolable LOCA with PORV Block Valve	2.90E-03	30	30	30
ZHEPR4	Failure to Isolate Stuck Open PORV W/O Reactor Trip	2.10E-03	30	30	30
ZHERF1	Failure to Switch to Recirculation from Injection Mode SLOCA w/spray	1.10E-03	30	30	30
ZHERF2	Failure to Switch to Recirculation from Injection Mode (ECCS) MLOCA/LLOCA	1.30E-03	30	30	30
ZHERF3	Failure to Switch to Recirculation from Injection Mode After core damage	4.51E-03	30	30	30
ZHERF5	Failure to Switch to Recirculation from Injection Mode SLOCA w/spray	1.20E-03	30	30	30
ZHERP1	Failure to Trip RCPS Given CCW Fails	1.50E-03	30	30	30

Table 3-6. Seismic Human Action Values					
Human Action Identifier	Human Action Description	Point Estimate Values Used for Internal Events Analysis	Multiplication Factors Used for Seismic Initiating Events		
			SEIS1 SEIS2	SEIS3 SEIS4	SEIS5 SEIS6
ZHERT1	Failure to Press Manual Push Button to Trip Reactor	6.79E-04	30	30	30
ZHERT2	Failure to Deenergize Bus to Trip Reactor	8.19e-04	30	30	30
ZHESE1	Failure to Align Fire Water to Charging Pumps	1.20E-02	1	5	30
ZHESR1	Failure to align for spray Recirculation Sump Recirculation Success	1.69E-03	30	30	30
ZHESV2	Failure to Open Doors to Inverter & 480V Switchgear Room	9.99E-04	1	5	30
ZHECV1	Failure to Start Standby train of control room ventilation	6.90E-03	30	30	30

Table 3-7. Seismic PRA Component Groupings For Top Events	
Seismic Top Event	Components/Structures
SOP - OFFSITE POWER	OFFSITE POWER, 230 KV
SDC - 125V DC POWER	AUXILIARY BUILDING BATTERIES DC SWITCHGEAR/BREAKER PANEL
STRUT - TURBINE BUILDING STRUT	STRUT FOR TURBINE BUILDING
SACSS - ALL 4KV VITAL AC POWER/STRUT SUCCESS	TURBINE BUILDING SHEAR WALL 4 KV SWITCHGEAR SAFEGUARD RELAY PANEL BATTERY CHARGERS 4KV/480V TRANSFORMERS BUS F POTENTIAL TRANSFORM BLOCK WALLS
SACSF - ALL 4KV VITAL AC POWER/STRUT FAILURE	TURBINE BUILDING SHEAR WALL BATTERY CHARGERS 4KV/480V TRANSFORMERS SWITCHGEAR/STRUT FAILURE SAFEGUARD RELAY PANEL/STR BUS F POTENTIAL TRANSFORM BLOCK WALLS
SDG - ALL SIX DIESEL GENERATORS	DIESEL GENERATORS DG RADIATOR/WATER PUMPS DG EXCITATION CUBICAL DG CONTROL PANEL
SFO - FUEL OIL TRANSFER	DG FUEL OIL PUMPS/FILTERS BOP PIPING AND SUPPORTS
SVI - ALL FOUR VITAL INSTRUMENT CHANNELS	INVERTERS PROCESS CONTROL AND PROTECTION PRESSURE AND DP TRANSMITTERS
SRT - REACTOR TRIP	REACTOR INTERNALS
SPT - PARTIAL REACTOR TRIP	REACTOR TRIP SWITCHGEAR
SCV - CONTROL ROOM VENTILATION	CONTROL ROOM SUPPLY FANS HVAC DUCTING AND SUPPORTS
SCC - COMPONENT COOLING WATER	RHR HEAT EXCHANGERS CCW PUMPS CCW HEAT EXCHANGERS CCW SURGE TANK BOP PIPING AND SUPPORTS

Table 3-7. Seismic PRA Component Groupings For Top Events

Seismic Top Event	Components/Structures
SAS - AUXILIARY SALTWATER	INTAKE STRUCTURE AUXILIARY SALTWATER PIPING BOP PIPING AND SUPPORTS
SSV - PARTIAL 480V SWG VENTILATION	HVAC DUCTING AND SUPPORTS
SCT - RELAY CHATTER	CHATTER, DG CONTROL PANEL CHATTER, 4KV SWITCHGEAR
SEL - EXCESSIVE LOCA	CONCRETE INTERNAL BIOSTRUCTURE REACTOR PRESSURE VESSEL BOP PIPING AND SUPPORTS
SSG - STEAM GENERATORS	STEAM GENERATOR
SID - CR AND HSDP INDICATION	MAIN CONTROL BOARDS HOT SHUTDOWN PANELS
SRW - REFUELING WATER STORAGE	REFUELING WATER STORAGE TANK RHR PUMPS CONTAINMENT SPRAY PUMPS BOP PIPING AND SUPPORTS
SPR - PRESSURIZER RELIEF/SMALL	POWER OPERATED RELIEF VALVE IMPULSE LINES BOP PIPING AND SUPPORTS
SSE - RCP SEAL COOLING	REACTOR COOLANT PUMPS BOP PIPING AND SUPPORTS
SCH - CENTRIFUGAL CHARGING	CENTRIFUGAL CHARGING PUMP BOP PIPING AND SUPPORTS
SAW - AUXILIARY FEEDWATER	AFW PUMP (STEAM DRIVEN) BOP PIPING AND SUPPORTS
SFC - CONTAINMENT FAN COOLER UNITS	AUX RELAY PANELS
SCS - PARTIAL CONTAINMENT SPRAY	SPRAY ADDITIVE TANK
SCP - LARGE CONTAINMENT FAILURE	CONTAINMENT BUILDING
SCB - CCW BYPASS	CONTAINMENT FAN COOLERS BOP PIPING AND SUPPORTS
SSH - CONTAINMENT SPRAY HEADER	BOP PIPING AND SUPPORTS

Table 3-8. DCPP Component and Structure Fragility and HCLPF Values						
RISKMAN ID	Description	Median Accl.(g)	HCLPF	Beta R	Beta U	Failure Mode and Source of Fragility Value
ZCBLDG	CONTAINMENT BUILDING	9.02	3.58	0.26	0.30	Concrete shell cracking, liner break.
ZBISTR	CONCRETE INTERNAL BIOSTRUCTURE	6.91	2.98	0.20	0.31	Shield wall shear cracks, attached components fail to function, reactor vessel failure and reactor trip failure.
ZINSTR	INTAKE STRUCTURE	8.55	3.23	0.28	0.31	Shear wall fails, intake pumps fail to function.
ZABLDG	AUXILIARY BUILDING	5.79	2.67	0.21	0.26	North-South shear wall fails, attached components fail to function.
ZTBSHR	TURBINE BUILDING SHEAR WALL	4.87	1.84	0.26	0.33	Shear wall at column 31 fails, 4kV switchgear fails to function, bracing strut fails, electrical panels fail to function.
ZRWSTK	REFUELING WATER STORAGE TANK	9.54	3.37	0.29	0.34	Failure of tank integrity, loss of contents.
ZASPIP	AUXILIARY SALTWATER PIPING	9.23	4.54	0.22	0.21	Pipe rupture, loss of contents.
ZRXPRV	REACTOR PRESSURE VESSEL	8.71	3.35	0.25	0.33	Line break, loss of coolant, excessive LOCA.
ZRXINT	REACTOR INTERNALS	11.42	4.04	0.40	0.23	Failure of reactor trip and of core cooling geometry.
ZSTMGN	STEAM GENERATOR	7.20	2.63	0.31	0.30	Main steam and other line breaks, excessive LOCA.
ZPORVL	POWER OPERATED RELIEF VALVES	7.62	2.32	0.30	0.42	Fails as is, binding.
ZRCPMP	REACTOR COOLANT PUMPS	8.82	2.83	0.37	0.32	Excessive seal leak, small LOCA.
ZRHRPP	RHR PUMPS	8.31	3.35	0.33	0.22	Failure to function, line break.
ZRHRHX	RHR HEAT EXCHANGERS	8.09	3.49	0.24	0.27	Intake and discharge line ruptures.
ZCCWPP	CCW PUMPS	8.53	3.74	0.29	0.21	Loss of pump function (not a line break).
ZCCWHX	CCW HEAT EXCHANGERS	6.31	2.55	0.27	0.28	Line break, loss of CCW and auxiliary saltwater.
ZCCWTK	CCW SURGE TANK	6.83	2.76	0.33	0.22	Loss of surge tank and possibly surge line low pressure leak.
ZCSPMP	CONTAINMENT SPRAY PUMPS	6.00	2.46	0.29	0.25	Line break, loss of contents.
ZSPATK	SPRAY ADDITIVE TANK	6.78	3.07	0.30	0.18	Loss of contents and one pump.
ZSDFWP	AFW PUMPS (STEAM DRIVEN)	7.71	3.38	0.29	0.21	Line break, loss of contents.
ZDGFPM	DG FUEL OIL PUMPS/FILTERS	8.33	3.65	0.27	0.23	Line break, loss of contents.
ZDSLGN	DIESEL GENERATORS	7.79	3.65	0.26	0.20	Failure to function.
ZDGRWP	DG RADIATOR/WATER PUMPS	8.78	3.66	0.29	0.24	Line break, loss of cooling, failing diesel.
ZDGEXC	DG EXCITATION CUBICAL	7.4	2.57	0.29	0.35	Failure to function.
ZDGCPN	DG CONTROL PANEL	4.66	2.24	0.30	0.13	Failure to function.

Table 3-8. DCPP Component and Structure Fragility and HCLPF Values						
RISKMAN ID	Description	Median Accl.(g)	HCLPF	Beta R	Beta U	Failure Mode and Source of Fragility Value
ZCONFC	CONTAINMENT FAN COOLERS	8.1	2.82	0.31	0.33	Failure to function, rupture of CCW piping.
ZSUPFN	CONTROL ROOM VENTILATION SUPPLY FANS	9.79	3.82	0.33	0.24	Failure to function.
ZSWGER	4 KV SWITCHGEAR (STRUCTURE)	8.47	3.20	0.31	0.28	Circuit disconnect.
ZTFRFF	BUS F POTENTIAL TRANSFORMER	10.83	3.47	0.31	0.38	Failure to function, bus F only.
ZSFGRP	4kV SAFEGUARD RELAY PANEL	10.76	3.39	0.34	0.36	Failure to function.
ZBATRY	125V DC BATTERIES	6.04	2.74	0.30	0.18	Failure to function.
ZBATCH	BATTERY CHARGERS	9.93	2.93	0.34	0.40	Failure to function.
ZSWGBP	125V DC SWITCHGEAR/BREAKER DC PANELS	6.67	2.36	0.35	0.28	Failure to function.
ZINVTR	120V AC INVERTERS	6.82	2.75	0.31	0.24	Failure to function.
ZTRANS	4KV/480V TRANSFORMERS	5.34	2.42	0.28	0.20	Failure to function.
ZARPNL	AUXILIARY RELAY PANELS	7.25	3.57	0.28	0.15	Loss of function, structural failure.
ZMCNTB	MAIN CONTROL BOARDS	7.77	2.98	0.31	0.27	Failure to function of all instrumentation readout (controls remain operable).
ZHSPNL	HOT SHUTDOWN PANEL	7.27	3.52	0.30	0.14	Failure to function.
ZPCAPS	NSSS PROCESS CONTROL AND PROTECTION SYSTEM	10.78	3.57	0.39	0.28	Failure to function, loss of indicators and control.
ZRTSWG	REACTOR TRIP SWITCHGEAR	7.9	3.14	0.30	0.26	Failure to function.
ZPADPT	PRESSURE AND DP TRANSMITTER	8.93	4.11	0.27	0.20	Loss of indicators.
ZIPLNS	IMPULSE LINES	7.09	2.63	0.28	0.32	Line break, small LOCA.
ZOSPWR	OFFSITE POWER 230 KV	1.40	.70	0.22	0.20	230kV substation function fails.
ZBOPPS	BOP ABOVE GROUND PIPING AND SUPPORTS	11.03	3.0	0.40	0.39	Line break, loss of contents.
ZHVDAS	HVAC DUCTING AND SUPPORTS	9.78	2.49	0.35	0.48	Duct joint break, loss of air, one segment per system.
ZSWGE2	SWITCHGEAR/STRUT	8.05	3.04	0.31	0.28	Circuit disconnect.
ZDCPCH	CHATTER DG CONTROL PANEL	5.0	2.63	0.26	0.14	Circuit interruption.
ZSGPCH	CHATTER 4KV SWITCHGEAR	3.53	1.31	0.35	0.25	Circuit interruption.
ZSTRUT	STRUT FOR TURBINE BUILDING	6.49	2.58	0.25	0.31	Buckling of the floor beam increasing seismic input to other components.
ZTFRF2	BUS F POTENTIAL TRANSFORMER/STRUT	5.85	1.87	0.31	0.38	Failure to function, Bus F only.
ZSFGR2	SAFEGUARD RELAY PANEL/STRUT	5.81	1.83	0.34	0.36	Failure to function.
ZCHPMP	CENTRIFUGAL CHARGING PUMPS	10.16	4.45	0.31	0.19	Line break loss of contents.
ZBLKWL	BLOCK WALLS	6.79	2.36	0.29	0.35	Loss of structural integrity

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
1	SEISMIC LEVEL 4 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	3.41E-06	10.02
2	SEISMIC LEVEL 5 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	2.02E-06	5.93
3	SEISMIC LEVEL 4 - OFFSITE POWER - ALL SIX DIESEL GENERATORS	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM	HANNI	1.09E-06	3.22

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
4	SEISMIC LEVEL 3 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	8.39E-07	2.47
5	SEISMIC LEVEL 5 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HANNS	7.80E-07	2.29

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- CONTAINMENT ISOLATION < 3 INCHES			
6	SEISMIC LEVEL 4 - OFFSITE POWER - 125V DC POWER	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNI	5.91E-07	1.74
7	SEISMIC LEVEL 5 - OFFSITE POWER - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	5.78E-07	1.70
8	SEISMIC LEVEL 5 - OFFSITE POWER - ALL SIX DIESEL GENERATORS	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HANNI	5.27E-07	1.55

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- CONTAINMENT ISOLATION < 3 INCHES			
9	SEISMIC LEVEL 6 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	5.07E-07	1.49
10	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	3.83E-07	1.13
11	SEISMIC LEVEL 4 - OFFSITE POWER - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	3.44E-07	1.01
12	SEISMIC LEVEL 3 - OFFSITE POWER - ALL SIX DIESEL GENERATORS	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS	HANNI	3.40E-07	1.00

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
13	SEISMIC LEVEL 5 - OFFSITE POWER - 125V DC POWER	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNI	3.18E-07	.93
14	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING	- OFFSITE GRID - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	2.97E-07	.87
15	SEISMIC LEVEL 1 - COMPONENT COOLING WATER SYSTEM	- CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	2.85E-07	.84

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
16	SEISMIC LEVEL 1 - OFFSITE POWER - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	2.58E-07	.76
17	SEISMIC LEVEL 4 - ALL 4KV VITAL AC POWER/STRUT SUCCESS - RCPS IN OPERATION	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - 480V SWITCHGEAR BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	2.56E-07	.75
18	SEISMIC LEVEL 5 - OFFSITE POWER - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNS	2.24E-07	.66
19	SEISMIC LEVEL 4 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - AUXILIARY FEEDWATER SYSTEM	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A	SXNNI	2.13E-07	.62

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
20	SEISMIC LEVEL 3 - OFFSITE POWER - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	2.09E-07	.61
21	SEISMIC LEVEL 5 - OFFSITE POWER - ALL SIX DIESEL GENERATORS - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNS	2.04E-07	.60
22	SEISMIC LEVEL 1 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING	- OFFSITE GRID - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	2.00E-07	.59
23	SEISMIC LEVEL 3 - OFFSITE POWER - 125V DC POWER	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12	SXNNI	1.99E-07	.58

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
24	SEISMIC LEVEL 6 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNS	1.96E-07	.58
25	SEISMIC LEVEL 3 - ALL 4KV VITAL AC POWER/STRUT SUCCESS - RCPS IN OPERATION	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - 480V SWITCHGEAR BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM	HANNI	1.67E-07	.49

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
26	SEISMIC LEVEL 1 - AUXILIARY SALT WATER SYSTEM - RCPS IN OPERATION	- COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	1.66E-07	.49
27	SEISMIC LEVEL 4 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNS	1.66E-07	.49
28	SEISMIC LEVEL 4 - OFFSITE POWER - ALL SIX DIESEL GENERATORS - SEISMIC RELAY CHATTER	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	1.59E-07	.47

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
29	SEISMIC LEVEL 1 - RCS PRESSURE RELIEF AND PORV RECLOSURE - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - MAKEUP TO RWST/HOT LEG SUCTION		INYCI	1.59E-07	.47
30	SEISMIC LEVEL 5 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - ALL SIX DIESEL GENERATORS	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	1.44E-07	.42
31	SEISMIC LEVEL 4 - OFFSITE POWER - COMPONENT COOLING WATER	- OFFSITE GRID - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	1.32E-07	.39
32	SEISMIC LEVEL 5 - OFFSITE POWER - ALL SIX DIESEL GENERATORS - SEISMIC RELAY CHATTER	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	1.26E-07	.37
33	SEISMIC LEVEL 5	- OFFSITE GRID	SXNNI	1.26E-07	.37

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
	- OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - AUXILIARY FEEDWATER SYSTEM	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
34	SEISMIC LEVEL 5 - OFFSITE POWER - 125V DC POWER - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNS	1.23E-07	.36
35	SEISMIC LEVEL 4 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - RCS PRESSURE RELIEF AND PORV RECLOSURE	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H	HXNNI	1.16E-07	.34

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
36	SEISMIC LEVEL 6 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - ALL SIX DIESEL GENERATORS	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	1.14E-07	.33
37	SEISMIC LEVEL 6 - OFFSITE POWER - ALL SIX DIESEL GENERATORS	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY	HANNI	1.11E-07	.33

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
38	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 13 - COMPONENT COOLING WATER SYSTEM	- OFFSITE GRID - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	9.91E-08	.29
39	SEISMIC LEVEL 3 - OFFSITE POWER - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	9.82E-08	.29
40	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 11 - COMPONENT COOLING WATER SYSTEM	- OFFSITE GRID - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	9.63E-08	.28
41	SEISMIC LEVEL 6 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - COMPONENT COOLING WATER	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	9.51E-08	.28
42	SEISMIC LEVEL 5 - OFFSITE POWER - 480V SWG VENTILATION	- OFFSITE GRID - 480V SWITCHGEAR VENTILATION SYSTEM - SAFETY INJECTION PUMPS - CONTROL ROOM INDICATIONS AND PLANT CONTROL	HAYCI	9.35E-08	.27

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- RHR PUMP TRAIN A - RHR PUMP TRAIN B - OPERATOR SWITCH TO CONT SUMP RECIRCULATION			
43	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 12 - COMPONENT COOLING WATER SYSTEM	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	9.34E-08	.27
44	SEISMIC LEVEL 2 - OFFSITE POWER - UNIT 2 DIESEL GENERATOR 22 - AUXILIARY SALT WATER SYSTEM	- OFFSITE GRID - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	8.76E-08	.26
45	SEISMIC LEVEL 5 - OFFSITE POWER - 125V DC POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNI	8.68E-08	.26
46	SEISMIC LEVEL 6 - OFFSITE POWER - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B	HANNI	8.56E-08	.25

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
47	SEISMIC LEVEL 4 - OFFSITE POWER - 125V DC POWER - SEISMIC RELAY CHATTER	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - TURBINE DRIVEN AFW PUMP - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNI	8.55E-08	.25
48	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 11 - AUXILIARY FEEDWATER SYSTEM	- OFFSITE GRID - SAFETY INJECTION PUMPS - OPERATOR INITIATES FEED AND BLEED COOLING - RHR PUMP TRAIN B - CONTAINMENT SUMP VALVE B	SXYAI	8.34E-08	.25
49	SEISMIC LEVEL 5 - OFFSITE POWER - COMPONENT COOLING WATER	- OFFSITE GRID - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	8.10E-08	.24
50	SEISMIC LEVEL 4 - OFFSITE POWER - ALL FOUR VITAL INSTRUMENT CHANNELS	- OFFSITE GRID - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - MANUAL SI ACTUATION - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS	SXNNS	7.93E-08	.23

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES - OPERATOR ACTION TO ISOLATE CONTAINMENT			
51	SEISMIC LEVEL 6 - OFFSITE POWER - 125V DC POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNI	7.92E-08	.23
52	SEISMIC LEVEL 6 - OFFSITE POWER - 125V DC POWER	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM	SXNNI	7.78E-08	.23

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
53	SEISMIC LEVEL 4 - OFFSITE POWER - CCW BYPASS	- OFFSITE GRID - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES - OPERATOR ACTION TO ISOLATE CONTAINMENT	HAYDS	7.75E-08	.23
54	SEISMIC LEVEL 3 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING	- OFFSITE GRID - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	7.71E-08	.23
55	SEISMIC LEVEL 3 - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	7.69E-08	.23
56	SEISMIC LEVEL 4 - OFFSITE POWER - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	7.67E-08	.23
57	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 13 - AUXILIARY SALT WATER SYSTEM	- OFFSITE GRID - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS	HAYDI	7.67E-08	.23

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION			
58	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	7.47E-08	.22
59	SEISMIC LEVEL 5 - OFFSITE POWER - 125V DC POWER - SEISMIC RELAY CHATTER	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - TURBINE DRIVEN AFW PUMP - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNI	7.25E-08	.21
60	SEISMIC LEVEL 2 - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	7.11E-08	.21
61	SEISMIC LEVEL 4 - OFFSITE POWER	- OFFSITE GRID	LNYAL	7.11E-08	.21

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
	- STEAM GENERATORS	- SEISMIC EXCESSIVE LOCA (>DBA) - AUXILIARY FEEDWATER SYSTEM - CONTAINMENT ISOLATION > 3 INCHES - OPERATOR ACTION TO ISOLATE CONTAINMENT			
62	SEISMIC LEVEL 5 - OFFSITE POWER - ALL SIX DIESEL GENERATORS - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	7.09E-08	.21
63	SEISMIC LEVEL 6 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - REFUELING WATER STORAGE TANK	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - REFUELING WATER STORAGE TANK - CENTRIFUGAL CHARGING PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANGI	6.98E-08	.21
64	SEISMIC LEVEL 5 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - RCS PRESSURE RELIEF AND PORV RECLOSURE	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22	HXNNI	6.84E-08	.20

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
65	SEISMIC LEVEL 4 - OFFSITE POWER - ALL SIX DIESEL GENERATORS - AUXILIARY FEEDWATER SYSTEM	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNI	6.82E-08	.20
66	SEISMIC LEVEL 5 - OFFSITE POWER - ALL FOUR VITAL INSTRUMENT CHANNELS	- OFFSITE GRID - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - MANUAL SI ACTUATION - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES - OPERATOR ACTION TO ISOLATE CONTAINMENT	SXNNS	6.82E-08	.20
67	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - CENTRIFUGAL CHARGING PUMPS - RHR PUMP TRAIN A - CONTAINMENT FAN COOLERS - MAKEUP TO RWST/HOT LEG SUCTION	HAYDI	6.77E-08	.20
68	SEISMIC LEVEL 1 - OFFSITE POWER - DIESEL GENERATOR 13 - COMPONENT COOLING WATER SYSTEM	- OFFSITE GRID - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A	HAYDI	6.67E-08	.20

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION			
69	SEISMIC LEVEL 5 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - PRESSURIZER RELIEF/SMALL LOCA	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - RCS PRESSURE RELIEF AND PORV RECLOSURE - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HXNNI	6.58E-08	.19
70	SEISMIC LEVEL 1 - OFFSITE POWER - DIESEL GENERATOR 11 - COMPONENT COOLING WATER SYSTEM	- OFFSITE GRID - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	6.49E-08	.19
71	SEISMIC LEVEL 1 - CENTRIFUGAL CHARGING PUMPS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - MAKEUP TO RWST/HOT LEG SUCTION		HAYCI	6.35E-08	.19
72	SEISMIC LEVEL 1 - OFFSITE POWER - DIESEL GENERATOR 12 - COMPONENT COOLING WATER SYSTEM	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	6.29E-08	.18
73	SEISMIC LEVEL 4 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING	- OFFSITE GRID - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS	HANNI	6.02E-08	.18

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
74	SEISMIC LEVEL 1 - OFFSITE POWER - UNIT 2 DIESEL GENERATOR 22 - AUXILIARY SALT WATER SYSTEM	- OFFSITE GRID - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	5.89E-08	.17
75	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - RCS PRESSURE RELIEF AND PORV RECLOSURE	- OFFSITE GRID - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT SPRAY - CONTAINMENT ISOLATION < 3 INCHES - OPERATOR ACTION TO ISOLATE CONTAINMENT	INYGS	5.89E-08	.17
76	SEISMIC LEVEL 4 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - ALL SIX DIESEL GENERATORS	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	5.86E-08	.17
77	SEISMIC LEVEL 6 - OFFSITE POWER - ALL SIX DIESEL GENERATORS - SEISMIC RELAY CHATTER	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS	HANNI	5.85E-08	.17

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
78	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 12 - DIESEL GENERATOR 11	- OFFSITE GRID - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	5.73E-08	.17
79	SEISMIC LEVEL 1 - RCS PRESSURE RELIEF AND PORV RECLOSURE - RHR PUMP TRAIN A - RHR PUMP TRAIN B		INYCI	5.65E-08	.17
80	SEISMIC LEVEL 1 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 11 - AUXILIARY FEEDWATER SYSTEM	- OFFSITE GRID - SAFETY INJECTION PUMPS - OPERATOR INITIATES FEED AND BLEED COOLING - RHR PUMP TRAIN B - CONTAINMENT SUMP VALVE B	SXYAI	5.61E-08	.16
81	SEISMIC LEVEL 5 - OFFSITE POWER - UNIT 2 DIESEL GENERATOR 21 - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	5.61E-08	.16
82	SEISMIC LEVEL 5 - OFFSITE POWER - UNIT 2 DIESEL GENERATOR 22 - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	5.60E-08	.16
83	SEISMIC LEVEL 5 - OFFSITE POWER - DIESEL GENERATOR 13 - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HANNI	5.58E-08	.16

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- CONTAINMENT ISOLATION < 3 INCHES			
84	SEISMIC LEVEL 5 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - ALL SIX DIESEL GENERATORS - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNS	5.58E-08	.16
85	SEISMIC LEVEL 5 - OFFSITE POWER - DIESEL GENERATOR 11 - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	5.42E-08	.16
86	SEISMIC LEVEL 4 - OFFSITE POWER - ALL SIX DIESEL GENERATORS - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNS	5.34E-08	.16
87	SEISMIC LEVEL 5 - OFFSITE POWER - DIESEL GENERATOR 12 - SEISMIC RELAY CHATTER	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - RCP SEAL INTEGRITY	HANNI	5.33E-08	.16

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
	- OPERATOR RESETS SEISMIC RELAY CHATTER	- CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
88	SEISMIC LEVEL 3 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - AUXILIARY FEEDWATER SYSTEM	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNI	5.23E-08	.15
89	SEISMIC LEVEL 1 - OFFSITE POWER - DIESEL GENERATOR 13 - AUXILIARY SALT WATER SYSTEM	- OFFSITE GRID - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	5.16E-08	.15
90	SEISMIC LEVEL 6 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - PRESSURIZER RELIEF/SMALL LOCA	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - RCS PRESSURE RELIEF AND PORV RECLOSURE - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B	HXNNI	5.11E-08	.15

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
91	SEISMIC LEVEL 6 - OFFSITE POWER - COMPONENT COOLING WATER	- OFFSITE GRID - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	5.07E-08	.15
92	SEISMIC LEVEL 1 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	5.03E-08	.15
93	SEISMIC LEVEL 5 - OFFSITE POWER - ALL SIX DIESEL GENERATORS - SEISMIC RELAY CHATTER - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNS	4.88E-08	.14
94	SEISMIC LEVEL 5 - OFFSITE POWER - ALL 4KV VITAL AC POWER/STRUT SUCCESS - AUXILIARY FEEDWATER SYSTEM - OPERATOR ACTION TO ISOLATE CONTAINMENT	- OFFSITE GRID - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM	SXNNS	4.86E-08	.14

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- CONTROL ROOM HVAC SYSTEM - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
95	SEISMIC LEVEL 4 - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- RCP SEAL INTEGRITY - CONTROL ROOM INDICATIONS AND PLANT CONTROL - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - OPERATOR SWITCH TO CONT SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	HANNI	4.77E-08	.14
96	SEISMIC LEVEL 5 - OFFSITE POWER - 125V DC POWER - SEISMIC RELAY CHATTER - OPERATOR RESETS SEISMIC RELAY CHATTER	- OFFSITE GRID - 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - UNIT 2 DIESEL GENERATOR 21 - DIESEL FUEL OIL TRANSFER SYSTEM - 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - CONTROL ROOM HVAC SYSTEM - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - TURBINE DRIVEN AFW PUMP - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES	SXNNI	4.63E-08	.14
97	SEISMIC LEVEL 4 - 125V DC POWER	- 125V DC POWER BUS F - 125V DC POWER BUS G - 125V DC POWER BUS H - VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - VITAL AC 4KV BUS H - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - 480V SWITCHGEAR BUS H - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - DIESEL GENERATOR 11 - DIESEL GENERATOR UNIT 1/2 COUPLING	SXNNI	4.63E-08	.14

Table 3-9. Top 100 Seismic Core Damage Sequences

Rank No.	Events: Sequence Description	Events: Guaranteed Events/Comments	End State	Frequency (per year)	Percent
		- 120V VITAL INSTRUMENT AC CHANNEL I - 120V VITAL INSTRUMENT AC CHANNEL II - 120V VITAL INSTRUMENT AC CHANNEL III - 120V VITAL INSTRUMENT AC CHANNEL IV - SSPS TRAIN A - SSPS TRAIN B - MANUAL SI ACTUATION - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - CONTAINMENT SPRAY - WATER LEVEL FOR SUMP RECIRCULATION - CONTAINMENT ISOLATION < 3 INCHES			
98	SEISMIC LEVEL 5 - OFFSITE POWER - COMPONENT COOLING WATER - RCP SEAL INTEGRITY	- OFFSITE GRID - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	4.62E-08	.14
99	SEISMIC LEVEL 1 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	- OFFSITE GRID - CONTROL ROOM HVAC SYSTEM - CENTRIFUGAL CHARGING PUMPS - RHR PUMP TRAIN A - CONTAINMENT FAN COOLERS - MAKEUP TO RWST/HOT LEG SUCTION	HAYDI	4.56E-08	.13
100	SEISMIC LEVEL 2 - OFFSITE POWER - DIESEL GENERATOR 13 - DIESEL GENERATOR 12	- OFFSITE GRID - UNIT 2 DIESEL GENERATOR 23 - UNIT 2 DIESEL GENERATOR 22 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	HAYDI	4.52E-08	.13

Table 3-10. Seismic Top Event Importance (Sorted by Fussel-Vesely)				
Rank	Top Event	Fraction Importance	Fussel-Vesely Importance	Top Event Description
1.	SOP	9.1476E-01	4.8219E-01	Seismic Offsite Power
2.	SACSS	3.4770E-01	2.8912E-01	Seismic Loss of All 4 Kv Vital Power / Strut Success
3.	GG	1.5635E-01	1.3262E-01	Diesel Generator 12
4.	SDG	1.4635E-01	1.0667E-01	Seismic All Six Diesel Generators
5.	OC	1.1595E-01	9.9938E-02	Operator Resets Seismic Relay Chatter
6.	GF	1.2193E-01	9.5265E-02	Diesel Generator 13
7.	SCT	1.6364E-01	9.0454E-02	Seismic Relay Chatter
8.	AS	9.1164E-02	8.9123E-02	Auxiliary Saltwater System
9.	GH	1.0132E-01	7.0045E-02	Diesel Generator 11
10.	SDC	8.5749E-02	5.9738E-02	Seismic 125V DC Power
11.	CC	6.0623E-02	5.9232E-02	Component Cooling Water System
12.	SCC	4.3517E-02	1.9600E-02	Seismic Component Cooling Water
13.	RF	2.4632E-02	1.8270E-02	Operator Switch to Cont. Sump Recirculation
14.	PR	3.4251E-02	1.3096E-02	RCS Pressure Relief and PORV Reclosure
15.	CH	1.3627E-02	1.3085E-02	Centrifugal Charging Pumps

Table 3-11. Seismic Component Importance (Sorted by Fussel-Vesely)

Component ID	Top Event	Fractional Importance	Fussel-Vesely Importance	Component Description
ZOSPRW	SOP	9.1476E-01	4.8216E-01	OFFSITE POWER, 230 KV
ZTBSHR	SACSS	2.2139E-01	1.8136E-01	TURBINE BUILDING SHEAR WALL
ZSGPCH	SCT	1.5755E-01	8.6632E-02	CHATTER, 4KV SWITCHGEAR
ZDGCPN	SDG	1.1645E-01	8.3770E-02	DG CONTROL PANEL
ZSWGBP	SDC	5.6108E-02	4.2539E-02	DC SWITCHGEAR/BREAKER PANEL
ZTRANS	SACSS	5.2386E-02	3.6936E-02	4KV/480V TRANSFORMERS
ZDGEXC	SDG	2.9992E-02	2.1489E-02	DG EXCITATION CUBICAL
ZBLKWL	SACSS	3.1103E-02	1.9533E-02	BLOCK WALLS
ZBLKWL	SACSS	3.1103E-02	1.9533E-02	BLOCK WALLS
ZCCWHX	SCC	2.4810E-02	1.5086E-02	CCW HEAT EXCHANGERS
ZBATRY	SDC	1.9555E-02	1.1936E-02	BATTERIES
ZSTMGN	SSG	1.9268E-02	1.1719E-02	STEAM GENERATOR
ZBATCH	SACSS	1.5505E-02	1.1310E-02	BATTERY CHARGERS
ZINVTR	SVI	1.2907E-02	6.6393E-02	INVERTERS
ZCONFC	SCB	1.0862E-02	5.7987E-03	CONTAINMENT FAN COOLERS

Table 3-12. Seismic Basic Event Importance (Sorted by Fussel-Vesely)				
Rank	Basic Event	Fractional Importance	Fussel-Vesely Importance	Basic Event Description
1.	CCOPC	3.5948E-02	3.5497E-02	Failure to Reduce CCW Heat Load with one CCW Pump
2.	ASOP2C	2.4483E-02	2.3700E-02	Failure to Locally Cross-Tie U1 and U2
3.	RFBKB2	1.9526E-02	1.9444E-02	Failure to Switch to Cont. Sump Recirculation
4.	PR-ZTR1SA	1.1089E-04	1.0696E-02	Pressurizer Safety Valve 8010A Fails to Reclose
5.	PR-ZTR1SB	1.1089E-04	1.0696E-02	Pressurizer Safety Valve 8010B Fails to Reclose
6.	PR-ZTR1SC	1.1089E-04	1.0696E-02	Pressurizer Safety Valve 8018C Fails to Reclose
7.	[ASBKCL]	9.3924E-03	9.2430E-03	Failure of Auxiliary Saltwater Pump 12 to Start
8.	CHBKE	9.7181E-04	8.3737E-03	Failure of Centrifugal Charging Pump 12 or Flow Path
9.	MUHEB	8.2653E-03	8.2349E-03	Operator Failure to Provide Makeup to RWST
10.	CHBKB	8.2715E-04	7.7863E-03	Failure of RWST Supply Valve to Charging Pump

Table 3-13. Highest Frequency Plant Damage States									
Highest Frequency Plant Damage States									
Rank	**PDS Identifier**	**RCS Pressure**	**Steam Generator Cooling**	**RWST Injected**	**Containment Spray**	**Containment Heat Removal**	**Containment Integrity**	**Annual Frequency**	**Percentage of Seismic CDF**
1	HANNI	High	Yes	No	No	No	Yes	1.66E-5	41.8
2	HAYDI	High	Yes	Yes	Yes	No	Yes	7.31E-6	18.4
3	SXNNI	Setpoint	No	No	No	No	Yes	3.79E-6	9.5
4	HANNS	High	Yes	No	No	No	No*	3.52E-6	8.8
5	SXNNS	Setpoint	No	No	No	No	No*	1.35E-6	3.4
6	HXNNI	High	No	No	No	No	Yes	8.24E-7	2.1
7	INYCI	Intermediate	N/A	Yes	Yes	Yes	Yes	7.25E-7	1.8
8	HAYCI	High	Yes	Yes	Yes	Yes	Yes	7.13E-7	1.8
9	LNNNL	Low	N/A	No	No	No	No	5.61E-7	1.4
10	LNYAL	Low	N/A	Yes	Yes	Yes	No	4.57E-7	1.1

* (<3-inch leak) Note: Frequencies are presented in exponential notation, i.e., 1e-1 = 1 X 10^{-1}.

Figure 3-1. Fragility Curve Representation

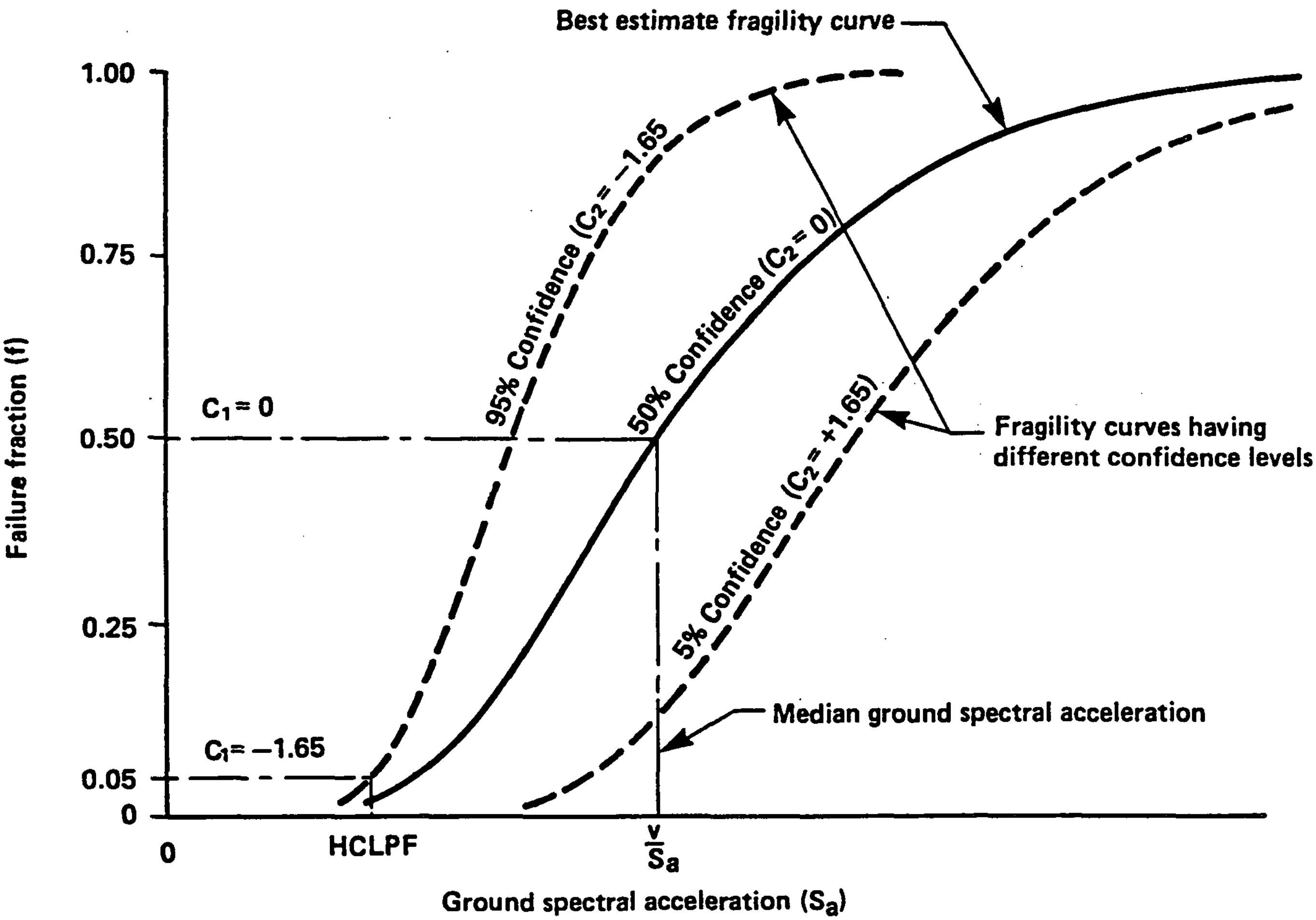


Figure 3-2. Plant Logic Analysis Process

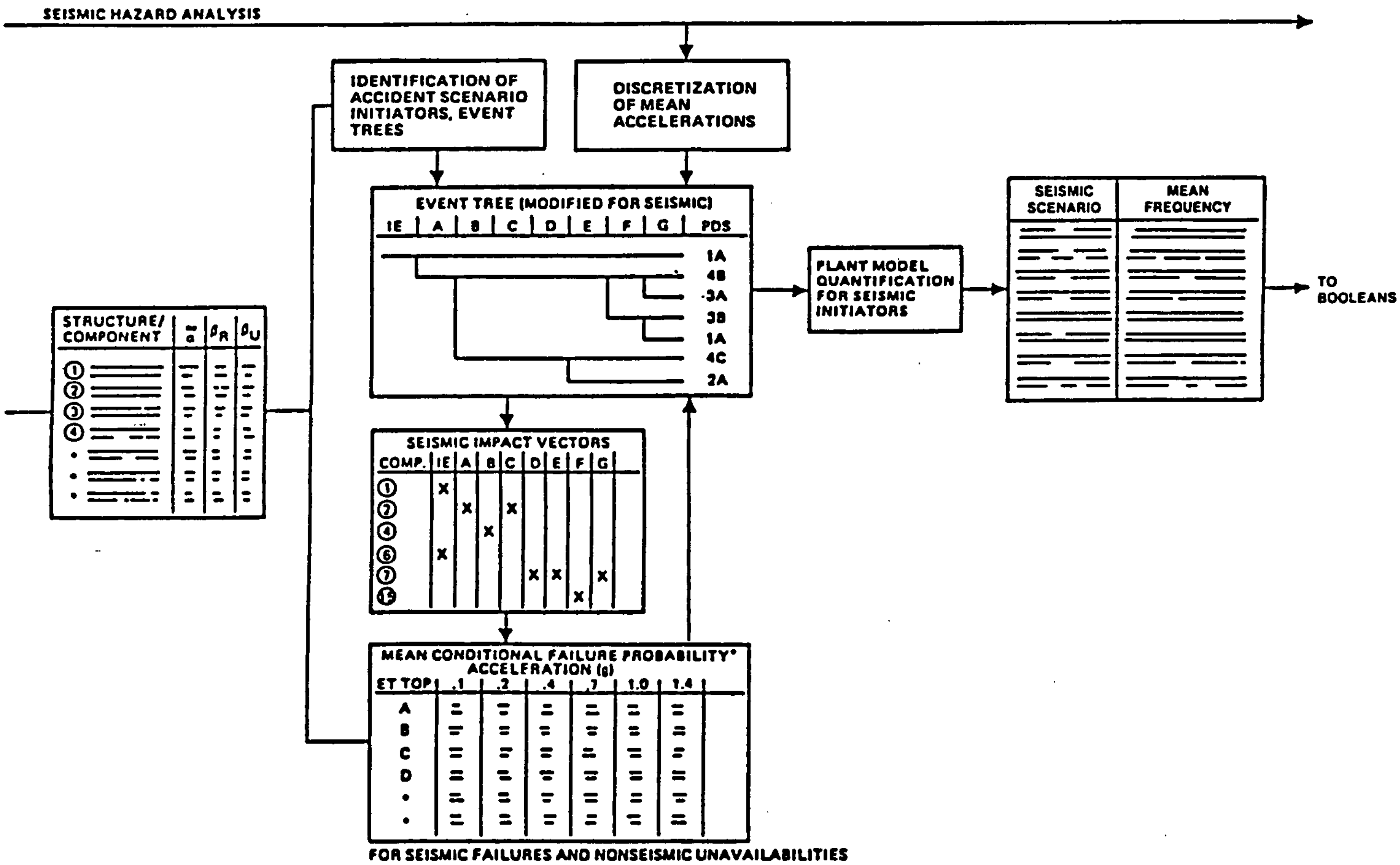


Figure 3-3. DCPP Seismic Hazard Curves

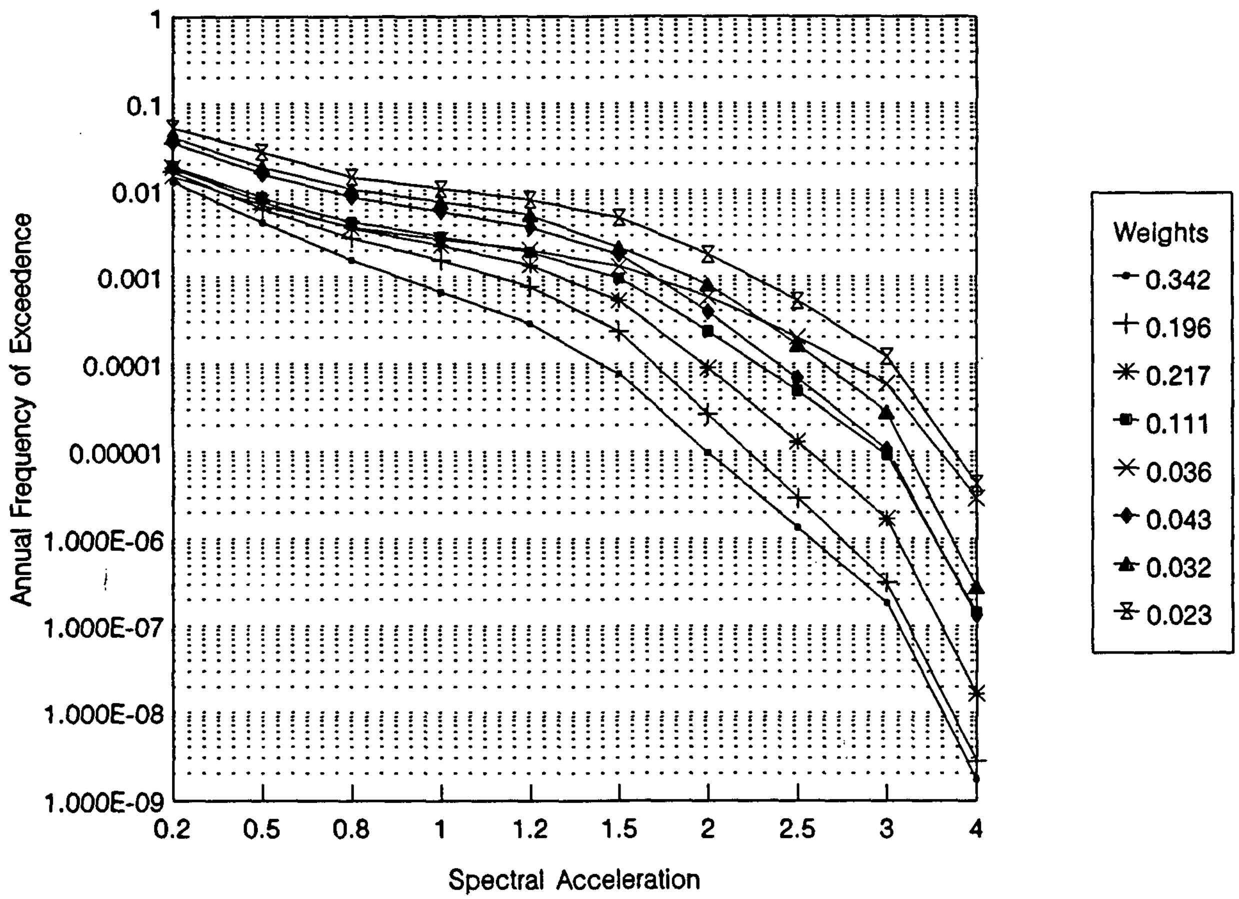


Figure 3-4. Seismic Hazard Mean Frequencies

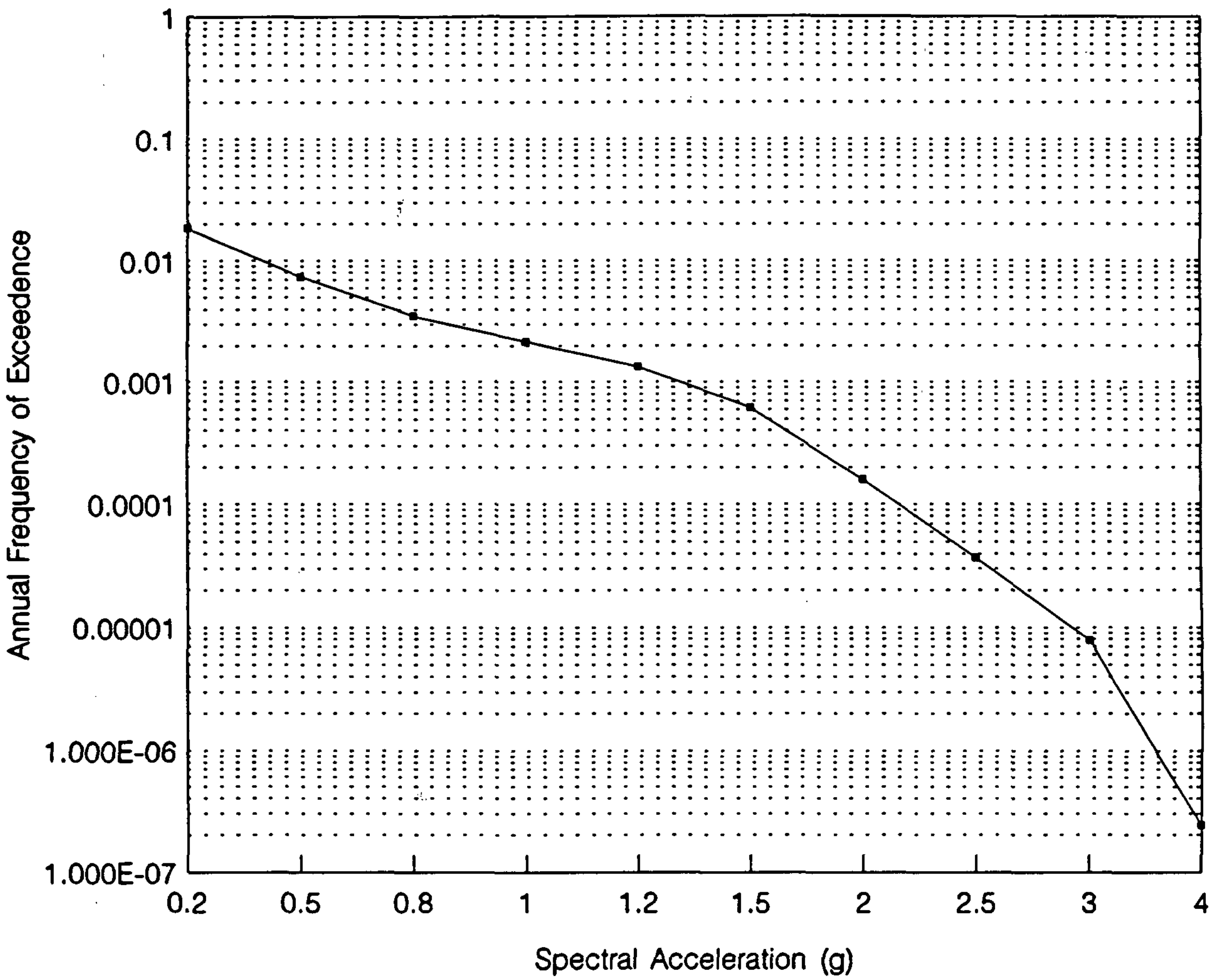


Figure 3-5. SEISPRE Event Tree

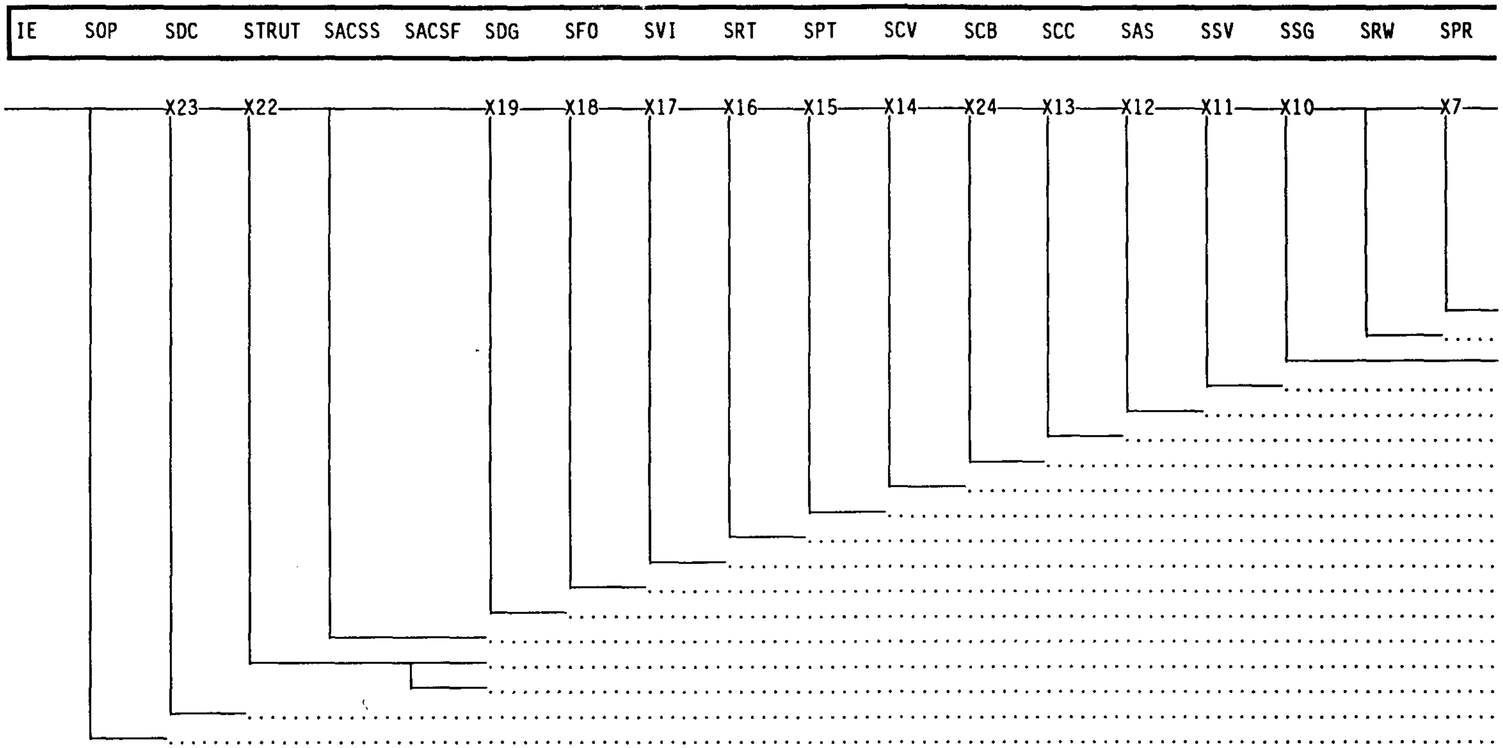
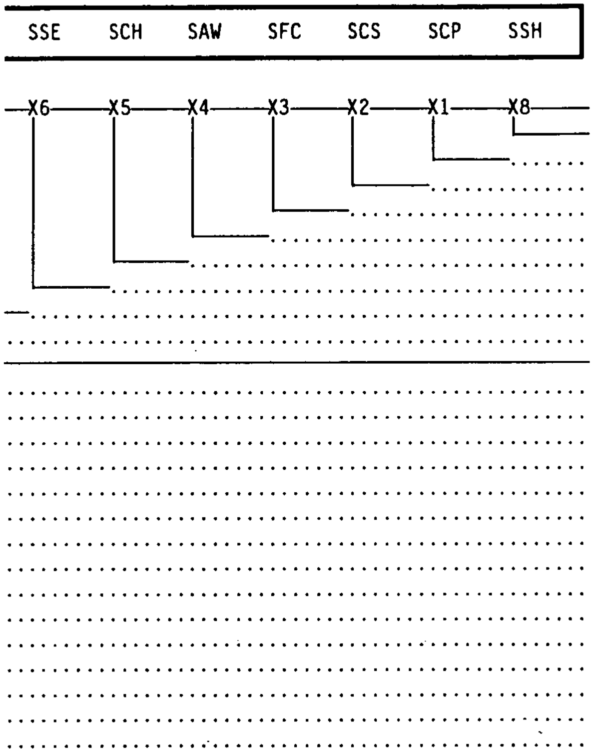


Figure 3-5. SEISPRE Event Tree



1		1
2		2
3	X8	3-4
4	X1	5-8
5	X2	9-16
6	X3	17-32
7	X4	33-64
8	X5	65-128
9	X6	129-256
10	X7	257-512
11		513
12	X10	514-1026
13	X11	1027-2052
14	X12	2053-4104
15	X13	4105-8208
16	X24	8209-16416
17	X14	16417-32832
18	X15	32833-65664
19	X16	65665-131328
20	X17	131329-262656
21	X18	262657-525312
22	X19	525313-1050624
23	X19	1050625-1575936
24	X19	1575937-2101248
25	X22	2101249-4202496
26	X23	4202497-8404992

Figure 3-5. SEISPRE Event Tree

Top Event Legend for Tree: SEISPRE

Top Event Designator.....	Top Event Description............................
IE	Initiating Event
SOP	OFFSITE POWER
SDC	125V DC POWER
STRUT	TURBINE BUILDING STRUT
SACSS	ALL 4KV VITAL AC POWER/STRUT SUCCESS
SACSF	ALL 4KV VITAL AC POWER/STRUT FAILURE
SDG	ALL SIX DIESEL GENERATORS
SFO	FUEL OIL TRANSFER
SVI	ALL FOUR VITAL INSTRUMENT CHANNELS
SRT	REACTOR TRIP
SPT	PARTIAL REACTOR TRIP
SCV	CONTROL ROOM VENTILATION
SCB	CCW BYPASS
SCC	COMPONENT COOLING WATER
SAS	AUXILIARY SALTWATER
SSV	480V SWG VENTILATION
SSG	STEAM GENERATORS
SRW	REFUELING WATER STORAGE TANK
SPR	PRESSURIZER RELIEF/SMALL LOCA
SSE	RCP SEAL COOLING
SCH	CENTRIFUGAL CHARGING PUMP
SAW	AUXILIARY FEEDWATER
SFC	CONTAINMENT FAN COOLER UNITS
SCS	PARTIAL CONTAINMENT SPRAY
SCP	LARGE CONTAINMENT FAILURE
SSH	CONTAINMENT SPRAY HEADER FAILURE

Figure 3-6. ELECPWR Event Tree

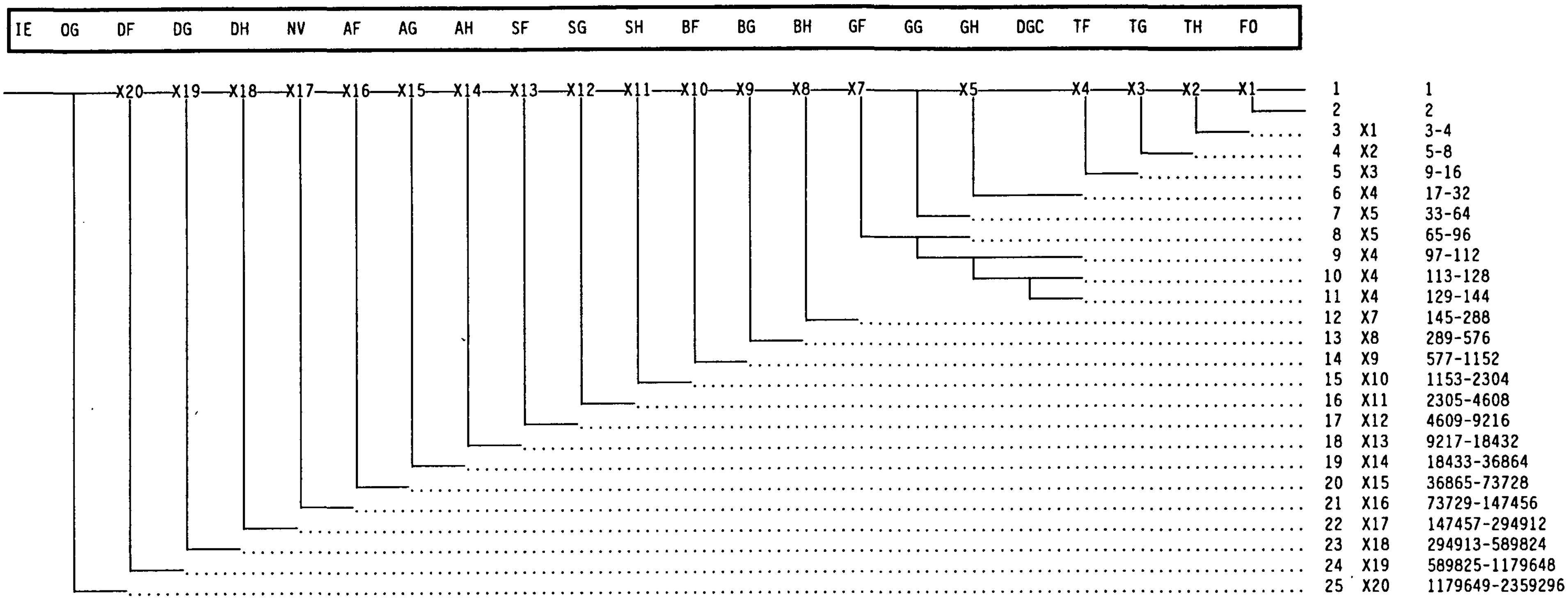
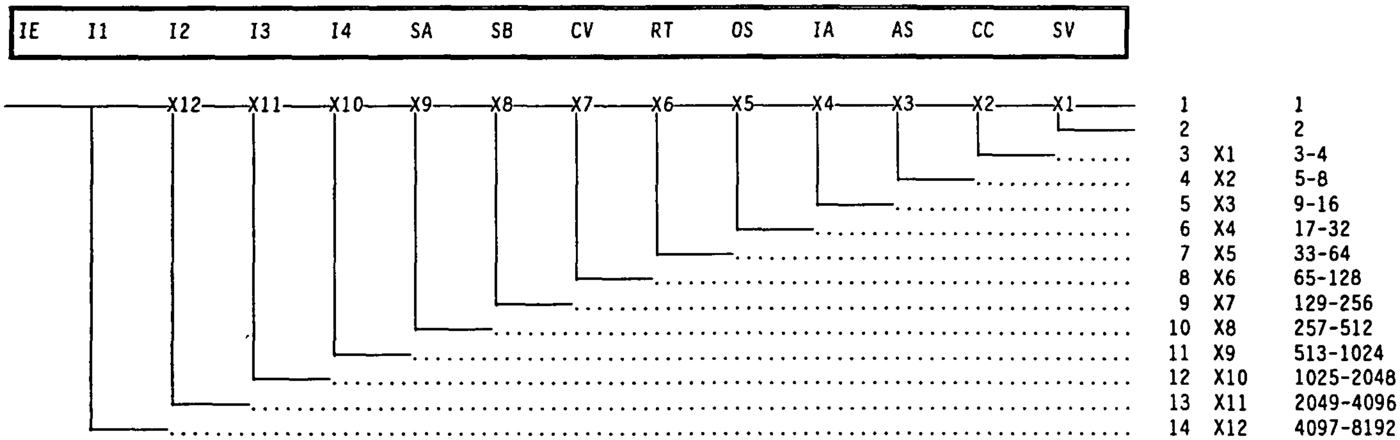


Figure 3-6. ELECPWR Event Tree

Top Event Legend for Tree: ELECPWR

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
OG	OFFSITE GRID
DF	125V DC POWER BUS F
DG	125V DC POWER BUS G
DH	125V DC POWER BUS H
NV	NONVITAL 4KV POWER
AF	VITAL AC 4KV BUS F
AG	VITAL AC 4KV BUS G
AH	VITAL AC 4KV BUS H
SF	480V SWITCHGEAR BUS F
SG	480V SWITCHGEAR BUS G
SH	480V SWITCHGEAR BUS H
BF	UNIT 2 125V DC 21, 480V 2F & 4KV HF
BG	UNIT 2 125V DC 22, 480V 2G & 4KV HG
BH	UNIT 2 125V DC 23, 480V 2H & 4KV HH
GF	DIESEL GENERATOR 13
GG	DIESEL GENERATOR 12
GH	DIESEL GENERATOR 11
DGC	DIESEL GENERATOR UNIT 1/2 COUPLING
TF	UNIT 2 DIESEL GENERATOR 23
TG	UNIT 2 DIESEL GENERATOR 22
TH	UNIT 2 DIESEL GENERATOR 21
FO	DIESEL FUEL OIL TRANSFER SYSTEM

Figure 3-7. MECHSP Event Tree



Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
I1	120V VITAL INSTRUMENT AC CHANNEL I
I2	120V VITAL INSTRUMENT AC CHANNEL II
I3	120V VITAL INSTRUMENT AC CHANNEL III
I4	120V VITAL INSTRUMENT AC CHANNEL IV
SA	SSPS TRAIN A
SB	SSPS TRAIN B
CV	CONTROL ROOM HVAC SYSTEM
RT	REACTOR PROTECTION SYSTEM
OS	MANUAL SI ACTUATION
IA	INSTRUMENT AIR
AS	AUXILIARY SALT WATER SYSTEM
CC	COMPONENT COOLING WATER SYSTEM
SV	480V SWITCHGEAR VENTILATION SYSTEM

Figure 3-8. GENTRN Event Tree

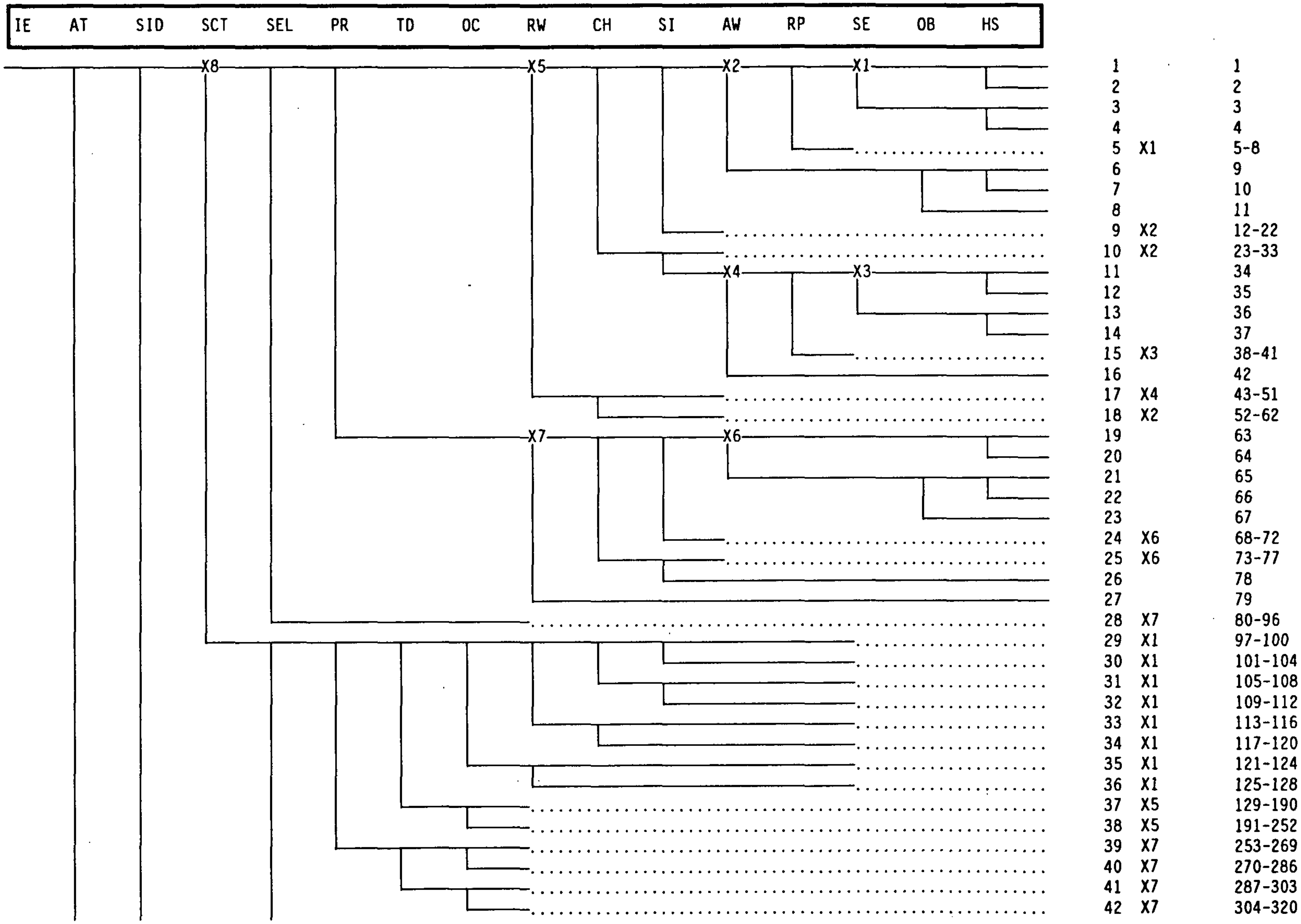


Figure 3-8. GENTRN Event Tree

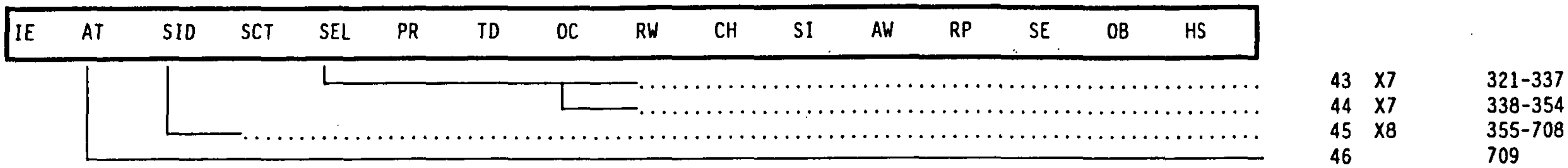


Figure 3-8. GENTRN Event Tree

Top Event Legend for Tree: GENTRN

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
AT	REACTOR TRIP SUCCESSFUL
SID	SEISMIC INDICATIONS TO OPERATOR
SCT	SEISMIC RELAY CHATTER
SEL	SEISMIC EXCESSIVE LOCA (>DBA)
PR	RCS PRESSURE RELIEF AND PORV RECLOSURE
TD	TURBINE DRIVEN AFW PUMP
OC	OPERATOR RESETS SEISMIC RELAY CHATTER
RW	REFUELING WATER STORAGE TANK
CH	CENTRIFUGAL CHARGING PUMPS
SI	SAFETY INJECTION PUMPS
AW	AUXILIARY FEEDWATER SYSTEM
RP	RCPS IN OPERATION
SE	RCP SEAL INTEGRITY
OB	OPERATOR INITIATES FEED AND BLEED COOLING
HS	CONTROL ROOM INDICATIONS AND PLANT CONTROL

Figure 3-9. LTREE Event Tree

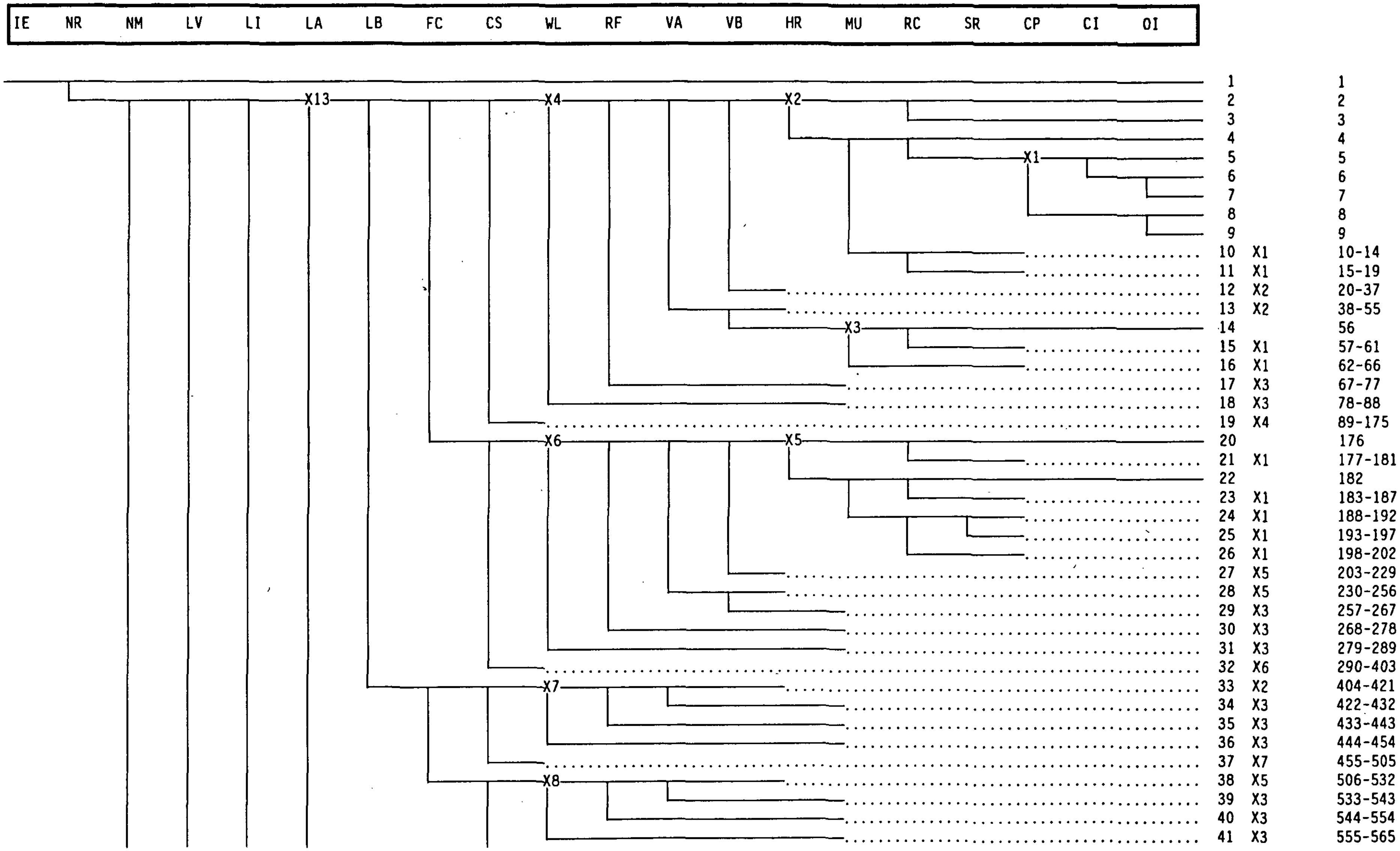


Figure 3-9. LTREE Event Tree

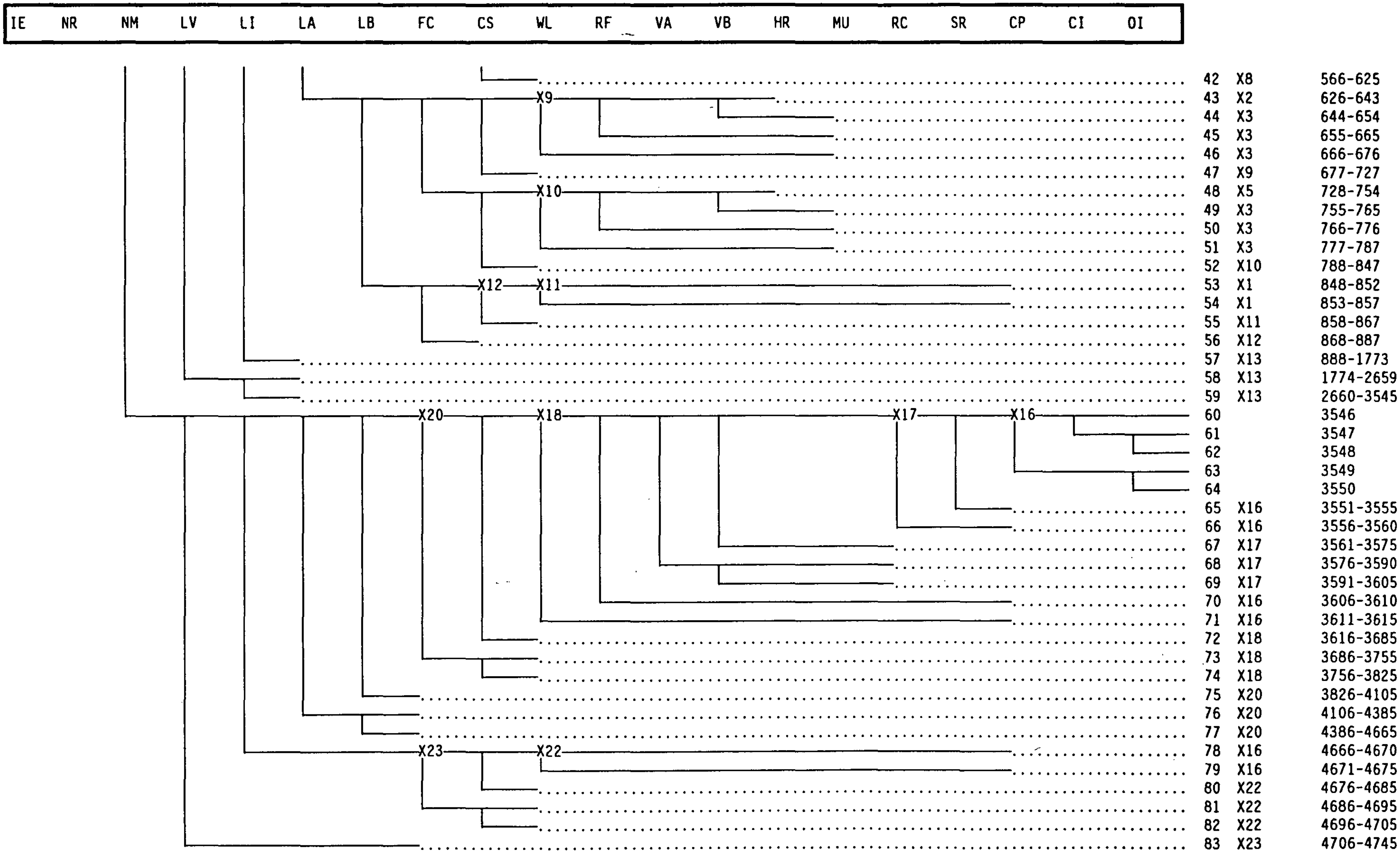


Figure 3-9. LTREE Event Tree

Top Event Legend for Tree: LTREE

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
NR	NO RECIRCULATION
NM	NO CORE DAMAGE
LV	RHR SUCTION FROM RWST
LI	COLD LEG INJECTION LINES
LA	RHR PUMP TRAIN A
LB	RHR PUMP TRAIN B
FC	CONTAINMENT FAN COOLERS
CS	CONTAINMENT SPRAY
WL	WATER LEVEL FOR SUMP RECIRCULATION
RF	OPERATOR SWITCH TO CONT SUMP RECIRCULATION
VA	CONTAINMENT SUMP VALVE A
VB	CONTAINMENT SUMP VALVE B
HR	HIGH PRESSURE RECIRCULATION
MU	MAKEUP TO RWST/HOT LEG SUCTION
RC	CCW COOLING TO RHR HEAT EXCHANGERS
SR	CONTAINMENT SPRAY RECIRCULATION
CP	CONTAINMENT ISOLATION > 3 INCHES
CI	CONTAINMENT ISOLATION < 3 INCHES
OI	OPERATOR ACTION TO ISOLATE CONTAINMENT

Figure 3-10. Core Damage Frequency by Initiating Event

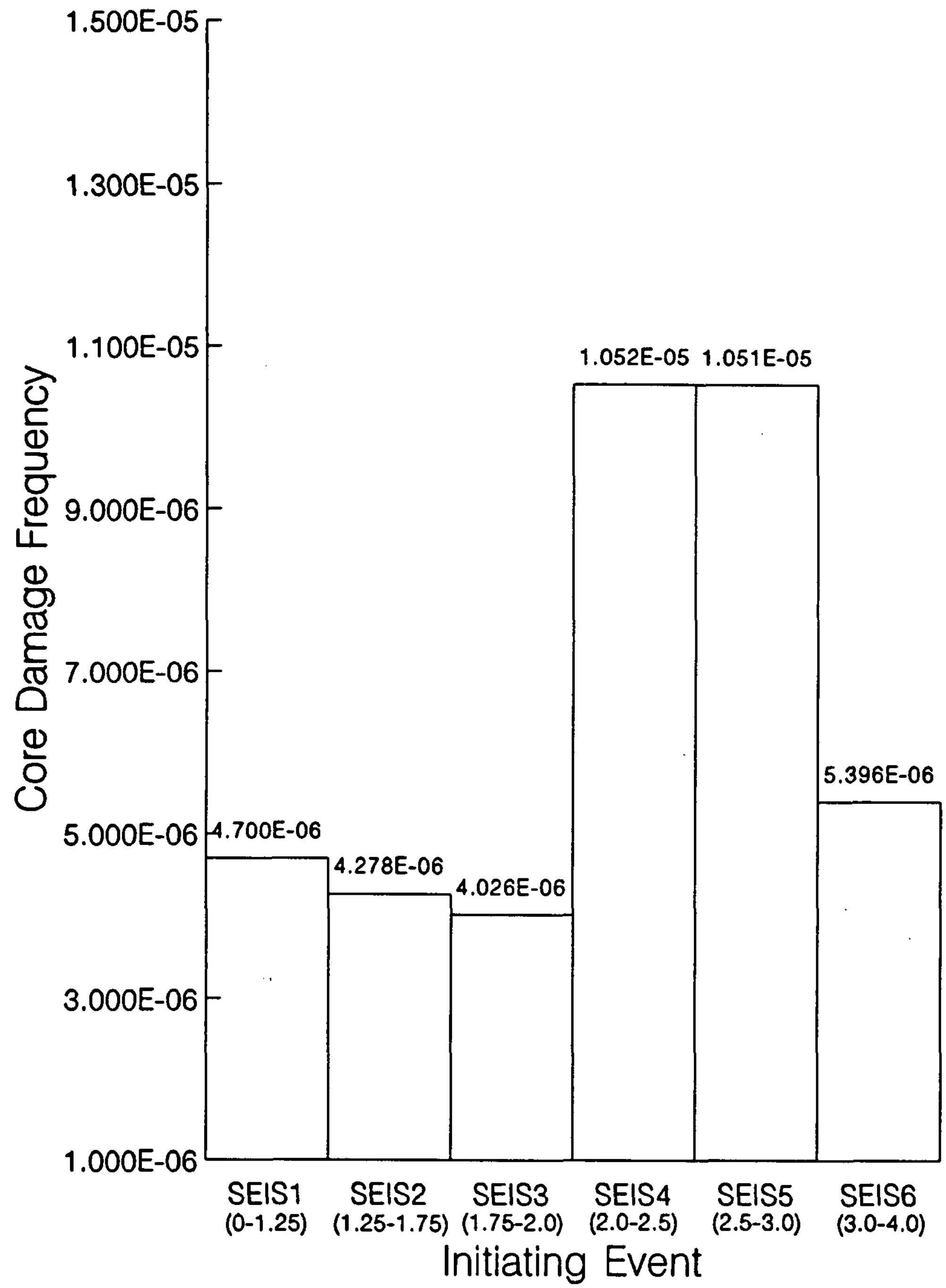


Figure 3-11. Plant Fragility Including Seismic Failure and Random Failure Modes for Seismic Initiators

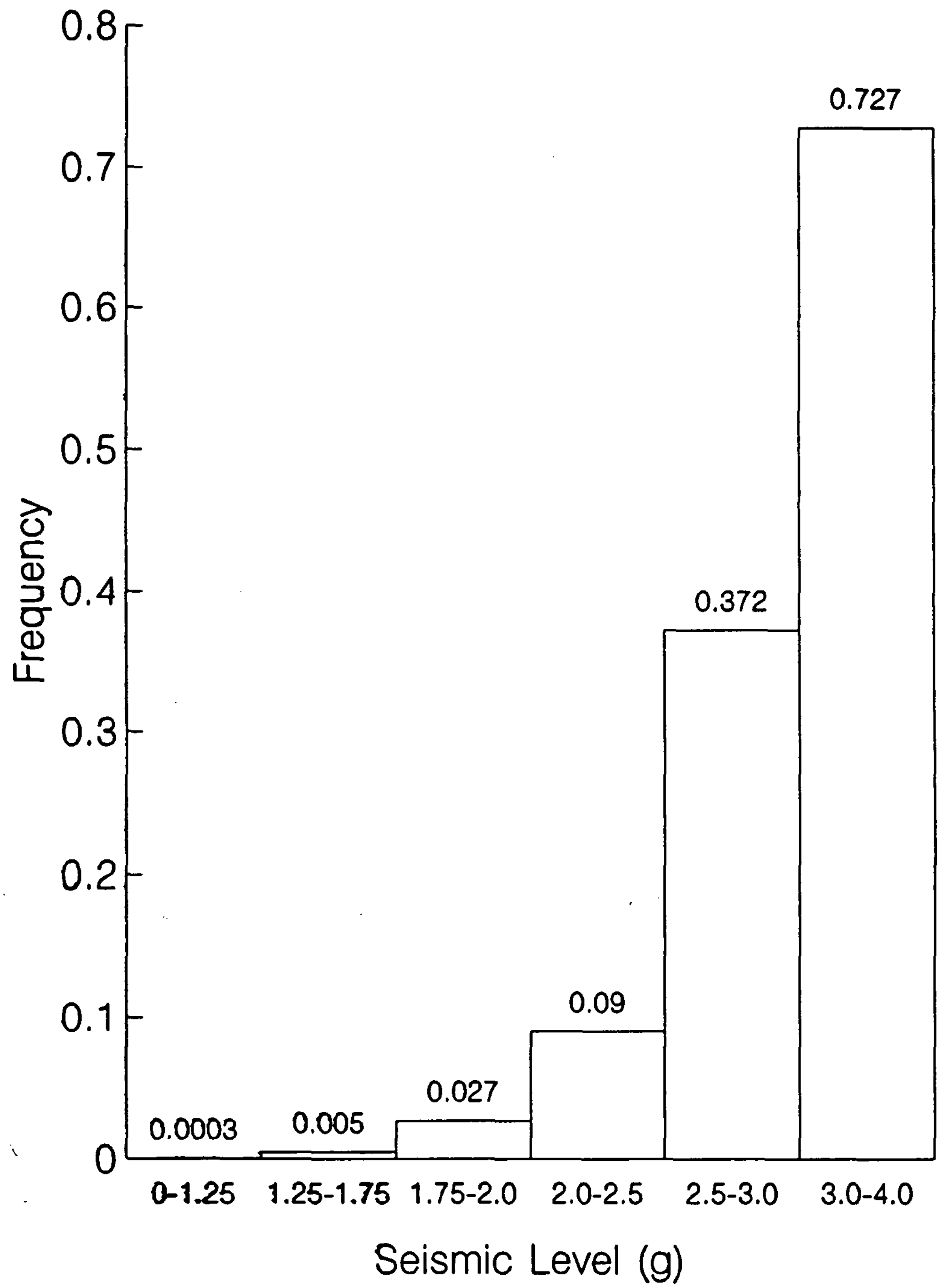


Figure 3-12. Highest Frequency Sequence Fragility Including Seismic and Random Failure Modes for Seismic Initiators

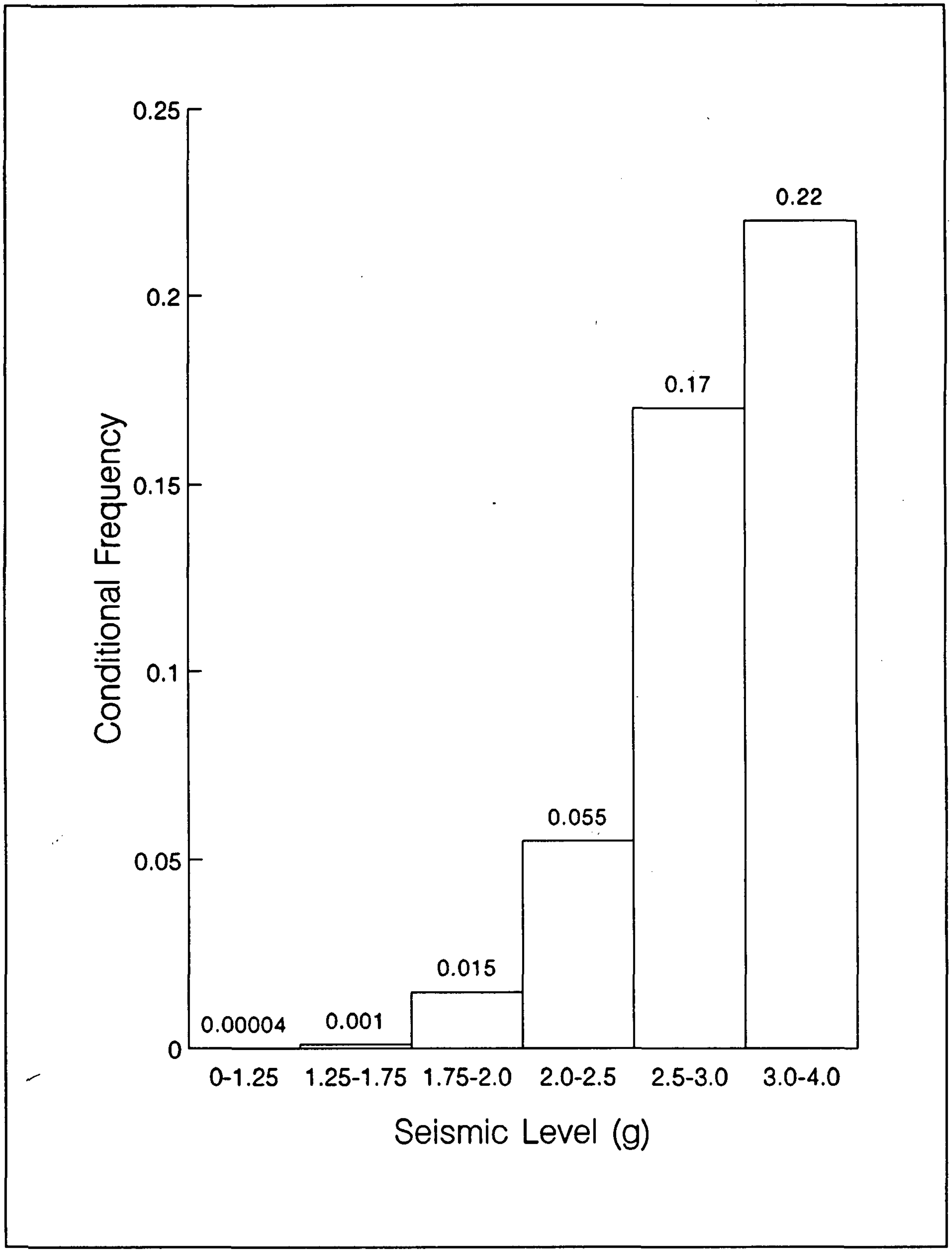


Figure 3-13. Seismic Uncertainty

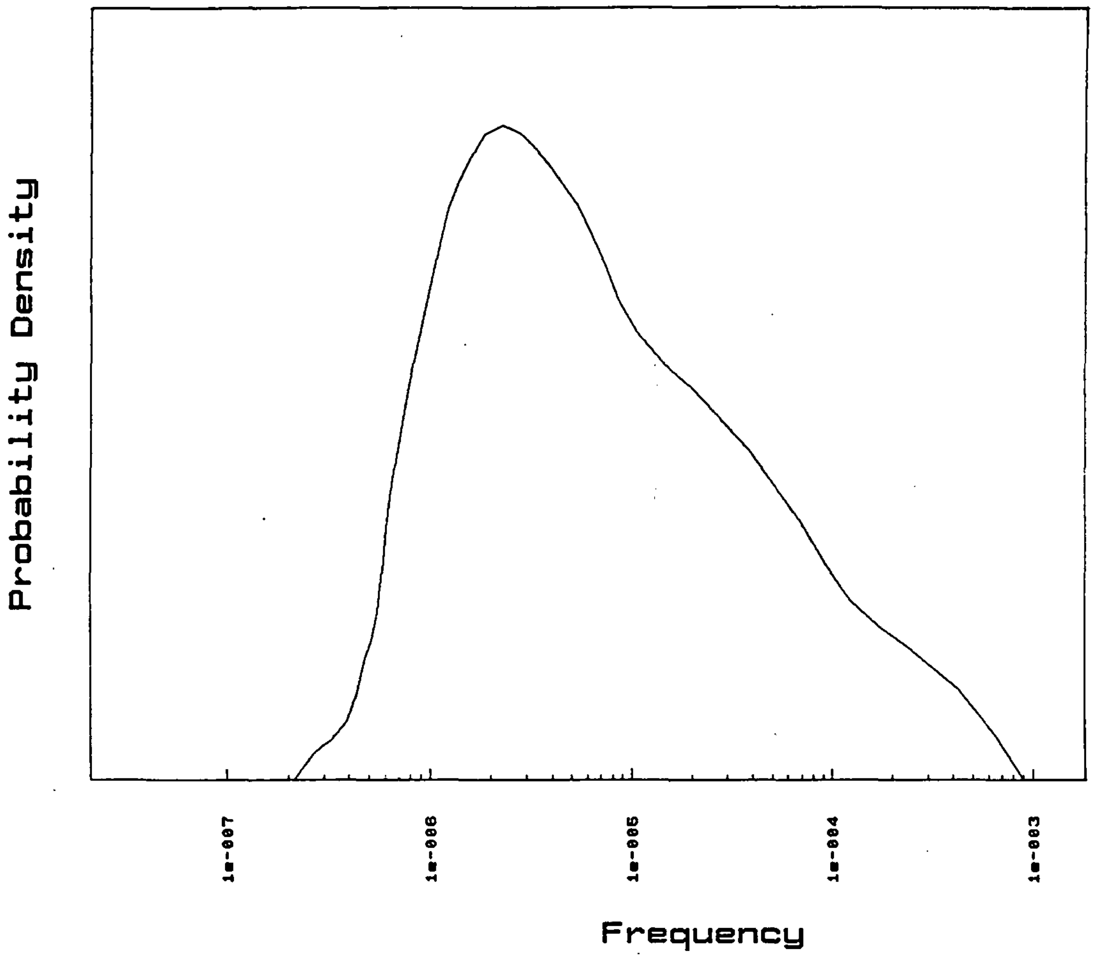


Figure 3-14. Containment Fragility for Large Opening Failures Including Seismic and Random Failure Modes for Seismic Initiators

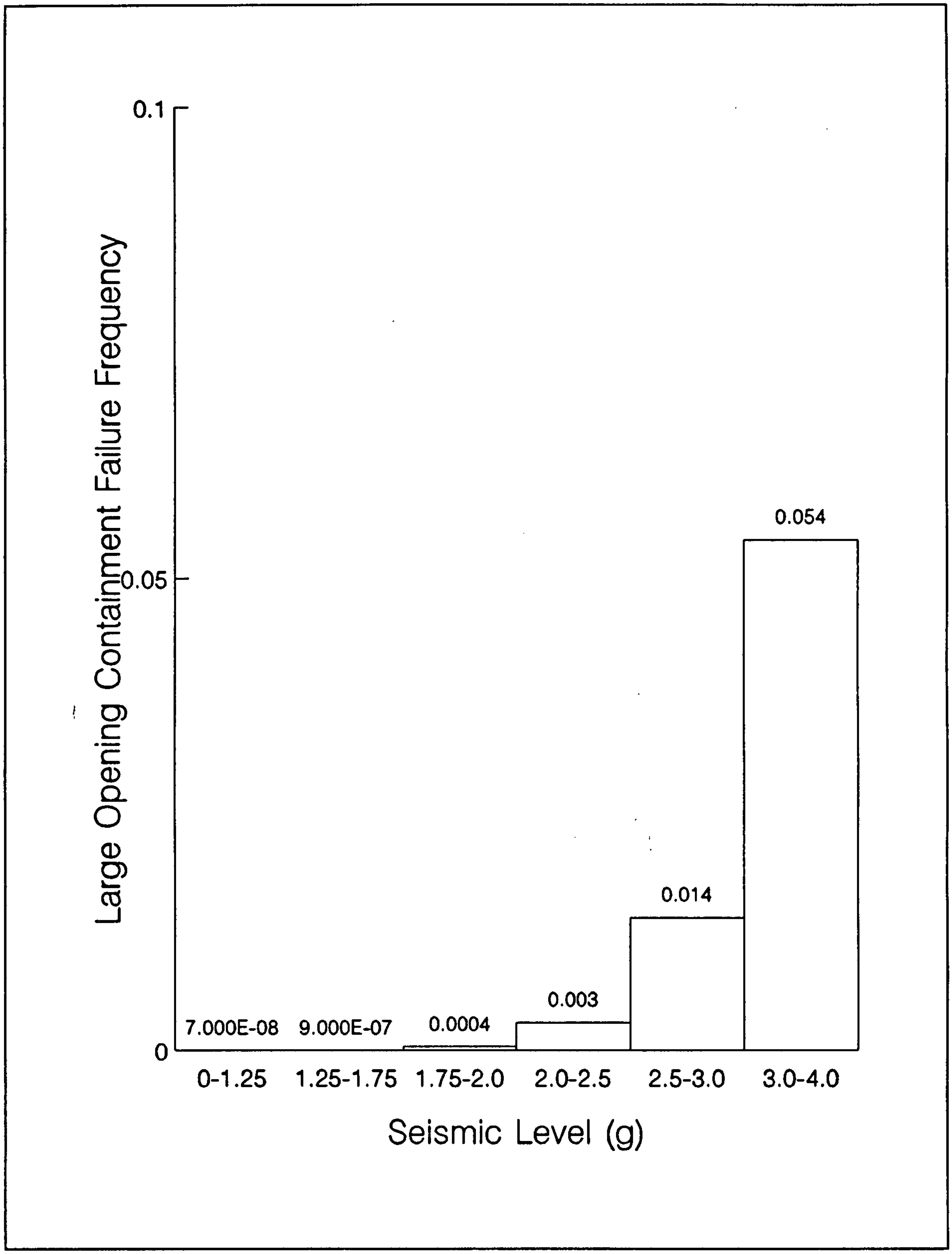
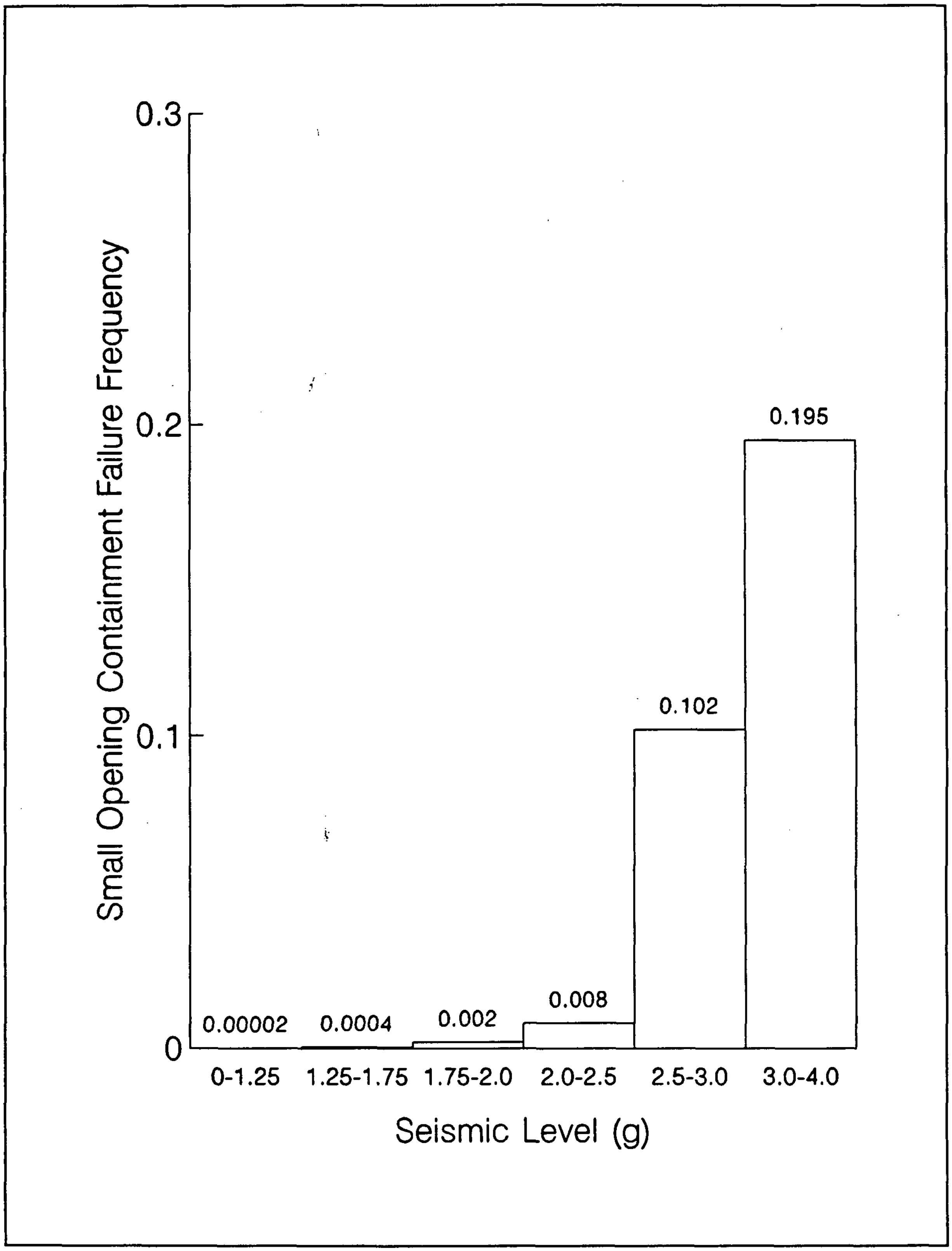


Figure 3-15. Containment Fragility for Small Opening Failures Including Seismic and Random Failure Modes for Seismic Initiators



4. INTERNAL FIRE ANALYSIS

4.0 METHODOLOGY SELECTION

4.0.1 USE OF EXISTING PRA METHODOLOGY

NUREG-1407 (Reference 4-1) defines the use of an existing Fire PRA as an acceptable methodology for performing the internal fires IPEEE as follows:

> "The use of an existing fire PRA for the internal fires IPEEE is acceptable provided the PRA reflects the current as-built and as-operated status of the plant and the licensee addresses the deficiencies of past PRAs that are identified in the Fire Risk Scoping Study." (Reference 4-2)

Issues related to configuration management are discussed in Section 4.2.2 of this report. The Fire Risk Scoping Study issues are addressed in Section 4.8 of this report.

The overall methodology of constructing the PRA model, that is, of describing the accident scenarios caused by various initiating events and rendering them amenable for quantification, is the "large event tree, small fault tree" approach. The theoretical and mathematical bases for the approach is given in the PLG methodology document (Reference 4-3). This methodology was employed in the original Diablo Canyon Probabilistic Risk Assessment (DCPRA-1988) (Reference 4-4) performed as part of the Long Term Seismic Program (LTSP) (Reference 4-5). The methodology was also updated and used for the Individual Plant Examination Report (IPE) (Reference 4-6). The DCPRA-1988 is a full-scope Level 1 PRA that evaluated the probable frequency of experiencing reactor and plant damage as the result of both internal and external initiating events. While it was performed only for Diablo Canyon Power Plant (DCPP) Unit 1, the DCPRA-1988 is equally applicable to Unit 2 because of the substantial similarities between the two units.

The Nuclear Regulatory Commission (NRC) reviewed the LTSP and issued Supplemental Safety Evaluation Report No. 34 for NUREG-0675 (SSER-34) (Reference 4-7) in June 1991, accepting the DCPRA-1988. The DCPRA-1988 was reviewed for the NRC primarily by Brookhaven National Laboratory (BNL). BNL concentrated its review on internal events and overall risk integration. The fire analysis was reviewed primarily by the NRC Staff. Among the conclusions of SSER-34 were the following:

> "The scope of the review was limited to a review of the methodology and the dominant fire core damage scenarios as reported in the PRA. The limited staff review of the fire events finds that the methodology used is acceptable and incorporates a very exhaustive spatial interaction analysis to develop scenarios. The methodology used in the fire portion of the PRA is acceptable, and judged capable of finding fire vulnerabilities in the plant, as well as ranking them in the order of importance."

In addition, the Advisory Committee on Reactor Safeguards accepted the NRC's review of the LTSP and DCPRA-1988 and concluded that

> "DCPP can be operated without undue risk to the health and safety of the public" (Reference 4-8).

In response to NRC Staff comments, documented in SSER-34, as well as the availability of more contemporary fire events data, the fire ignition frequency methodology has been reevaluated for the IPEEE 1993 Fire PRA. A study was completed to reassess the fire ignition frequencies in light of the availability of more contemporary fire events data. The fire ignition frequencies have been updated where appropriate. Additionally, the methodology for allocating fire ignition frequency in the turbine building has been enhanced in response to the above NRC Staff comments. The combination of the new database with the enhanced methodology accurately reflects the current state of knowledge on the likelihood of turbine operating deck fires in pressurized water reactors (PWRs). Details of the fire ignition frequency study are summarized in Sections 4.1.1 and 4.1.2. Details of the modeling of fire scenarios involving the turbine operating deck are included in Section 4.3.3.

The IPEEE 1993 Fire PRA analysis quantifies the effects of internal fire initiating events on core damage using the same linked event tree models used for the IPE. The event trees have been updated as needed to reflect plant changes over time and are updated through October 1993. For most fire initiators resulting from fire scenarios throughout the plant, the plant response was modelled using the following sequential event trees:

Electrical Support System Event Tree (ELECPWR)
Mechanical Support System Event Tree (MECHSP)
General Transient Event Tree (GENTRN)
Late Tree (LTREE)
Recovery Event Tree (RECV)

For fire scenarios involving the control room or the cable spreading room, separate event trees utilizing conservative assumptions were developed to model fires in specific locations, impacting specific equipment and operator responses.

Details of the event trees and the predominant plant responses to specific initiators are provided in Section 4.6.

4.0.2 SUMMARY OF FIRE ANALYSIS METHODOLOGY

Evaluation of DCPP fire events follows the scenario approach in which a large list of scenarios that may potentially take place is envisioned. A scenario is a chain of events starting with the ignition of a combustible. A scenario includes the initiation of a fire, its growth, the ignition of other combustibles, its detection, its suppression, and its impact on plant equipment.

The DCPRA-1988 fire analysis included a two-phase evaluation of potential fire hazard scenarios. The first phase is referred to as the spatial interactions analysis; the second phase is the fire risk assessment. The analysis for the IPEEE builds upon the DCPRA-1988 effort.

The spatial interactions analysis (see Section 4.1.3) is an integrated effort that identifies physical interactions among various power plant hazards to determine the important scenarios. As part of the spatial interaction study, a large set of internal fire scenarios was generated. The scenario list includes scenarios which impact a zone of origin plus one or more propagation zones as well as scenarios confined to one fire zone. The spatial interactions analysis includes a conservative estimate of scenario frequency based on the ignition frequency results for each fire zone combined with industry data on fire severity and scenario dependent geometry data. The spatial interactions analysis also screens the scenarios for significance based on frequency or impact on overall plant risk.

The fire risk assessment (see Section 4.1.4.1) provides a more detailed evaluation of the more significant scenarios generated from the spatial interactions analysis. Those fire scenarios judged to be significant in the fire risk assessment, were binned into fire initiators (see Section 4.4.2.1) according to the resulting equipment impact.

For fires in the control room (see Section 4.1.4.2) and the cable spreading room (see Section 4.1.4.3), a different process was used, wherein potential equipment damage states were postulated. The postulated equipment damage states serve as initiators for customized control room and cable spreading room fire event trees. The frequency of fire scenarios leading to each postulated damage state was estimated.

The initiators developed in the fire risk assessment and the control room and the cable spreading room fire analyses were quantified with the event trees as described in Section 4.6.

4.0.3 STATUS OF APPENDIX R MODIFICATIONS

Unit 1

SSER-23 (Reference 4-9), issued in June 1984, addressed PG&E compliance with the requirements of 10 CFR 50, Appendix R for Unit 1 and common plant areas. Included in SSER-23 were references to committed Appendix R modifications. The NRC required that all of the modifications referenced in the SSER be completed before 5 percent of rated power was exceeded.

PG&E's letter to the NRC dated July 6, 1984 (Reference 4-10) documented completion of the modifications. This was documented by the NRC in SSER-27 (Reference 4-11).

Unit 2

SSER-31 (Reference 4-12), issued in April 1985, addressed PG&E's compliance with Appendix R for Unit 2. Included in SSER-31 were references to committed Appendix R

modifications. The NRC stated that all of the modifications referenced in the SSER must be completed before Unit 2 fuel load.

PG&E's letter to the NRC dated June 6, 1985 (Reference 4-13) documented completion of the modifications. This was documented by the NRC in SSER-32 (Reference 4-14).

4.1 FIRE HAZARD ANALYSIS

4.1.1 IGNITION FREQUENCY SENSITIVITY STUDY

The fire analysis included in the DCPRA-1988 used a PLG proprietary fire database to generate annual fire ignition frequencies for each structure (or fire area) in the plant. These structure fire frequencies were apportioned to individual fire zones within each fire area using a Weighted Area Method. The Weighted Area Method involves apportioning fires in proportion to the effective floor area in each fire zone relative to the total effective floor area in the fire area. Prior to computing the area sums or area ratios, each fire area was weighted by a correction factor. The correction factor (ranging from 0.0 to 2.0) was determined by expert judgment, considering the combustible loads, the equipment (potential ignition sources), and personnel traffic in each fire zone. The actual physical area of each fire zone was multiplied by its correction factor to obtain an "effective area". The correction factors serve to apportion a greater fire ignition frequency to fire zones having a greater supply of potential ignition sources. At that time, this methodology represented an enhancement over earlier PRAs that relied solely on physical area ratios for apportioning fires.

SSER-34 concluded that the fire portion of the PRA was acceptable and capable of finding and ranking fire vulnerabilities in the plant. The NRC Staff requantified a fire scenario originating on the turbine operating deck, making the following observation:

> "The PRA used 2E-3/yr as the initiating event frequency. The PRA estimate was based on weighting the estimate of the frequency for all fires in turbine buildings by the area fraction for the turbine building operating deck. However, a disproportionate number of fires that occur in turbine buildings occur on the operating deck, and, since there are enough operating data to obtain an estimate from experience directly, that was done in the staff analysis."

In response to this comment and in light of the availability of more contemporary fire events data, an ignition frequency sensitivity study (Reference 4-15) reassessed the fire ignition frequencies used in the DCPRA-1988. The DCPRA-1988 fire ignition frequencies were compared with frequencies derived from data appearing in NUREG/CR-4840 (Reference 4-16). Also, the EPRI fire events database, FEDB (Reference 4-17), was used to estimate updated fire ignition frequencies. Additionally, an enhancement to the methodology for apportioning fire ignition frequency to specific fire zones within the Turbine Building was developed and applied. To facilitate comparison, consistent fire area definitions were employed for each of the three methods. The fire areas (and sub-areas) are listed below:

- Control Room
- Cable Spreading Room
- Diesel Generator Area
- Turbine Building
  - Transformer Yard
  - Turbine Building Switchgear

Turbine Building Non-Switchgear
Auxiliary Building
Auxiliary Building Balance
Auxiliary Building Switchgear
Battery Rooms
Intake
Radwaste

The results designated herein as FEDB were derived from US PWR experience, at power, post-commercial operation fire events. The fire events were assigned to the appropriate fire area based on the locations listed in the database. Additionally, the correction factors and area weighting methodology from the DCPRA-1988 were used to apportion the applicable plant-wide component fire events to each fire area. The years of PWR commercial operating experience from the FEDB was used to convert the sum of fire events (location specific plus plant-wide components) to an annual fire ignition frequency for each fire area.

Table 4.1-1 lists the resulting fire ignition frequency for each area as computed from each method investigated in the fire ignition sensitivity study. The DCPRA-1988 results are conservative with respect to the other two methods for control room and cable spreading room fires. For the other main areas (diesel generator, turbine building, and auxiliary building) the DCPRA-1988 ignition frequencies were less than the ignition frequencies computed by the other two methods. The plant totals for annual fire frequency from each of the three methods shows the DCPRA-1988 method producing the lowest value at 0.0934 fires per year. The total frequency for the NUREG/CR-4840 approach was 0.132 fires per year, while the plant total for the FEDB approach was 0.2075 fires per year. Most of the increase of the plant-wide fire frequency in the FEDB data is attributed to a greater turbine building fire frequency.

4.1.2 UPDATED IGNITION FREQUENCY

Based on the comparisons described above, a conservative strategy was adopted for updating ignition frequencies for the IPEEE DCPRA-1993 Fire PRA, wherein the FEDB-based results replace the DCPRA-1988 results for those fire areas where the FEDB result exceed the DCPRA-1988 result by more than a factor of 2 (greater than 100 percent). Following this approach, the ignition frequencies for the control room, cable spreading room, and auxiliary building remained unchanged; the ignition frequency for the turbine building was increased by a factor of 5.8, and the diesel generator area ignition frequency was increased by a factor of 2.3. The resultant plant total fire frequency, designated as IPEEE 1993 Fire PRA in Table 4.1-1, was 0.1942 fires per year. Except for the turbine building, the updated area ignition frequencies were apportioned to individual fire zones using the same area and equipment weighting correction factors employed in the DCPRA-1988.

In addition to updating the building annual fire ignition frequency as described above, an enhancement to the methodology for apportionment of fire frequency to individual fire zones within the turbine building non-switchgear (TB Non-SWGR) sub-area was applied.

This approach, designated as the Assigned Fire Method, used judgement based on the event descriptions and affected components available in the FEDB to assign 27 applicable fire events to an appropriate fire zone at DCPP within the TB Non-SWGR sub-area. Additionally, the plant-wide component fire events in the database associated with the TB Non-SWGR area were allocated to fire zones using the Weighted Area Method. The sum of fire events from these two sources was converted to an annual ignition frequency for each fire zone.

A comparison of TB Non-SWGR fire zone ignition frequencies, as computed by the Assigned Fire Method versus ignition frequencies computed by the Weighted Area Method alone, is provided in Table 4.1-2. The IPEEE Fire PRA analysis uses the Assigned Fire Method results described above for the TB Non-SWGR fire zones only. The Weighted Area Method is used for all other areas.

New fire zone ignition frequencies used in the IPEEE 1993 Fire PRA are compared with the DCPRA-1988 frequencies in Table 4.1-3.

4.1.3 SPATIAL INTERACTIONS ANALYSIS

The analysis of potential fire hazard scenarios, which forms the basis of the fire PRA for the DCPRA-1988 analysis and the IPEEE analysis, is a two-phase evaluation process. Each phase is characterized by its own set of assumptions and analysis detail. The first phase is referred to as the spatial interactions analysis. The spatial interactions analysis makes use of the physical location of equipment in the plant in relation to hazard sources to generate the fire scenarios. The second phase of the evaluation process is the fire risk assessment. The fire risk assessment focuses a more detailed analysis on the more significant scenarios. The fire risk assessment is discussed in Section 4.1.4. The goal of the process is to identify those significant frequency fire hazard scenarios that could lead to combinations of equipment damage that might contribute to overall plant risk. The significant scenarios are binned into fire initiators according to the type of equipment damage resulting from the scenarios. The sum of fire hazard scenario frequencies with the same equipment damage serves as the initiator frequency. The initiator is then applied to the relevant event trees which model the plant response and allow quantification of core damage frequency. The fire initiators are discussed in Section 4.4.2. Plant response models are discussed in Section 4.6.

As part of the spatial interaction study, a large set of internal fire scenarios was generated. The spatial interactions analysis postulates both localized and propagation scenarios for each fire zone. Propagation scenarios are postulated to originate in a fire zone and propagate to one or more adjacent fire zones. A unique scenario identifier is applied to each propagation scenario.

The spatial interactions analysis generated 323 fire and explosion scenarios originating in 87 fire zones. Based on the spatial interactions analysis screening assessment of these postulated scenarios, 140 scenarios were subjected to the Fire Risk Assessment.

The primary objective of the spatial interactions analysis was to identify, based on spatial commonalities at DCPP, those physical interactions involving power plant environmental hazards, such as fire, flood, and steam, that could cause an initiating event and intersystem dependent failures that would contribute significantly to risk. Databases were developed to cross-reference equipment items, hazard sources, mitigative features, and their associated locations to support the identification and analysis of spatial interactions scenarios. Systematic data collection and scenario identification methods were used to ensure a high level of rigor and completeness in the results. A scoping assessment was performed to determine the relative importance of the identified scenarios. The scoping assessment estimated the occurrence frequency of each environmental hazard scenario, then, based on the impact each scenario had on plant systems, estimated the quantitative impact each scenario had on overall plant risk. The scenarios judged to be of significant importance to overall plant risk were then reintroduced into the fire risk assessment for detailed quantification "assessment" and incorporation into the risk model.

The spatial interactions analysis task can be divided into two parts: (1) the identification of plant environmental hazard scenarios, and (2) the assessment of their relative importance to risk. Several computerized databases were developed for the first part. These databases were presented in the DCPRA-1988 report as tables. The ultimate objective of developing these databases was to define classes of environmental hazard scenarios that could be important to plant risk. Once these classes of scenarios were defined in terms of affected DCPRA-related equipment, the occurrence frequency of each scenario was estimated. For fire and explosion scenarios, the occurrence frequency estimate is based on the fire zone ignition frequency. For propagation scenarios, conservative multiplicative geometry, severity, and propagation factors are used with the ignition frequency to estimate scenario occurrence frequency. Then, each fire scenario was evaluated by the DCPRA plant model analysts to determine whether it should be reintroduced into the fire risk assessment for detailed analysis and quantification based on its relative importance to risk.

Briefly, the spatial interactions analysis proceeded as follows. First, the analysis team collected some basic information tables; i.e., a list of location designator specific to Diablo Canyon, a generic list of hazard types applicable to all nuclear plants, and a list of equipment included in the PRA internal event models. Next, the analysis team prepared the other tables of spatial analysis information. The information was compiled from reviews of existing references and by a comprehensive plant walkdown. For each location in the plant, the spatial analysis team identified the equipment located within that location, including the control and power cables needed to support the equipment of interest. They also identified the hazard sources found in these locations, the forms of mitigation available to minimize the impact of each hazard source, and the available paths for propagation of the impact of the hazard source from one location to another. All of this information was documented in the form of tables, as described in Section F.2.1 of the DCPRA-1988.

The next step in the analysis was to identify environmental hazard scenarios specific to Diablo Canyon. This was done in stages by first developing tables that cross-referenced the basic information in the first tables; e.g., equipment and location cross-reference,

source of hazard and location cross-reference, etc. This process is described in detail in the DCPRA-1988, Section F.2.2. Sources of hazards related to equipment included in the PRA internal event models and from other equipment were considered. A localized scenario (i.e., one that only affected the location in which the hazard originates) was first identified for each hazard in each location. Additional scenarios were then identified that involved the same hazard sources but included propagation to one or more other locations. The potential for propagation between locations was considered for each hazard source in the initial location.

The frequency of each environmental hazard scenario identified was then estimated. These point estimate frequencies were developed from data gathered from the entire U.S. industry. The frequencies assigned were intended to be conservative so that they could be used to screen the insignificant scenarios from further consideration. The frequencies were conservative in that very little credit was assumed for hazard mitigation, particularly within the fire area of origination.

The final task of the spatial interaction analysis was to determine the impact on plant equipment of each scenario identified. For this task, all equipment found in each location impacted by a scenario was conservatively assumed to fail. This screening approach was taken so that scenarios with minimal impact could quickly be identified. The lists of equipment impacted by each scenario were then reviewed by the PRA plant and system model analysts. Scenarios with minimal impact on plant systems were easily identified and dismissed from further consideration. The impacts of some scenarios were compared against similar sequences from other causes (i.e., from the internal event results) and, by comparison, seen to be of relatively small importance. Such scenarios were also dismissed from further consideration. Scenarios judged to be of potential significance were identified for additional review in the fire analysis. The disposition of each identified environmental hazard scenario, especially whether it is considered further for more detailed analysis, is documented in the scenario tables.

Details of the development of the databases used in the spatial interactions analysis are described in the DCPRA-1988 report, Sections F.2.1 through F.2.3. The database manipulations serve to map hazard sources, equipment designator, equipment susceptibility, and forms of mitigation to locations. From the tables, localized scenarios were generated for each fire zone. Propagation scenarios were generated based on the available propagation paths in each fire zone.

Each scenario is really a class of scenarios that can be broken down into separate, more refined scenario subclasses or into individual scenarios if they are chosen to be considered in greater detail in the fire risk assessment. The mitigating features associated with each scenario are listed in the tables but do not quantitatively impact the scenario frequencies in the spatial analysis.

For frequency of fire and smoke scenarios, the fire zone ignition frequency results of Section 4.1.2 were used. Many of the fire scenarios involve fire propagation to a number of components or adjacent areas. The frequencies of these scenarios were evaluated according to the methodology described in Reference 4-18. The fire frequencies were

adjusted to reflect the location of the originating fire (geometry factor), severity of the originating fire (severity factor), the likelihood of the fire not being extinguished prior to damage to important equipment, and other factors such as human error or an ineffective door. Typical values of these factors used in the propagation scenario frequency quantification are shown in Table 4.1-4. The selection of the severity factor data in Table 4.1-4 was based on a cumulative probability curve of fire severity developed for the scenario screening purposes (see Figure 4.1-1). In this figure, the fire severity is measured in terms of the fire radius. The probability associated with a specific fire radius represents the likelihood of a fire at a given location to be severe enough to damage the equipment within that fire radius. For example, the value 0.01 is used to indicate 1 out of 100 fires that has a fire radius greater than or equal to 30 feet. The data for the propagation factor are based on similar scenarios quantified in other studies. The nonsuppression factor is conservatively taken as 1.0 in the spatial interaction analysis. Should any fire scenario become risk significant, a more detailed examination of the fire suppression capability for the scenario may be conducted in the fire risk assessment.

The final task in the spatial interactions analysis was to screen all the fire scenarios to determine which ones should be reintroduced in the fire risk assessment for detailed quantification and incorporation into the PRA model. The following assumptions were used in the evaluation of hazard scenario impact on overall plant safety.

1. Each scenario is assumed to precipitate an initiating event. If the PRA-related equipment affected by the hazard does not directly cause an initiating event, it is assumed that the reactor is manually tripped in response to the hazard.

2. The mitigation features associated with each hazard scenario are listed for future reference, but are assumed not to work in this screening analysis.

3. For each location affected by a given scenario, all equipment susceptible to the hazard type associated with that scenario is assumed to fail.

Although conservative, these assumptions provided a good basis for scenario significance screening. When the scenarios found to be important were reintroduced in the fire risk assessment, these conservative assumptions were modified to make the risk quantification as realistic as practical for the most significant scenarios.

To establish their importance to risk, the fire hazard scenarios were first divided into two categories. The first category included all scenarios that impact only one PRA system. The contribution of these scenarios was compared to the system unavailability or system failure frequency determined by the individual system analysis. The second category included all scenarios that impact more than one system. Using knowledge of the logic models developed for the event sequence and systems analyses, the DCPRA-1988 systems analysts screened these second category scenarios for importance to risk. For scenarios leading directly to core damage, the core damage frequency equals the fire hazard scenario frequency. For scenarios that do not directly lead to core damage, (i.e., additional, independent equipment failures are necessary to cause core damage) the conditional split fractions of independent equipment from the systems analyses were used

with the scenario frequency to estimate core damage frequency. Scenarios were first evaluated individually, then as groups, to ensure that similar scenarios that may be insignificant on an individual basis, but are significant when considered collectively, were not prematurely screened.

As discussed in Section 4.1.2, some fire zone fire ignition frequencies were updated for the IPEEE 1993 Fire PRA. As a result of the increased ignition frequencies, the screening process was revisited for the affected scenarios. It was not necessary to reintroduce any scenarios that had been previously eliminated by the screening process.

4.1.4 INTERNAL FIRE ANALYSIS METHODOLOGY

The fire risk assessment provided a more detailed evaluation of the more significant fire scenarios as identified in the spatial interactions analysis using the approach discussed in Section 4.1.4.1.

For the analysis of fires in the control room (Section 4.1.4.2) and the cable spreading room (Section 4.1.4.3), a specialized evaluation process was used.

4.1.4.1 Fire Risk Assessment

The starting point for the fire risk assessment is based on the results of the spatial interactions analysis. The fire risk assessment represents a more detailed reanalysis of the important fire scenarios that survived the spatial interactions analysis screening process. In the spatial interaction analysis, it was conservatively assumed that a fire would disable all PRA equipment and cables within a fire area, independent of the severity or location of the given fire. In contrast to the spatial interactions analysis, the fire risk assessment used a detailed review of cable and conduit routings and the layout and arrangement of plant components to estimate event frequencies and equipment damage for a given fire scenario. Included in this evaluation was a review of the findings identified by the Diablo Canyon Appendix R (Reference 4-19) review. Based on these additional reviews, the fire risk assessment provides a more realistic estimate of the fire scenario frequency through the generation of appropriate geometry and severity factors, and also supports the determination of the extent of equipment damage.

The generic expression of fire scenario frequency, R_I, inside a fire area that causes a combination of equipment damage can be expressed as

$$R_I = R_{AREA} F_{G,I} F_{S,I} F_{NS,I} F_{HE,I} \qquad \text{(Eq. 4.1-1)}$$

where

R_{AREA} = the annual frequency of fire of any severity in a given fire area.

$F_{G,I}$ = the conditional frequency of fire scenario I occurring at a specified location, given that a fire has occurred in that fire area (geometry factor).

$F_{S,I}$ = the conditional frequency of fire scenario I that is initiated and has sufficient severity to cause failure of a combination of plant equipment and cables (severity factor).

$F_{NS,I}$ = the conditional frequency of fire scenario I that is not suppressed by the suppression features before it affects equipment.

$F_{HE,I}$ = the conditional frequency of operators' failure to carry out the recovery actions for fire scenario I.

For each fire scenario leading to core damage the scenario frequency was calculated using the above equation. As a conservative measure, for most scenarios, the non-suppression factor, $F_{NS,I}$ is set equal to 1.0. The estimation of the geometry factor is based on the fraction of floor area covered by the plant equipment of interest in relation to the total area of a given fire area or zone. The severity factor can be evaluated from the physical separation between the PRA equipment of interest.

After establishing the geometry and severity factors, an evaluation of the scenario frequency was performed. Another screening process was carried out based on the comparison of fire scenario frequencies against the likelihood of system failure due to all other causes. If the accumulated fire scenario frequencies for one category of system failure amount to more than 10 percent of all other causes, then these fire frequencies were incorporated into the event tree models. Otherwise, the fire scenarios were considered to have an insignificant contribution to core damage frequency and were not included in the risk quantification. Table 4.4-2 in Section 4.4 lists all fire scenarios included in the core damage frequency quantification.

4.1.4.2 Control Room Fires

Specific control room fire scenarios were postulated to model the impact of fires in the control room. To establish the potential and significant fire scenarios and to analyze related accident sequences, all of the electrical cabinets and control boards were reviewed. This review was based on the consideration of the impact of the fire on the equipment included in the PRA event tree and fault tree models.

The main control boards were analyzed section by section. For each section, it was assumed that fire damaged a contiguous area on the control panel. Different damage areas were considered for each panel, depending on the combination of plant equipment affected.

Fires starting outside the control panels may cause the same extent of damage to the control and instrumentation circuitry but are deemed to be significantly less likely than panel fires because, inside the panels, there are electrical components and cables; therefore, the amount of combustibles is much greater than that outside the panels. Finally, a fire outside the panels would very likely be detected and extinguished within a short time by control room personnel.

The following control panels have been identified as potential contributors to risk:

- Vertical Board VB-1. Contains controls and instrumentation for auxiliary saltwater (ASW), component cooling water (CCW), containment fan cooler units (CFCUs), containment spray (CS), safety injection (SI), and residual heat removal (RHR) systems.

- Vertical Board VB-2. Contains controls and instrumentation for the chemical volume control system (CVCS), pressurizer instrumentation and valves, and reactor coolant pumps.

- Vertical Board VB-3. Contains controls and instrumentation for the steam generator, MSIVs, auxiliary feedwater system (AFW), and the main turbine.

- Vertical Board VB-4. Contains controls for 4 kV buses F, G, and H; containment HVAC; turbine control; and circulating water system.

Based on the layout of instrumentation and controls above, specific representative control room fire scenarios were postulated to model the impact of control room fires. These control room fire scenarios, detailed in Section 4.4.2.2, serve as initiators to the control room event trees. The control room fire event trees in Section 4.6.1 serve to quantify the impact of control room fires on core damage frequency.

For analysis of control room fires, detailed equipment damage scenarios were developed and quantified using a variation of the general fire frequency methodology described in the previous section. The generic expression of fire scenario frequency, R_I, inside the control room that causes a combination of equipment damage can be expressed as

$$R_I = R_{CR} F_{G,S_I} F_{HE,I} \qquad \text{(Eq. 4.1-2)}$$

where

R_{CR} = the annual ignition frequency of fire of any severity in the control room.

F_{G,S_I} = the combined geometry and severity factors for fire scenario I, (the conditional frequency of fire scenario I occurring at a specified location (geometry factor), having sufficient severity to cause failure of a combination of plant equipment and cables (severity factor) given that a fire has occurred in the control room).

$F_{HE,I}$ = the conditional frequency of operators' failure to carry out the recovery actions for fire scenario I. This includes consideration of control room evacuation and control from the hot shutdown panel.

Details of the derivation of the combined geometry and severity factors for control room fire scenarios are included in Section 4.3.1. Because the likelihood of suppression is not separable from the data used in developing the control room fire severity curve, the non-

suppression factor of Equation 4.1-1 is contained within the combined geometry and severity factor.

Several operator recovery actions contribute to the recovery failure frequency factor of Equation 4.1-2. These actions are described in Section 4.4.3.2.

4.1.4.3 Cable Spreading Room Fires

The cable spreading room is located directly below the control room. A fire inside this region could disable the control features provided by the main control room. Alternate plant shutdown capabilities are available through the hot shutdown panel, dedicated shutdown panel, breakers inside the switchgear rooms, and local manual control features associated with individual pieces of equipment.

There is much similarity between the cable spreading room and the control room from the standpoint of analyzing the risk of a fire-induced scenario. The cable spreading room, like the control room, contains the control and instrumentation cables of much of the key equipment of the plant. It also contains, among other things, control and instrumentation racks associated with plant operation.

To perform a detailed analysis on the impact of fires in the cable spreading room, information about the exact location of the important cables is needed. All available cable-routing diagrams for the cable spreading room were carefully inspected. However, due to the compactness of the arrangement of the cables inside this room, engineering judgments must be exercised in the development of critical fire scenarios. Therefore, a somewhat conservative approach is followed to establish the frequency of core damage from a cable spreading room fire.

Based on the layout of instrumentation and controls discussed above, specific representative cable spreading room fire scenarios were postulated to model the impact of cable spreading room fires. These cable spreading room fire scenarios, detailed in Section 4.4.2.3, serve as initiators to the cable spreading room fire event tree. The cable spreading room fire event tree in Section 4.6.1 serves to quantify the impact of cable spreading room fires.

The frequency of a cable spreading room fire scenario, R_{CS_I}, resulting in core damage is obtained from

$$R_{CS_I} = R_{CS}\, F_{G,CS_I}\, F_{S,CS_I}\, F_{HE,CS_I} \qquad \text{(Eq. 4.1-3)}$$

Where

R_{CS} = the annual ignition frequency of fire of any severity in the cable spreading room.

F_{G,CS_I} = the conditional frequency of fire affecting a critical set of cables in such a way that damage to the equipment postulated to be impacted

in scenario I may occur, given that a cable spreading room fire has occurred (geometry factor).

F_{S,CS_I} = the conditional frequency of fire affecting a critical set of cables being of sufficient severity to damage the combination of equipment postulated to be impacted in scenario I (severity factor).

F_{HE,CS_I} = the conditional frequency of operators' failure to carry out the recovery actions for cable spreading room fire scenario I.

Details of the derivation of the geometry and severity factors for cable spreading room fire scenarios are included in Section 4.3.1. The cable spreading room fire analysis assumes no explicit credit for suppression. Several operator recovery action contribute to the recovery failure frequency factor of Equation 4.1-3. These actions are described in Section 4.4.3.3.

Table 4.1-1. Comparison of Annual Fire Frequency Results					
		Annual Fire Frequency			
Area	Sub-Area	NUREG/ CR-4840	DCPRA-1988	FEDB	IPEEE 1993 Fire PRA
Control Room		4.41E-3	4.90E-3	4.624E-3	4.90E-3
Cable Spreading Room		2.68E-3	6.70E-3	5.281E-3	6.70E-3
Diesel Generator		2.31E-2	1.78E-2	4.153E-2	4.153E-2
Turbine Building		3.45E-2	1.60E-2	9.304E-2	9.304E-2
	Transformer Yard	-	-	1.146E-2	1.146E-2
	TB SWGR	2.43E-3	3.51E-3	1.851E-2	1.851E-2
	TB Non-SWGR	3.21E-2	1.25E-2	6.308E-2	6.308E-2
Auxiliary Building		6.74E-2	4.80E-2	6.299E-2	4.80E-2
	AB Balance	6.39E-2	3.63E-2	4.536E-2	3.63E-2
	AB SWGR	5.38E-4	3.26E-3	4.091E-3	3.26E-3
	Battery	2.97E-3	3.53E-3	7.030E-3	3.53E-3
	Intake	-	4.92E-3	3.971E-3	4.92E-3
	Radwaste	-	-	2.546E-3	-
TOTAL		1.32E-1	9.34E-2	2.075E-1	1.942E-1

Table 4.1-2. Turbine Building (Non-SWGR) Fire Ignition Frequency				
Fire Zone	Fire Zone Location Name	Assigned Fire Method	Weighted Area Method	DCPRA-1988
11-D	Diesel Generator Corridor El. 85	1.040E-4	2.29E-4	4.52E-5
14-A-85	T/B Bldg Main Comps El. 85	1.512E-2	1.64E-2	3.25E-3
14-B	Clean & Dirty Lube Oil El. 85	1.370E-3	2.13E-4	4.22E-5
14-E	CCW HX Area El. 85	9.818E-4	2.16E-3	4.27E-4
14-A-104	T/B Bldg Main Comps El. 104	8.756E-3	1.64E-2	3.25E-3
15	T/B Lube Oil Reservoir El. 104-119	1.814E-3	1.19E-3	2.35E-4
14-A-119	T/B Bldg Main Comps El. 119	1.130E-2	1.64E-2	3.25E-3
14-C	Electric Load Center El. 119	1.723E-3	9.90E-4	1.96E-4
14-D	T/B Bldg Operating Deck El. 140	2.191E-2	8.98E-3	1.78E-3
16	Machine Shop El. 85-140	0	0	0
17	Old Unit 1&2 Warehouse El. 119 (No Longer Warehouse)	0	0	0
TOTAL		6.308E-2	6.308E-2	1.248E-2

Note: Both methods above employ the Weighted Area Method for allocating fire ignition risk due to plant-wide component fires. The IPEEE 1993 Fire PRA uses the Assigned Fire Method results for the turbine building non-switchgear fire zones only.

Table 4.1-3. Fire Zone Fire Ignition Frequencies (DCPRA-1988 Zone Values and Updated Zone Values for IPEEE)					
Location				Annual Fire Frequency	
Area	Sub-Area	Zone	Zone Location Name	DCPRA-1988	IPEEE
Control Room		8-C	Main Control Rm El. 140	4.90E-3	4.90E-3
Cable Spreading Room		7-A	Cable Spreading Rm U1 El. 127	6.70E-3	6.70E-3
Diesel Generator				1.78E-2	4.153E-2
		11-A-1	Diesel Generator Rm 1-1 El. 85	3.33E-3	7.78E-3
		11-A-2	Diesel Generator Air Supply and Exhaust 1-1 El. 107	4.94E-4	1.16E-3
		11-B-1	Diesel Generator Rm 1-2 El. 107	3.33E-3	7.78E-3
		11-B-2	Diesel Generator Air Supply and Exhaust 1-2 El. 107	7.41E-4	1.73E-3
		11-C-1	Diesel Generator Rm 13 El. 85	3.33E-3	7.78E-3
		11-C-2	Diesel Generator Air Supply and Exhaust 1-1 El. 107	5.72E-4	1.34E-3
		13-F	Electric Shop and Office El. 119	5.98E-3	1.40E-2
Turbine Building				1.60E-2	9.304E-2
	Transformer Yard	28	Main Transformer	-	1.146E-2
	TB SWGR			3.51E-3	1.851E-2
		10	12-kV SWGR Room El. 85	2.01E-3	1.06E-2
		12-A	4-kV Cable Spreading Room El. 107 - F Bus	1.34E-4	7.07E-4
		12-B	4-kV Cable Spreading Room El. 107 - G Bus	1.34E-4	7.07E-4
		12-C	4-kV Cable Spreading Room El. 107 - H Bus	1.34E-4	7.07E-4
		12-E	Isophase Bus Area El. 104	6.70E-5	3.53E-4
		13-A	ESF SWGR Rm F El. 119	2.81E-4	1.48E-3
		13-B	ESF SWGR Rm G El. 119	2.81E-4	1.48E-3
		13-C	ESF SWGR Rm H El. 119	3.02E-4	1.59E-3
		13-D	Excitation SWGR Rm El. 119	4.19E-5	2.21E-4
		13-E	SWGR Ventilation Rm El. 119	1.28E-4	6.75E-4
	TB Non-SWGR				6.308E-2
		11-D	Diesel Generator Corridor El. 85	4.52E-5	1.04E-4
		14-A-85	Turbine Building Main Comps El. 85	3.25E-3	1.51E-2
		14-A-104	Turbine Building Main Comps El. 104	3.25E-3	8.76E-3
		14-A-119	Turbine Building Main Comps El. 119	3.25E-3	1.13E-2
		14-B	Clean & Dirty Lube Oil El. 85	4.22E-5	1.37E-3
		14-C	Electric Load Center El. 119	1.96E-4	1.72E-3

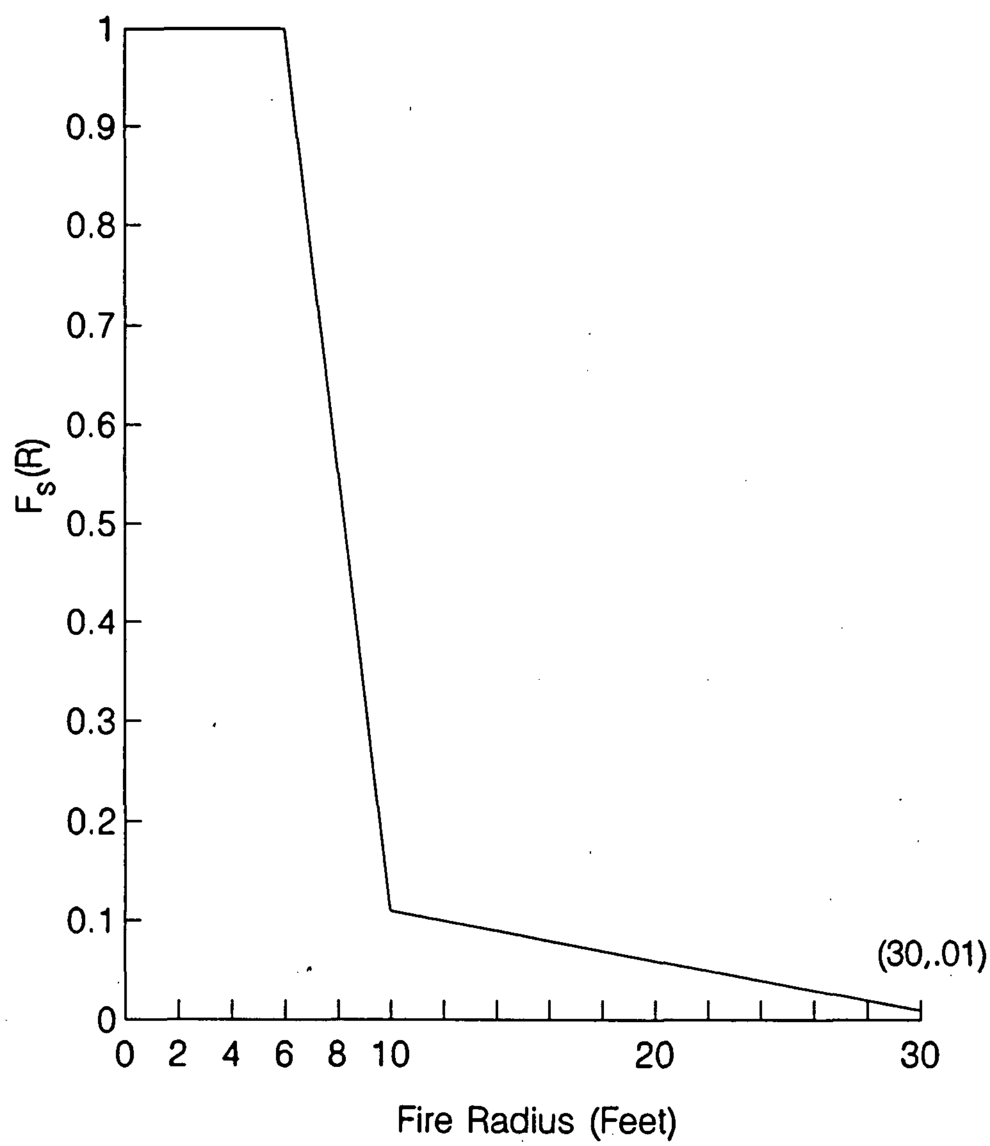
Table 4.1-3. Fire Zone Fire Ignition Frequencies (DCPRA-1988 Zone Values and Updated Zone Values for IPEEE)					
Location				Annual Fire Frequency	
Area	Sub-Area	Zone	Zone Location Name	DCPRA-1988	IPEEE
		14-D	Turbine Building Operating Deck El. 140	1.78E-3	2.19E-2
		14-E	CCW HX Area El. 85	4.27E-4	9.82E-4
		15	Turbine Building Lube Oil Reservoir El. 104-119	2.35E-4	1.81E-3
		16	Machine Shop El. 85-140	0	0
		17	Old Unit 1&2 Warehouse El. 119 (No Longer Warehouse)	0	0
Auxiliary Building				4.80E-2	4.80E-2
	AB Balance			3.63E-2	3.63E-2
		3B1	RHR PP&HX Rm 11 El. 60-104	0	0
		3B2	RHR PP&HX Rm 12 El. 60-104	0	0
		3-A	Hold-Up Tank Rm El. 54-115	0	0
		3-AA	CVCS Comps Area El. 115	2.93E-3	2.934E-3
		3-B-1	RHR PP&HX Rm 11 El. 60-104	2.88E-4	2.881E-4
		3-B-2	RHR PP&HX Rm 12 El. 60-104	2.88E-4	2.881E-4
		3-B-3	BIT Area U1 El. 60-75	2.58E-4	2.586E-4
		3-C	Aux Building Pipe Tunnel El. 54	2.13E-3	2.133E-3
		3-F	Containment Spray PPS Rm U1 El. 75	1.20E-3	1.197E-3
		3-H-1	Centrifugal Charging PPS Rm U1 El. 75	1.52E-3	1.523E-3
		3-H-2	P/D Charging PPS Rm U1 El. 75	1.65E-4	1.651E-4
		3-J-1	Comp Cooling Water PP Rm 11 El. 75	4.05E-4	4.047E-4
		3-J-2	Comp Cooling Water PP Rm 12 El. 75	4.05E-4	4.047E-4
		3-J-3	Comp Cooling Water PP Rm 13 El. 75	5.73E-4	5.734E-4
		3-L	CVCS Comps Area El. 85-104	7.49E-4	7.490E-4
		3-M	SI PPS Rm U1 El. 85	9.80E-4	9.802E-4
		3-S	Hot Machine Shop El. 140	2.20E-3	2.198E-3
		3-X	Pipe Way Opening El. 104	4.05E-4	4.054E-4
		4-A	Counting & Chem Lab El. 85	1.39E-3	1.386E-3
		4-A-1	Chem Lab Area & G Bus Compartment El. 85	2.46E-5	2.459E-5
		4-A-2	Chem Lab Area & H Bus Compartment El. 85	2.46E-5	2.459E-5
		4-B	Shower Locker & Access CRL El. 85	9.31E-4	9.310E-4
		6-A-5	Electrical Area U1 El. 115	2.37E-4	2.375E-4
		7-C	Communication Rm U1 El. 127	2.63E-4	2.631E-4
		8-B-1	Fan Rm U1 El. 140	1.34E-3	1.345E-3

Table 4.1-3. Fire Zone Fire Ignition Frequencies (DCPRA-1988 Zone Values and Updated Zone Values for IPEEE)					
Location				Annual Fire Frequency	
Area	Sub-Area	Zone	Zone Location Name	DCPRA-1988	IPEEE
		8-B-3	CRL Rm Ventilation Equipment Rm U1 El. 154	1.83E-3	1.832E-3
		8-B-5	480v SWGR Fan Rm U1 El. 163	7.90E-5	7.905E-5
		8-B-7	Fan Rm El. 163	5.07E-4	5.077E-4
		8-A	Computer Rm U1, CAL Rm El. 140	1.84E-4	1.844E-4
		8-E	Office & Storage Rm, CRL Rm El. 140	0	0
		8-G	SSPS Rm U1 El. 140	2.11E-4	2.108E-4
		S-1	Stairway, West El. 57-154	5.69E-5	5.691E-5
		S-2	Stairway, Center-East El. 85-163	7.90E-5	7.905E-5
		S-3	Stairway, North-East El. 64-140	1.26E-4	1.265E-4
		S-5	Stairway, Center-West El. 85-140	0	0
		3-BB	Penetration Area El. 85-115	2.31E-3	2.307E-3
		2	Aux. Boiler El. 85	1.10E-3	1.096E-3
		31	Fuel Building Corridor	0	0
		3-O	Spent Fuel Pool PPS & HXS Area, Fuel Handling Building El. 100	1.18E-3	1.180E-3
		3-P-10	Corridor El. 140	0	0
		3-P-11	Corridor El. 140	0	0
		3-P-12	Ventilation Rm El. 85	0	0
		3-P-1	Supply Fan Rm El. 85	4.50E-4	4.497E-4
		3-P-2	Air Duct Rm El. 102	1.72E-3	1.723E-3
		3-P-2	Air Duct Rm El. 102	2.05E-3	2.052E-3
		3-P-3	Exhaust Fan & Filter Rm 2 El. 115	1.20E-3	1.200E-3
		3-P-4	Exhaust Fan & Filter Rm 1 El. 115	9.78E-4	9.781E-4
		3-P-5	Supply Fan Room El. 115	0	0
		3-P-6	Filter Rm El. 140	2.11E-4	2.108E-4
		3-P-7	Fan & Filter Rm El. 140	1.26E-4	1.265E-4
		3-P-8	Fan & Filter Rm El. 140	1.26E-4	1.265E-4
		3-Q-1	T/D AFW PP Rm El. 100	7.16E-4	7.167E-4
		3-Q-2	M/D AFW PPS Rm El. 100	3.88E-4	3.879E-4
		3-R	Spent Fuel Pool U1 El. 91-104	1.96E-3	1.956E-3
	AB SWGR			3.26E-3	3.26E-3
		5-A-1	480V Vital SWGR, 1F Bus El. 100	7.12E-4	7.127E-4
		5-A-2	480V Vital SWGR, 1G Bus El. 100	7.12E-4	7.127E-4
		5-A-3	480V Vital SWGR, 1H Bus El. 100	7.12E-4	7.127E-4
		5-A-4	480V N/V SWGR & HSD PNL U1 El. 100	1.12E-3	1.122E-3
	Battery			3.53E-3	3.53E-3

Table 4.1-3. Fire Zone Fire Ignition Frequencies (DCPRA-1988 Zone Values and Updated Zone Values for IPEEE)					
Location				Annual Fire Frequency	
Area	Sub-Area	Zone	Zone Location Name	DCPRA-1988	IPEEE
		6-A-1	Battery & DC SWGR, 11 Bus El. 115	7.33E-4	7.325E-4
		6-A-2	Battery & DC SWGR, 12 Bus El. 115	7.33E-4	7.325E-4
		6-A-3	Battery & DC SWGR, 13 Bus El. 115	7.33E-4	7.325E-4
		6-A-4	RX Trip SWGR U1 El. 115	1.33E-3	1.332E-3
	Intake			4.92E-3	4.92E-3
		30-A-5	CW PP Area El. 2	2.97E-3	2.971E-3
		30-B	CW PP CRL Rm El. 18	1.95E-3	1.949E-3
	Radwaste			-	-
TOTAL				9.34E-2	1.942E-1

Table 4.1-4. Factors Used in the Fire Propagation Scenario Frequency Quantification	
Item	Typical Values Used
1. Geometry Factor (The fraction of the floor area around the identified propagation path (e.g., door) to the total floor area of the fire initiating location)	0.01 0.05 0.1 0.125 0.17 0.2 0.25 0.3 0.5 0.75 1.0
2. Severity Factor (The fraction of fires in the area around the propagation path that are large enough to damage all the PRA-related equipment contained in both fire-originated and fire-destinated locations)	See Figure 4.1.3-1
3. Propagation Factor (The likelihood the propagation path is available for fire propagation)	
A. Door (Depending on the traffic and activity level at a specific location)	0.01 0.02 0.05 0.1 0.2
B. Hatch (Depending on the level of maintenance activity involved in opening the hatch)	0.01 0.05 0.1
C. Fire Damper (Depending on the types of fire damper and the smoke detector)	0.05 0.1
4. Nonsuppression Factor (The likelihood that the fires are not suppressed before equipment damage)	1.0

Figure 4.1-1. Severity Factor (F_S) As A Function Of Fire Radius For Spatial Interaction



4.2 REVIEW OF PLANT INFORMATION AND WALKDOWN

4.2.1 WALKDOWNS

A series of plant walkdowns were conducted to support the development of the external events portion of the DCPRA-1988 and to address IPEEE issues. The purpose of the walkdowns, the people involved, and any findings are summarized below.

As part of the DCPRA-1988 external events effort, a one-week, spatial interactions walkdown was conducted by members of the DCPRA spatial interactions team and PG&E personnel. The purpose of the spatial interactions walkdown was to identify and confirm certain component locations, fire hazard sources, and potential propagation paths.

As part of the IPE internal flooding assessment, a two-day plant walkdown with four PRA analysts was conducted to collect additional information and to confirm previous documentation and judgements on flood sources and their potential impact, propagation paths, and detection. The internal flooding assessment walkdown was documented by photographs of the important equipment and a table that includes the location, flooding sources, propagation paths, mitigating features, and possible PRA impacts in each location (Reference 4-20).

During the early stages of the IPEEE effort, a fire walkdown was conducted by two PRA analysts to collect data and photographs pertaining to ignition sources and location and numbers of electrical cabinets in particular fire areas.

An IPEEE fire walkdown was conducted on March 16 and 17, 1994. The participants included:

- Lead Fire PRA Analyst
- PRA Group Supervisor
- Fire Protection Engineers (2)
- On-site Engineering Fire Protection Engineer
- Lead Seismic PRA Analyst
- Seismic Interactions Specialist
- DCPP Fire Marshall

A walkdown plan was created prior to performing the IPEEE fire walkdown. The purpose of the IPEEE fire walkdown was to verify modeling assumptions and to address Fire Risk Scoping Study (Reference 4-2) issues. The scope of the IPEEE fire walkdown is summarized in Table 4.2-1. The following walkdown objectives were accomplished:

- As part of the Fire Risk Scoping Study response to the issue of "Seismically Induced Fires," the walkdown team inspected the conditions and locations for storage of compressed gas cylinders containing hydrogen and other flammable gases. The walkdown team inspected the storage of flammable gases and liquids in the chemistry laboratory area. The walkdown team also traced the hydrogen

line through the plant and verified the existence of guard piping in areas containing safety-related equipment.

- As part of the review of the Fire Risk Scoping Study issue of "Seismic Actuation of Fire Suppression Systems," the walkdown team verified aspects of the plant related to inadvertent actuation and suppression-induced damage, including train separation and the proximity of PRA equipment to fire water suppression system piping, sprinkler heads, drains, and flood barriers.

- As part of the Fire Risk Scoping Study response to the issue of "Seismic Degradation of Fire Suppression Systems" the walkdown team inspected the various degrees of structural support for fire water suppression systems throughout the plant, noting the distinctions between seismically qualified, seismically supported, and supported in accordance with good industrial practice (ANSI B31.1). The seismic interactions specialist also pointed out the valves which serve, in the event of a pipe break, to isolate the seismically qualified portions of the firewater system from the non-seismically qualified portions (see Section 4.8.1.3).

- As part of the configuration management effort, the walkdown team observed component and circuit train separation features on both an interzone basis and an intrazone basis. The fire protection engineers also pointed out aspects of the plant that are pertinent to Appendix R, including recent design changes, protective fire wrap and barriers, human recovery actions, and Fire Hazard Appendix R Evaluation (FHARE) issues.

- As part of the confirmatory portion of the IPEEE fire walkdown, the walkdown team discussed fire hazards and fire fighting preplans with the DCPP Fire Marshall. The Fire Marshall provided his perspective on severe turbine building and transformer fire scenarios.

- As part of the confirmatory portion of the IPEEE fire walkdown, the PRA analysts observed critical fire areas, confirming the reasonableness of geometry, severity, and propagation factors assumed in the fire scenario evaluation in relation to equipment locations within a zone and in relation to propagation pathways.

- As part of the confirmatory portion of the IPEEE fire walkdown, the walkdown team verified that severe turbine building fire scenario assumptions are reasonable. The issues investigated included the following:

  1. separation between the main unit turbine and generator and the 4-kV switchgear exhaust ventilation ducts;

  2. the activation mechanisms for switchgear exhaust ventilation duct fire dampers;

3. the geometry of the switchgear exhaust ventilation ducts in relation to fire, smoke, debris, oil or water spray and flood hazards on the turbine deck and the target equipment in the switchgear rooms;

4. the locations of fire fighting equipment and proximity to switchgear exhaust ventilation ducts;

5. the aspects and features which protect the 4-kV switchgear against a turbine deck flowing burning pool fire (or flood hazard) on the turbine deck as well as pathways for and protective features against a flowing burning pool or flood hazard through stairways to lower levels and under doors.

6. the DCPP Fire Marshall's perspective on severe turbine building fire and smoke hazards, turbine building vent capacity, Diablo Canyon turbine building fire experience, and fire fighting plans and response.

Table 4.2-1. IPEEE Walkdown Plan	
WALKDOWN TOPIC	ISSUES / LOCATIONS
Fire Water System - Wet Pipe Sprinklers	Become familiar with the components that supply firewater to the sprinkler system. Become familiar with the distinctions between seismically qualified, seismically supported, and not seismically supported. Make observations about the arrangement of firewater piping and nozzles with respect to safety-related equipment in the following fire areas: CCW HEX Room (TB 85 / U1 14-E / U2 19-E) "Other Areas" in TB El. 85' MDAFW Pump Room (FHB 104 / U1 3-Q-2 / U2 3-T-2)
Fire Water System - Deluge Fire Suppression System	Look at the valves that supply firewater to the deluge system. Look at the deluge system serving the Hydrogen Seal Oil skid (TB El. 85). Look at the deluge system serving the U1 main unit transformer yard.
Hydrogen Line	Observe the hydrogen line entering the turbine building. Observe the hydrogen line excess flow shutoff valve. Observe the "guarded" hydrogen line in the penetration area.
Other Explosives / Flammables	Observe the compressed gas storage area in the tool room for seismic vulnerability. Observe the gas bottle storage area outside the CCW HEX room for seismic vulnerability. Observe the storage conditions of flammables in the Chem Lab (Fire Area 4-A)

Table 4.2-1. IPEEE Walkdown Plan	
WALKDOWN TOPIC	ISSUES / LOCATIONS
Interview the Fire Marshall in the Turbine Building El. 140'	Note the Fire Marshall's perspective on Severe Turbine Building Fires. Address the following topics: Principal hazards Severity of 'worst case fire' Fire Preplans / Fire Brigade Activities Potential to impact TB structural steel/roof Severity of smoke hazard Draining versus cascading of water Potential for impact to 4-kV SWGR Fire Brigade actions to protect 4-kV SWGR 4-kV SWGR and/or 4-kV CSR fires
Review of Critical Fire Scenarios and Propagation Paths	Severe Turbine Building Fire 4-kV SWGR Room Fire Main Unit Transformer and 12-kV SWGR CCW HEX Room Vital 480VAC SWGR Non-Vital 480VAC SWGR CCW Pump Room MDAFW Pump Room Containment Penetration Area (El. 100' and 115') Access Control Chemistry Lab Electrical Area

4.2.2 CONFIGURATION MANAGEMENT

4.2.2.1 Living PRA Model

One requirement for the use of an existing fire PRA for the internal fires IPEEE is to ensure that the PRA "reflects the current as-built and as-operated status of the plant." One aspect providing that assurance is the use of a "living PRA model." The Diablo Canyon PRA is updated every 18 months. The update effort includes the data update that uses Diablo Canyon specific component failures to update basic failure rate data. As part of the system model update, all design changes are reviewed by the PRA systems analysts to ensure that pertinent changes are reflected in the updated system models. The system models determine the split fraction values used in the event trees for quantification. The event trees themselves are modified as needed to reflect changes in the plant design, plant operating procedures, or event sequence modeling. Also, updated values for human action error rates are generated as appropriate. The product of these changes at the data level, the system model level, and the plant response model level is the updated Diablo Canyon PRA model. The IPEEE 1993 Fire PRA uses data, split fractions, and event trees from the contemporary 1993 Diablo Canyon PRA (DCPRA-1993) model. Quantification of fire risk for the IPEEE 1993 Fire PRA, using the DCPRA-1993 ensures that the current best state of knowledge about the plant is incorporated into the modeling. In addition to the updates inherent in the DCPRA-1993, the custom event trees and human action values developed for modeling control room and cable spreading room fire scenarios have been updated and enhanced to provide a more explicit model of potential accident sequences.

4.2.2.2 Review of Design Changes Since DCPRA-1988

Another aspect of the configuration management effort involved the review of all design change notices issued between July 1988, when the DCPRA-1988 work was completed, and September 1993. Over 4700 design change notices were reviewed for the potential to impact assumptions implicit to the spatial relationship of components and fire zones or assumptions associated with the scenario evaluations. No new items were identified as a result of this review.

4.2.2.3 Review of Spatial Database Components Against Appendix R Report

The original PRA spatial database used Appendix R information from 1985, as well as layout drawings and walkdowns. Subsequent to the DCPRA-1988, PG&E performed an Appendix R Design Basis Documentation Enhancement Project, which resulted in reviews and revisions to Appendix R calculations and analyses (Reference 4-21). Changes to Appendix R supporting documentation have been the result of the following:

- design changes and procedure changes
- a more detailed review of safe shutdown functions and equipment
- additional cable and raceway information that was not utilized in the 1985 Appendix R analysis.

For all fire zones contributing to critical fire scenarios, the contained components and circuits listed in the contemporary Appendix R documentation were compared with the components and circuits listed in the PRA spatial interactions database. Where the Appendix R documentation listed safe shutdown components that were not in the fire PRA spatial database, the critical fire scenarios were reevaluated to incorporate the additional components. In some cases the PRA Analyst and a Fire Protection Engineer reviewed the routing and proximity of circuits within the fire zone. As a result of this process, some new fire scenarios were added to the list of critical fire scenarios and some were removed. The overall impact on the fire PRA of the differences between the Appendix R documentation and the spatial interactions database was judged to be small.

The Fire Protection Engineering group has worked closely with the PRA group in the development of the IPEEE 1993 Fire PRA. In reviewing the critical fire scenarios, the Fire Protection Engineering group has provided invaluable insights into fire induced failure mechanisms in electrical control circuits and details of circuit routing within fire zones and between fire zones.

4.2.3 REMOTE SHUTDOWN AND CONTROL ROOM CIRCUITRY

A thorough review of the DCPP safe shutdown analysis was performed under the Appendix R Documentation Enhancement Project. The review included a detailed review of the impact of a control room or cable spreading room fire on the ability to safely shutdown.

Design changes were issued and implemented to provide the necessary circuit isolation to ensure that control of 4-kV pumps and diesel generators (DGs) remained available at their respective remote control stations following a postulated control room/cable spreading room fire.

Details of this effort are addressed in section 4.8.5.

4.3 FIRE GROWTH AND PROPAGATION

4.3.1 FIRE SPREAD WITHIN A ZONE

The fire PRA methodology used in the DCPRA-1988 and for the IPEEE did not explicitly model fire growth, size, duration, or spread. However, the methodology models fire scenario frequencies using geometry, severity, and propagation factors that implicitly account for such factors. These factors are defined in Section 4.1.4.1.

It is important to distinguish differences in assumptions and modeling regarding fire growth within a fire zone between the different portions of the analysis. In the spatial interactions analysis for localized scenarios (scenarios confined to one fire zone) it was conservatively assumed that a fire could disable all PRA equipment and cables within a fire area, independent of the severity of the given fire. This assumption is modeled by setting geometry and severity factors equal to 1.0. In the fire risk assessment, layout and arrangement drawings of plant components and cables were examined, and the appropriate geometry and severity factors for scenario frequency quantification were developed based on expert judgement. This approach was approved by the NRC review comments in SSER-34:

> "The geometric and severity factors were obtained by making use of the analyst's engineering judgement, and did not make use of calculations with fire propagation codes. However, an experienced fire analyst can make judgements concerning geometric and severity factors with adequate accuracy."

For the control room fire scenarios, a specialized severity curve was developed to describe the conditional frequency of an electrical panel fire in the control room propagating a given distance or greater. The DCPRA control room fire severity curve was developed by PLG. To develop the control room severity curve, the updated industry fire database was first searched for fires that occurred in control rooms. Four control room fires were found. To improve the breadth of the database, the other fires in the database were examined to determine if any could occur in the control room, giving consideration to similar kinds of equipment, sources, and manning levels. Several such fires were identified and included; these were primarily electrical panel fires. Based on a review of these fire events, estimates of the area involved in each fire were made. An equivalent radius was then calculated, and these data were used to plot the severity factor curve. The severity factor curve is presented as Figure 4.3-1. This severity factor curve was used to analyze control room fires.

Scenario specific combined geometry and severity factors were obtained using the method described below. The combined geometry and severity factor for all locations on a control board is obtained from an integration:

$$F_{G,S} = \int_A F_s(r)\, F_g(r)\, dA \qquad \text{(Eq. 4.3-1)}$$

where A is the area behind the panel from which a fire can affect the equipment specified in the fire scenario, $F_g(r)$ is the geometry factor expressed in terms of conditional frequency of fire occurrence per unit area, and $F_s(r)$ is the conditional frequency of fire being so severe that an area with radius r is impacted. For $F_s(r)$, the graph in Figure 4.3-1 is used, which defines $F_s(r)$ as a function of the radial distance from the fire.

To simplify the computations, $F_s(r)$ is discretized, and the integral is written as

$$F_{G,S} = \sum F_{si} F_{gi} \Delta a_i \qquad \text{(Eq. 4.3-2)}$$

and for $F_g(r)$, the function used is

$$F_g(r) = 1/A \qquad \text{(Eq. 4.3-3)}$$

where A is the total panel area of the control room.

$$F_{G,S} = \sum F_{si} \cdot (\Delta a_i/A) = (1/A) \sum F_{si} \Delta a_i \qquad \text{(Eq. 4.3-4)}$$

The integral of the combined geometry and severity factor is performed by taking the summation of the products of conditional fire occurrence probability in each postulated panel area and the associated severity factor that covers the relevant control circuits for a given scenario.

As an example, the combined geometry and severity factor for scenario VB-1 is obtained by following Equation 4.3-4 with some simplifications. This integration is performed by dividing the possible fire area into 1-foot-wide strips, then combining the geometry and severity factors for each strip, and summing the contributions over the selected panel area. The geometry factor for each strip is the ratio of a 1-foot-wide area on a given control board to the total linear width of the Unit 1 main control board panels (control console plus vertical boards).

For the cable spreading room fire scenarios, conservative geometry factors were used due to the uncertainty in cable routing inside this room. For Cable Spreading Room Scenario One, CSR1, the affected systems are ASW and CCW. Expert judgement, based on the evaluation of available drawings combined with review of detailed drawings from other power plants, concluded that less than 15 percent of the total cable spreading room floor area would be close to a critical set of cables of these two systems. Cable Spreading Room Scenario Two, CSR2, postulates a fire damaging a critical set of cables that affects the PORVs and the auxiliary relays of the pressurizer pressure and temperature controls. The geometry factor for cable spreading room scenario 2 was similarly determined to be 0.25.

The severity factor for CSR1 (the conditional frequency of fire at the critical locations of sufficient severity to cause CSR1) was judged to be no greater than 0.5. The same severity factor was also applied to CSR2.

4.3.2 FIRE SPREAD ACROSS ZONES

The fire PRA methodology discussed in Section 4.1 uses the scenario approach in both the spatial interactions analysis and the fire risk assessment. A major assumption common to both phases of the analysis is that Appendix R barriers will function as intended to prevent the spread of fire across fire zones. However, the scenario approach postulates propagation scenarios that serve to model the impact of a fire door failing to close or a fire damper failing to close. For example, Table 4.3-1 illustrates the propagation scenarios generated from one fire area. Some propagation scenarios propagate to multiple parallel fire zones. Some propagation scenarios propagate across an intermediate zone(s) to a third zone(s).

Section 4.1.3 discusses the assessment of propagation scenarios, including a table (see Table 4.1-4) of typical propagation factors, as well as typical geometry and severity factors used in the propagation scenarios. It is important to note that in the propagation scenarios, the target to which the geometry factor applies is the propagation path itself. The severity factor used in the propagation scenarios applies to target equipment in both zones.

4.3.3 SEVERE TURBINE BUILDING FIRE ANALYSIS

In the DCPRA-1988 fire analysis, a propagation fire scenario (14-D-FS-3) was postulated, originating on the turbine deck and spreading through ventilation exhaust ducts to all three 4-kV vital switchgear rooms (13-A, 13-B, and 13-C). In the fire risk assessment, the evaluation of this scenario assumed a propagation factor of 1.0, a geometry factor of 0.05, and a severity factor of 0.05. This scenario was binned for event tree quantification into initiator FS8, delayed loss of all three 4-kV vital buses F, G, and H. The NRC review questioned the likelihood of this scenario. Additionally, several severe fires have occurred in turbine buildings in plants outside of the United States. In light of these fires, this scenario has been reviewed in considerable detail and reassessed.

As a result of the ignition frequency update, turbine deck fire ignition frequency increased from 1.78E-3 per year to 2.191E-2 per year; i.e., see Table 4.1-3.

Each of the ventilation exhaust ducts is equipped with a backdraft fire damper. The fire dampers are heat activated and gravity operated. For the FS8 initiator to occur, all three backdraft dampers must fail. The following failure rate was used in the internal events PRA analysis for common cause failure to close two of two backdraft dampers:

D2DBDD	2 of 2	1.94E-5.

This value serves as a conservative estimate with respect to the common cause failure of three of three backdraft dampers. This failure rate represents an appropriate propagation factor for this scenario. For comparison, NUREG/CR-4550 provides a failure rate for a single damper of 2.7E-3.

Using the updated fire ignition frequency (2.191E-2), the geometry and severity factors of 0.05 each, and the estimated propagation factor due to failure of multiple backdraft dampers to close (1.94E-5), results in an initiator frequency for FS8 of 1.063E-9.

The generator excitor end of the turbine generator (a likely source of a severe fire on the turbine operating deck) is over 90 feet from the nearest exhaust ventilation duct. The exhaust ducts are raised above the operating deck such that the base of each ducts acts as a curb. There is also an open stairway between the generator and the ducts. Because of the distance, curbing, and interspersed flow paths, a flowing oil spill is not seen as a credible hazard. It is estimated that the only credible turbine deck fire hazard for impacting the 4-kV switchgear would be a severe fire such as a generator hydrogen seal oil fire. Several severe turbine building fires have occurred at international plants. A review of international turbine building severe fires produced the following five fire incidents:

Muhleberg	(July 1971)
Maanshan-1	(July 1985)
Vandelos-1	(October 1989)
Chernobyl-2	(October 1991)
Narora-1	(March 1993)

These five severe turbine building fires occurred in approximately 7,000 reactor-years of international operating experience. This represents an estimated severe turbine deck fire ignition frequency of 7.221E-4 per year. Assuming no credit is taken for any geometry effects, use of this severe fire ignition frequency combined with the damper-based propagation factor above, results in an FS8 initiator frequency of 1.40E-8. This alternative calculation is judged to confirm the previous calculation (1.063E-9), in concluding that the likelihood of the turbine building fires leading to failure of all three vital switchgear trains is extremely small.

4.3.4 FIRE DETECTION AND SUPPRESSION

In the internal fire analysis, credit for detection and suppression was assumed in the analysis of control room fires only. No credit was assumed for detection and suppression in any scenarios in the spatial interactions analysis. As a conservative measure, in the fire risk assessment, the non-suppression factor was taken to be 1.0, except in the control room. Also, no credit for detection and suppression was explicitly assumed in the cable spreading room fire analysis.

Quantification of the control room fire scenarios includes the integrated geometry and severity factors discussed in Section 4.1.4.2. The control room fire severity curve was developed from a review of control room fire events and "control room-like" fire events. Estimates of the area involved in each fire were made and an equivalent radius was calculated for each. Because suppression was applied in these actual fire events, suppression is not separable from the resultant severity curve. Thus, the combined geometry and severity factors for control room panel fires also implicitly model the likelihood of non-suppression.

The validity of applying this suppression assumption to control room fire analysis is supported by the following. The control room is continuously manned for each plant operation mode. There are 28 smoke detectors inside the electrical control panels of the main control room, and there are 4 room smoke detectors. Of the 28 control panel smoke detectors, 13 are located on the main control boards (vertical boards) and 3 are located on the control console. These detectors are ionization detectors that feed into several different annunciator systems; control board annunciator, "Fire/Smoke Detector," will provide both an audible alarm and a visual window alarm that informs the control room staff of the actuation of a smoke detector. Each of the 13 smoke detectors on the vertical boards also has a red light associated with it (located on top of the vertical boards) that will provide an indication to the operators of the exact location of the fire.

The control room does not have an automatic fire suppression system; however, within the control room, there are seven Halon fire extinguishes available for manual suppression. The procedures for fires (EP M-6, Reference 4-51) instruct the operators (immediate action) to change the ventilation system to Mode 2 to provide 100 percent outside makeup air during a control room fire. In addition, the control room Fire Fighting Preplan (of EP M-6) instructs the operators to establish additional portable ventilation, if necessary. These actions will ensure the greatest likelihood of maintaining control room habitability.

4.3.5 FIRE SUPPRESSION-INDUCED EQUIPMENT DAMAGE

Section 4.8.1.2 includes information on seismically induced actuation of fire suppression systems. Section 4.8.4.1 addresses spurious or inadvertent actuation of fire suppression systems. This section addresses the potential for additional equipment damage scenarios resulting from fire suppression actuation in response to a fire.

The IPEEE fire walkdown identified two areas as being potentially vulnerable to combinations of fire and fire suppression induced damage to equipment. These areas were the motor-driven AFW pump room and the centrifugal charging pump room. Water fire suppression system induced damage has been considered under several programs at PG&E including the SISIP the moderate energy line break analysis, and the PRA internal floods analysis portion of the IPE.

In addressing suppression-induced damage, the PRA internal floods analysis (Reference 4-20) concluded the following regarding fire water suppression systems:

> "At DCPP, sprinkler heads rather than deluge systems are used to protect safety-related equipment. The inadvertent actuation of a sprinkler head is unlikely and the consequences of actuation are minor."
>
> "...whereas the water sprinkler system only directly affects the area directly below the sprinkler head within a radius of approximately 10 feet. All equipment required for safe shutdown is protected with sprinklers, and as a general rule redundant safe shutdown equipment is separated by compartments and is no closer than 20 feet to each other. Finally, all

rooms with safety related equipment are built with redundancy and in such a way that initiation of one sprinkler head or fire water pipe leak/break will not affect the redundant component."

The only flood scenario identified which could be initiated by actuation of a wet pipe sprinkler fire suppression system is one involving spray damage to both motor-driven AFW pumps.

The moderate energy line break analysis has been subsumed into the high energy line break program. The moderate energy line break reports (Reference 4-22) reached the following conclusions for the AFW pump motors and the centrifugal charging pump motors:

> "The Auxiliary Feedwater (AFW) System's three pumps and two tanks require no protection. The motor driven pumps have moderate energy lines in the vicinity that pose hazards. But considering the maximum expansion of the half angle of ten degrees, no two pumps motors will be impinged simultaneously. ...it is assumed that all the equipment in the AFW motor driven pump room are wetted, but these motors have open dripproof casings which enable them to withstand a 100% humid, dripping wet environment. Direct impingement by a liquid into the air vents is the only concern for the motor internals (e.g., stator and field coils).
>
> "The two centrifugal charging pumps won't be impinged simultaneously by the only Moderate Energy line in the room which is the overhead firewater line."

Suppression-induced damage in response to a fire represents an additional failure mechanism not quantified by this fire PRA methodology. Based on the conclusions of the internal floods analysis and the moderate energy line break analysis, fire water suppression induced damage is not considered to be a likely equipment damage mechanism. Nonetheless, a sensitivity study is presented herein to quantitatively bound the core damage impact of fire water suppression-induced damage.

The geometry factor and severity factor used to quantify a specific fire scenario serve to estimate the fraction of fires occurring in a fire zone that results in the postulated equipment damage. Conservatively assuming that any fire in the fire zone leads to the postulated equipment damage from either the fire itself or any suppression induced damage, a bounding estimate of the frequency of potential suppression induced damage is obtainable from the difference between the product of the geometry and severity factor and 1.0.

A fire in the motor-driven AFW pump room was modeled by fire scenario 3-Q-2-FS-1. The scenario quantification assumed a geometry factor of 0.6 and a severity factor of 0.5 applied to a fire ignition frequency of 3.879E-4 per year. Thus, the scenario quantification predicts that 30 percent of fires in the fire zone result in the postulated fire-induced equipment damage. This scenario contributes to the initiator FS1, which has a conditional

probability of core damage given by 3.5126E-3. Assuming that the remaining 70 percent of fires in the fire zone lead to the postulated equipment damage caused by either fire or fire water suppression, the core damage frequency would increase 9.5E-7.

A fire in the centrifugal charging pump room was modelled by fire scenario 3-H-1-FS-1. The scenario quantification assumed a geometry factor of 1.0 and a severity factor of 1.0. Thus, all fire ignition in this area is assumed to result in the postulated equipment damage. Any potential for suppression induced damage is already captured in the quantification.

Table 4.3-1. Example Propagation Scenarios from Fire Zone 3-H-1				
Scenario	Geometry Factor	Severity Factor	Propagation Factor	Propagates to Fire Zone
3-H-1-FS-2	0.5	0.1	0.1	3-C
3-H-1-FS-3	0.05	0.05	0.1	3-B-1
3-H-1-FS-4	0.05	0.05	0.1	3-B-2
3-H-1-FS-5	0.05	0.05	0.1	3-B-3
3-H-1-FS-6	0.5	0.1	0.1	3-H-2
3-H-1-FS-7	0.5	0.05	0.1	3-J-3

Figure 4.3-1. Severity Factor (F_s) For Control Room Fires

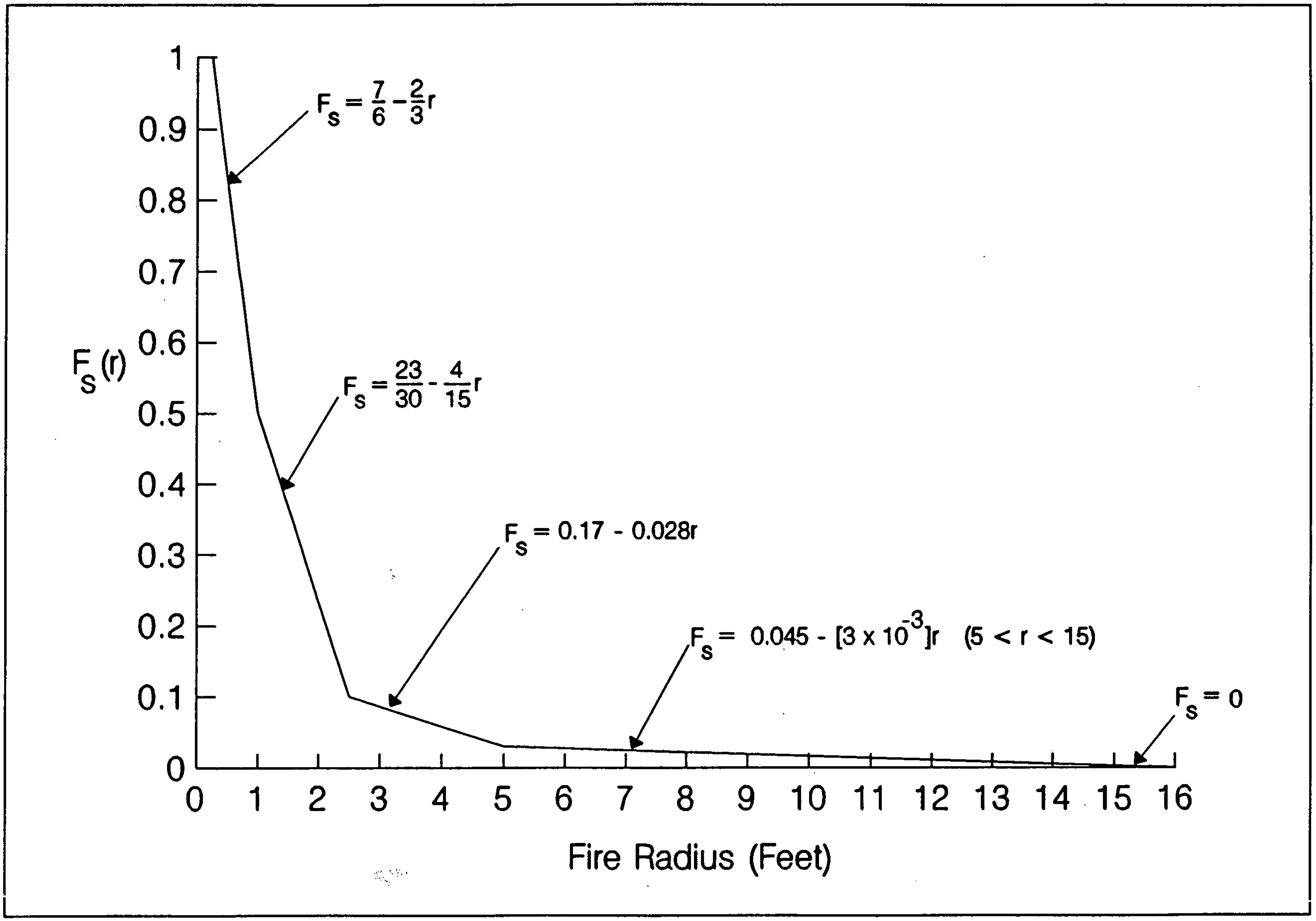


Figure 4.3-2. Configuration Used In Calculating the Combined Geometry and Severity Factor for Scenarios VB-2A, VB-2B, and VB-2/3

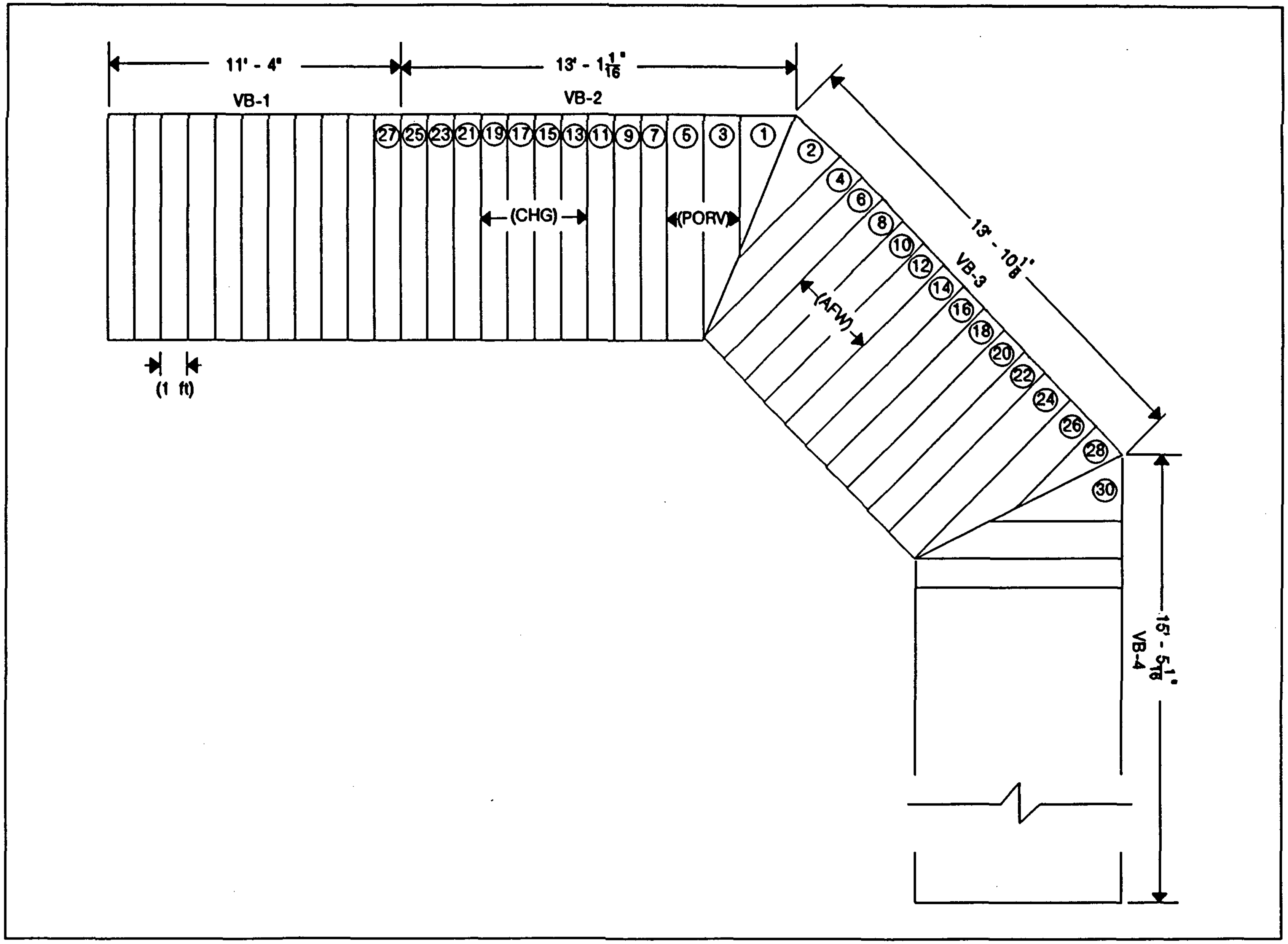


Figure 4.3-3. Configuration Used in Calculating the Combined Geometry/Severity Factor for Scenario VB-1

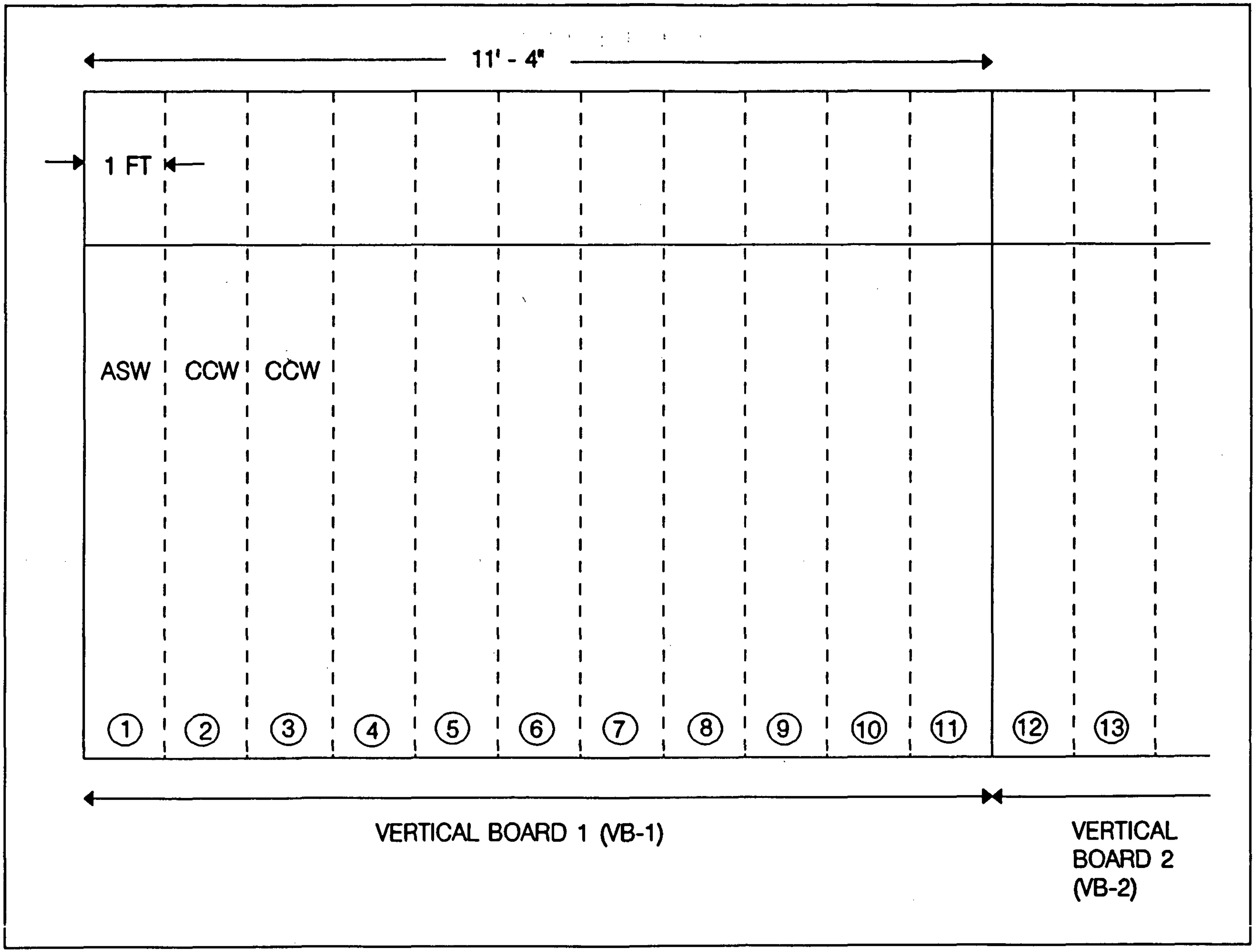
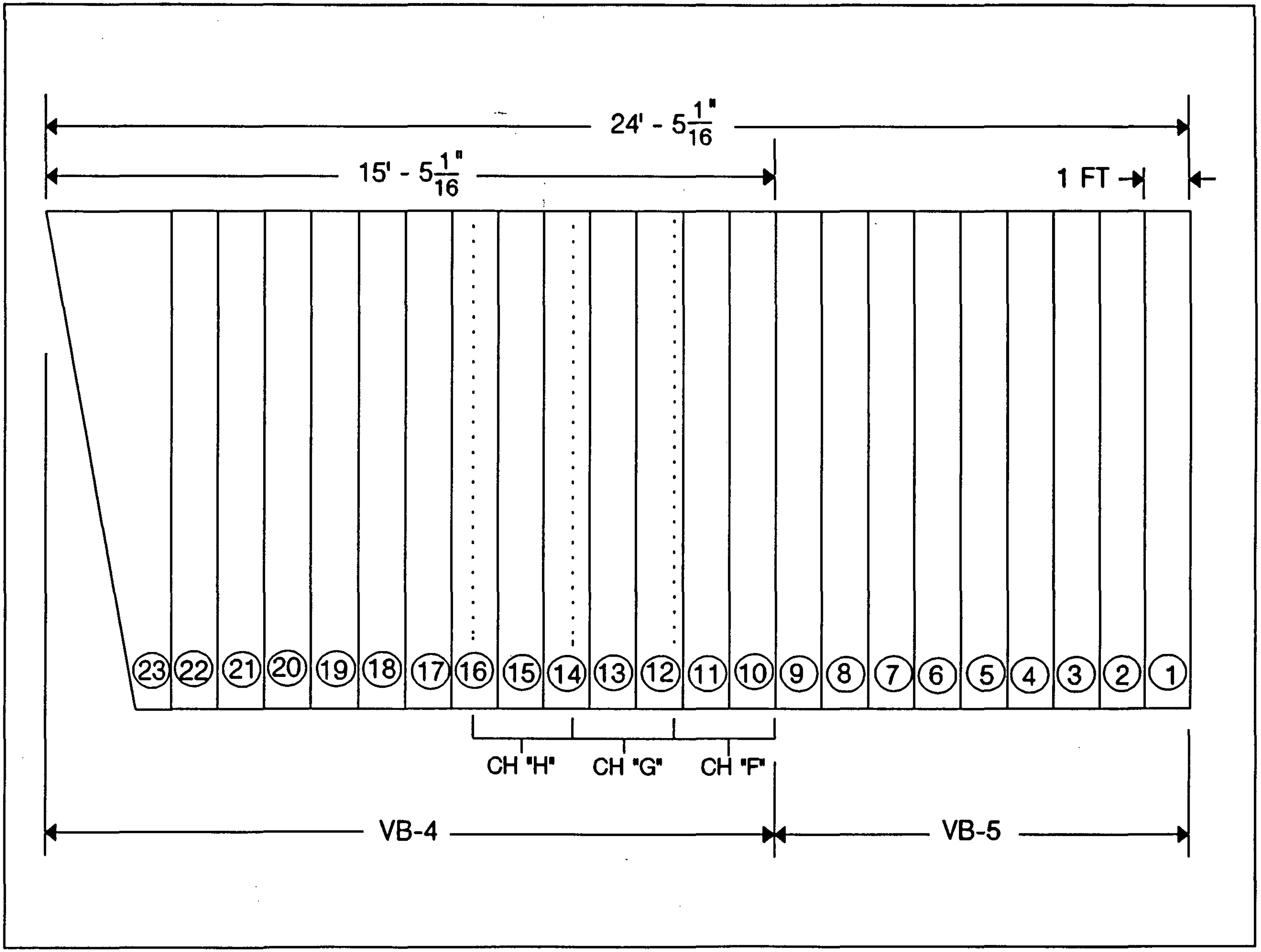


Figure 4.3-4. Configuration Used in Calculating the Combined Geometry/Severity Factor for Scenario VB-4



4.4 EVALUATION OF COMPONENT FRAGILITIES AND FAILURE MODES

4.4.1 COMPONENT FRAGILITIES

In the spatial interactions analysis, all equipment in the affected fire zone is assumed to be disabled. In the fire risk assessment, in general, equipment impacted by a fire is assumed to be disabled. Therefore, component fragility is not explicitly modeled. The likelihood of equipment being impacted by fires is modeled using the geometry factor and severity factor modifiers to the zone ignition frequency. There are some scenarios where it is postulated that regardless of the location of the origin of the fire, smoke will damage electrical switchgear in all propagation areas if an open propagation pathway exists. The assessment models such scenarios with a geometric factor of 1.0.

In the control room and cable spreading room analyses, the equipment damage in each scenario is determined by the scenario definition. The likelihood of equipment being impacted is reflected by the integrated geometry and severity factors in the control room fire scenarios and similarly, by the geometry and severity conditional frequencies in the cable spreading room scenarios.

4.4.2 EQUIPMENT DAMAGE BINS AND FIRE INITIATORS

4.4.2.1 Plant Response Fire Initiators

The product of the spatial interactions analysis and fire risk assessment is a set of significant frequency fire scenarios that could lead to combinations of equipment damage which are judged to be of significant importance to overall plant risk. The significant scenarios are binned into fire initiators according to the type of equipment damage the scenarios produce. There are eight initiators associated with the general plant internal fire analysis. The sum of the scenario frequencies serves as the initiator frequency for event tree quantification. Table 4.4-1 lists the equipment damage bins with the initiator designation and the contributing fire scenarios.

The fire scenario assessments of the significant contributors resulting from the spatial interactions analysis and fire risk assessment are detailed in Table 4.4-2. The geometry factors, severity factors, propagation factors, and other values used in the assessment of each scenario are included in the table.

4.4.2.2 Control Room Fire Initiators

Based on the review and identification of critical control boards inside the main control room, four fire scenarios were established as being representative of the most hazardous of all fire scenarios occurring in the control room. All other fires have less impact on plant equipment, require greater severity to inflict the same damage, or have a low frequency of occurrence. The four fire scenarios analyzed are as follows:

Scenario VB-1. A fire that affects ASW and/or CCW control circuitry in board VB-1.

Scenario VB-2. A fire that affects the control circuitry of the PORVs and charging pumps in VB-2. This scenario was divided into two subscenarios: one that affects only the PORVs, and one that affects both the PORVs and the charging pumps.

Scenario VB-2/3. A fire at the interface of boards VB-2 and VB-3, affecting PORVs and the AFW system control.

Scenario VB-4. A fire that affects 4-kV buses F, G, and H in board VB-4.

The control room fire ignition frequency serves as the initiating event frequency for all control room fire scenarios. Several control room fire event trees were developed to quantify these fire scenarios (see Section 4.6.1). In each of these event trees, the integrated geometry and severity factors discussed in Section 4.3.1 are applied as split fractions for top event FEF.

The control room fire initiators and the results of the integration of geometry factors and severity factors are tabulated in Table 4.4-3.

4.4.2.3 Cable Spreading Room Fire Initiators

Two fire scenarios were identified as being representative and the most important of all fire scenarios affecting the cable spreading room. All other fires either have the same impact but have to be more severe to inflict damage or fail a more limited amount of equipment. The two fire scenarios analyzed are:

Cable Spreading Room Fire Scenario One (CSR1). A fire affects both the ASW and the CCW system controls.

Cable Spreading Room Fire Scenario Two (CSR2). A fire affects the PORVs and pressurizer instrumentation.

For scenario CSR1, the sequence of events is a cable spreading room fire leading to the failure of the ASW and CCW systems.

For scenario CSR2, the sequence of events is a cable spreading room fire that damages the control circuitry of the PORVs. Other plant equipment that might be affected by this scenario includes the auxiliary relays of the pressurizer pressure control. As was shown by the schematic drawing of the PORVs control, the relays can open the power-operated relief valves at the preset level, and are located at the rack nuclear auxiliary safeguard cubicle A (RNASA) inside the cable spreading room. Fire-induced energization of the relays could generate a spurious signal to open the PORVs.

Each of the cable spreading room fire scenarios could be initiated by a fire occurring either in the region directly below the respective vertical board or in another part of the cable spreading room that could lead to the hypothesized failure. Due to the uncertainty

in cable routing inside this room, conservative geometry factors are used in the analysis of each of the cable spreading room fire scenarios.

The cable spreading room fire ignition frequency serves as the initiating event frequency for both cable spreading room fire scenarios. The cable spreading room event tree includes the combined geometry and severity factors appropriate to each scenario in top event FEF. The recovery actions discussed in Section 4.4.3.3 for each scenario are quantified using top event FRE.

The cable spreading room fire initiators and the geometry factors and severity factors used in the analysis are tabulated in Table 4.4-4.

4.4.3 RECOVERY ACTIONS AND HUMAN INTERVENTION

4.4.3.1 Recovery in Plant Response Event Trees

Recovery actions are incorporated into the plant response model by defining a recovery action event tree with top event RE for electric power recovery. The split fraction values for the top event RE are the non-recovery probabilities (the likelihood that the recovery action is unsuccessful) associated with the recovery actions. Each event sequence entering the recovery action tree is evaluated to determine whether recovery is applicable. Electric power recovery is not assumed for any fire initiators. A separate top event, RA, is also included in the recovery action event tree to model the effect of reassignment of top event split fractions for the ASW system top event, AS. Some fire initiators do credit recovery of ASW.

The ASW split fractions assigned in the mechanical support tree do not take credit for recovery of ASW for the following types of initiating events: LOCAs, possible transient-induced LOCAs, and steamline breaks. These events add heat to the containment and, consequently, increase the heat load on the CCW system. The heatup of the CCW system limits the time available to crosstie to Unit 2 ASW before exceeding equipment temperature limits. Other initiating events take credit for the Unit 2 ASW recovery.

A loss of AC power or DC power on the ASW electric power trains will guarantee failure of the ASW system in the mechanical support event tree. There are sequences where this assumption is overly conservative. Where appropriate, top event RA serves to credit recovery of ASW for fire initiators in one of two ways. First, opening the inter-unit crosstie valve allows Unit 2 ASW pumps to supply the Unit 1 CCW heat exchangers. This recovery method is modelled with split fraction RA4. Second, for sequences where the electrical support failure on the F bus is restricted to DC control power, credit is taken for the already running Unit 1 F bus ASW pump 1-1, and split fraction RAD serves to quantify this form of ASW recovery.

No ASW crosstie recovery is assumed for those fire initiators that would also make Unit 1 CCW unavailable. These scenarios include FS3 (loss of CCW) and FS8 (loss of 4-kV buses F, G, and H).

The values for the RA split fraction are computed from the sum of an ASW system failure frequency and a CCW system failure frequency. The CCW system model incorporates a human action to reduce heat loads on the CCW system (ZHECC1). The effect of the human action is to reduce the success criteria for the number of CCW pumps necessary from 2 of 3 to 1 of 3.

4.4.3.2 Recovery in Control Room Fire Scenarios

One important element of control room fires is the operator response to putting out the fire and bringing the plant to a stable shutdown status. Many different operator-related scenarios can be envisioned. In general, the detection time will be short, given the large number of smoke detectors within the vertical boards and the type of annunciation provided. The operators' immediate response will be to extinguish the fire and follow the appropriate fire response procedures. In addition, the operators will have to respond to equipment failures caused by the fire; this may involve restoring control to the affected equipment from outside the control room (at the hot shutdown panel, the dedicated shutdown panel, the 4-kV switchgear rooms, 480 V switchgear rooms, or at specific equipment locations).

Only in extreme cases would control room evacuation be required; before evacuating the control room, the operators would attempt all means to extinguish the fire and provide adequate ventilation. If necessary, use of self-contained breathing apparatus would also be considered. If control room evacuation is necessary, the operators would follow the Abnormal Operating Procedure OP AP-8A (Control Room Inaccessibility) and establish control of the plant from locations outside the control room.

To quantify the conditional frequency of failure of the operator recovery action, the following parameters are considered in the evaluation:

- Whether the control room must be evacuated.
- If evacuation is required, whether recovery can be accomplished prior to evacuation.
- The available time window to accomplish the designated operator task.
- The indications available to the operators.
- The stress level of the operators.
- The procedural guidance as it relates to the required mitigation actions.

The control room fire scenarios credit certain human actions and recovery actions. Each of these is modeled in the applicable control room fire scenario event tree.

Trip Reactor Coolant Pumps Prior to Seal Damage (FTP)

Top event FTP is only applicable to fire scenarios VB1 and VB4. The top event models the operator action to trip the RCPs within 10 minutes to prevent motor bearing damage after the loss of CCW flow. If the pumps are left running, shaft vibrations are assumed to develop when the bearings fail and cause damage to the pump seal assemblies, resulting in a small LOCA. The loss of CCW flow may be induced by failures in the CCW system (VB1) or by a loss of all emergency AC power (VB4). If the control room must be evacuated, no credit is taken for the action to trip the RCPs. This is because the control room evacuation procedures do not call for tripping the RCPs just before evacuation. Tripping RCP's prior to evacuation of the control room (with a backup action in the 12-kV switchgear room) is being evaluated as a potential enhancement to procedures (Reference 4-23). The event trees model this by assigning a guaranteed failure split fraction for this action after control room evacuation. If the control room remains habitable, plant abnormal and alarm procedures do instruct the operators to trip the RCPs. Human action ZHEF11 is used to quantify the split fraction for top event FTP under such conditions.

Recovery of ASW and CCW Prior to LOCA (FRE)

In control room fire scenarios VB1 and VB4, successfully tripping the RCPs increases the available timeframe for recovery actions prior to seal damage. Following success of top event FTP, the restoration of ASW/CCW in the longterm, before seal damage, is modeled as human action ZHEF12 in top event FRE. Credit for restoration of ASW/CCW from outside the control room prior to RCP seal damage is taken only if the control room remains habitable.

LOCA Termination (FLT)

Control room fire scenarios for event trees VB2 and VB23 include a fire-caused hot short leading to a stuck open PORV induced LOCA. Top event FLT models the operator action to terminate the LOCA by closing the PORV emergency close switch at the hot shutdown panel. This human action is quantified using ZHEF21 (VB2) and ZHEF31 (VB23).

Secondary Heat Removal Restored (FSH)

In addition to the PORV LOCA, control room fire scenario VB23 includes a failure of the auxiliary feedwater system. Recovery of the auxiliary feedwater includes starting the auxiliary feedwater pumps and controlling steam generator level from the hot shutdown panel. Electrically independent indications are available at the dedicated shutdown panel. This recovery action is modeled using human action ZHEF32 in top event FSH.

LOCA Mitigation (FML)

Each of the control room fire event trees concludes with the top event FML. If the operators do not isolate the open PORV from the hot shutdown panel, they may still mitigate the LOCA from the control room if it remains habitable. If the control room

remains habitable, action to mitigate a LOCA or to initiate feed and bleed cooling is considered.

No credit is given to mitigate a LOCA from outside the control room. To mitigate the LOCA from the control room, the operators must still restore cooling to the charging or safety injection pumps for high pressure injection and, eventually, restore CCW for cooling to the RHR heat exchangers for sump recirculation.

4.4.3.3 Recovery in Cable Spreading Room Fire Scenarios

One of the most important elements of a cable spreading room fire incident is the control room operator's response to the fire. In general, the recovery actions considered are similar to those for the control room fire scenario. However, there are significant differences between the two separate events. For example, the diagnosis of equipment failures due to a cable spreading room fire may be more difficult than that of a control room fire because spurious signals in instrumentation cables or control cables may lead to conflicting indications that might result in misperception of true plant status prior to identification of a fire. For cable spreading room fire scenario CSR1, the required operator actions are similar to those for the control room fire scenario VB-1 ; i.e., ZHEF51 and ZHEF52. Recovery actions for cable spreading room scenario CSR2 are similar to operator activities in control room fire scenario VB-2. ZHEF61 represents this required action.

To evaluate the operator recovery actions for cable spreading room fire scenario CSR1, it is assumed that fire of this magnitude would disable the control features of the ASW system and the CCW system from the main control room. Whenever necessary, the operators would supplement their actions to carry out the required shutdown procedures from the hot shutdown panel located at elevation 100'. Similar to the circumstances confronted by the operators in the control room fire scenario VB-1, the recovery actions include tripping the RCPs and restoring the RCP seal cooling source to prevent seal LOCA. ZHEF51 models the necessary steps to trip the RCPs before bearing failure causes subsequent seal damage. ZHEF52 models the operator actions to reestablish CCW and ASW flow from the hot shutdown panels.

To evaluate the operator recovery actions for cable spreading room fire scenario CSR2, it is assumed that fire of this magnitude would disable the operators from closing the PORV inside the main control room. Thereafter, whenever necessary, the operators would supplement their actions to carry out the required shutdown procedures from the hot shutdown panel located at elevation 100'. The circumstances facing the operators in this scenario are similar to those of control room fire scenario VB-2. The recovery actions include closure of the PORVs and are modelled by ZHEF61.

Table 4.4-1. Initiator Equipment Damage Bins				
Initiator	Frequency	Equipment Damage	Fire Scenario	Frequency
FS1	4.356E-4	Loss of Both Motor-Driven AFW Pumps	3-Q-2-FS-1	1.164E-4
			14-A-85-FS-1	1.008E-4
			14-A-104-FS-1	8.756E-5
			6-A-5-FS-1	6.888E-6
			S-3-FS-1	3.163E-5
			5-A-4-FS-1A	1.683E-6
			12-A-FS-1	8.838E-5
			5-A-4-FS-1B	2.244E-6
FS2	2.967E-3	Loss of All Charging Pumps	3-H-1-FS-1	1.523E-3
			3-C-FS-5	4.266E-4
			3-J-2-FS-1A	4.074E-4
			3-J-3-FS-1	5.734E-4
			3-AA-FS-1	3.668E-5
FS3	1.659E-6	Loss of All CCW Pumps	3-J-2-FS-1B	1.659E-6
FS4	1.832E-3	Loss of Control Room Ventilation Fans	8-B-3-FS-1	1.832E-3
FS5	1.174E-4	Loss of Both ASW Pumps	4-A-FS-1B	2.772E-5
			4-B-FS-1	6.517E-5
			14-E-FS-1	2.455E-5
FS6	6.933E-5	Loss of Buses F and G	4-A-FS-1A	6.930E-6
			5-A-1-FS-3	8.909E-7
			5-A-2-FS-3	8.909E-7
			12-A-FS-2	7.070E-7
			12-B-FS-2	7.070E-7
			13-A-FS-3	2.960E-5
			13-B-FS-2	2.960E-5
FS7	6.460E-5	Loss of Buses G & H	5-A-2-FS-4	8.909E-7
			5-A-3-FS-3	8.909E-7
			12-B-FS-3	7.070E-7
			12-C-FS-2	7.070E-7
			13-C-FS-2	3.180E-5
			13-B-FS-3	2.960E-5
FS8	1.401E-8	Delayed Failure of Buses F, G, and H	14-D-FS-3	1.401E-8

Table 4.4-2. Fire Scenarios Contributing to Fire Initiators										
Initiator	Fire Scenario	Ignition Freq	Geometry Factor	Severity Factor	Propagation Factor	Note	Scenario Freq	Equip	Prop To	Equip
FS1	3-Q-2-FS-1	3.879E-4	0.6	0.5	-		1.164E-4	2 MDAFW Pumps	-	
FS1	14-A-85-FS-1	1.512E-2	0.02	1/3	-	a	1.008E-4	2 MDAFW Pumps	-	
FS1	14-A-104-FS-1	8.756E-3	0.02	0.5	-	a	8.756E-5	2 MDAFW Pumps	-	
FS1	6-A-5-FS-1	2.375E-4	0.5 (0.1)	0.05 (1)	-	b	6.888E-6	Loss of 2 MDAFW Pumps or LCV.	-	
FS1	S-3-FS-1	1.265E-4	.25	1.0	-		3.163E-5	Loss of 2 MDAFW Pumps.	-	
FS1	5-A-4-FS-1A	1.122E-3	0.03	0.05	-	-	1.683E-6	Loss of LCVs for 2 MDAFW Pumps	-	
FS1	12-A-FS-1	7.07E-4	0.25	0.5	-	a	8.838E-5	Fail Auto Start for 2 MDAFW Pumps	-	
FS1	5-A-4-FS-1B	1.122E-3	0.05	0.04	-		2.244E-6	Loss of 2 MDAFW Pumps.		
FS2	3-H-1-FS-1	1.523E-3	1.0	1.0	-		1.523e-3	Loss of All Charging Pumps	-	

Table 4.4-2. Fire Scenarios Contributing to Fire Initiators										
Initiator	Fire Scenario	Ignition Freq	Geometry Factor	Severity Factor	Propagation Factor	Note	Scenario Freq	Equip	Prop To	Equip
FS2	3-C-FS-5	2.133E-3	1.0	0.2	1.0		4.266E-4	-	3-J-1 3-J-2(*) 3-J-3(*) 3-F	(*) Loss of all Charging Pumps
FS2	3-J-2-FS-1A	4.047E-4	1.0	1.0	-		4.047E-4	Loss of All Charging Pumps	-	
FS2	3-J-3-FS-1	5.734E-4	1.0	1.0	-		5.734E-4	Loss of All Charging Pumps	-	
FS2	3-AA-FS-1	2.934E-3	0.05	0.25	-		3.668E-5	Loss of Charging Pump Suction Valves.	-	
FS3	3-J-2-FS-1B	4.047E-4	0.5	1.0	-	c	1.659E-6	Loss of CCW Pumps 1 and 2 Plus 4-kV Breaker Trip	-	
FS4	8-B-3-FS-1	1.832E-3	1.0	1.0	-		1.832E-3	Loss of All CR Ventilation Fans	-	
FS5	4-A-FS-1B	1.386E-3	0.02	1.0	-		2.772E-5	Loss of 2 ASW Pumps	-	

Table 4.4-2. Fire Scenarios Contributing to Fire Initiators										
Initiator	Fire Scenario	Ignition Freq	Geometry Factor	Severity Factor	Propagation Factor	Note	Scenario Freq	Equip	Prop To	Equip
FS5	4-B-FS-1	9.31E-4	0.07	1.0	-		6.517E-5	ASW and CCW (circuits for valves FCV-602, 603)	-	
FS5	14-E-FS-1	9.818E-4	0.25	0.1	-	a	2.455e-5	ASW and CCW (circuits for valves FCV-602, 603)	-	
FS6	4-A-FS-1A	1.386E-3	0.05	0.1	-		6.930e-6	Loss of AFW and Bus F and G	-	
FS6	5-A-1-FS-3	7.127E-4	0.25	0.1	.05		8.909e-7	4-kV Bus F	5-A-2	4-kV Bus G
FS6	5-A-2-FS-3	7.127E-4	0.25	0.1	.05		8.909e-7	4-kV Bus G	5-A-1	4-kV Bus F
FS6	12-A-FS-2	7.07E-4	0.1	0.1	0.1	a	7.070e-7	4-kV Bus F	12-B	4-kV Bus G
FS6	12-B-FS-2	7.07E-4	0.1	0.1	0.1	a	7.070e-7	4-kV Bus G	12-A	4-kV Bus F
FS6	13-A-FS-3	1.48E-3	1.0	0.2	0.1	a	2.960e-5	4-kV Bus F	13-B	4-kV Bus G
FS6	13-B-FS-2	1.48E-3	1.0	0.2	0.1	a	2.960e-5	4-kV Bus G	13-A	4-kV Bus F
FS7	5-A-2-FS-4	7.127E-4	.25	.1	.05		8.909e-7	4-kV Bus G	5-A-3	4-kV Bus H
FS7	5-A-3-FS-3	7.127E-4	.25	.1	.05		8.909E-7	4-kV Bus H	5-A-2	4-kV Bus G

Table 4.4-2. Fire Scenarios Contributing to Fire Initiators										
Initiator	Fire Scenario	Ignition Freq	Geometry Factor	Severity Factor	Propagation Factor	Note	Scenario Freq	Equip	Prop To	Equip
FS7	12-B-FS-3	7.070E-4	0.1	0.1	0.1	a	7.070E-7	4-kV Bus G	12-C	4-kV Bus H
FS7	12-C-FS-2	7.070E-4	0.1	0.1	0.1	a	7.070E-7	4-kV Bus H	12-B	4-kV Bus G
FS7	13-C-FS-2	1.59E-3	1.0	0.2	0.1	a	3.180E-5	4-kV Bus H	13-B	4-kV Bus G
FS7	13-B-FS-3	1.48E-3	1.0	0.2	0.1	a	2.960E-5	4-kV Bus G	13-C	4-kV Bus H
FS8	14-D-FS-3	2.191E-2	-	-	-	a d	1.401e-8	-	13-A, 13-B, 13-C, 12-A, 12-B, 12-C	4-kV Bus F 4-kV Bus G 4-kV Bus H

NOTES:

(a) IPEEE ignition frequency was increased from the DCPRA-1988 value for these scenarios.

(b) 6-A-5-FS-1 This scenario initiator frequency calculation includes separate geometry factors and severity factors for two subscenarios; one modeling the impact on the pump power cables, and the other models the impact on the cables for the discharge LCVs. LCV failure includes the probability for the fail open LCVs to remain closed (0.04).

$$(0.5 * 0.05 + 0.1 * 1.0 * 0.04) = 0.029$$

(c) 3-J-2-FS-1B This subscenario frequency calculation includes as a multiplier, a human action failure rate (ZHECC1 = 8.2E-3) to start the standby CCW pump, or reduce loads on the CCW system such that 1 CCW pump alone will meet the system success criteria.

(d) 14-D-FS-3 Analysis of this fire scenario is detailed in Section 4.3.3.

Table 4.4-3. Control Room Fire Initiators			
Initiator	Initiator Frequency	Equipment Damage	Combined Geometry and Severity Factor
VB1	4.90E-3	Failure of Auxiliary Saltwater and Component Cooling Water Systems	2.46E-2
VB2A	4.90E-3	Hot Short causes PORV to stick open	4.4E-2
VB2B	4.90E-3	Stuck Open PORV combined with failure of Charging Pumps	2.2E-3
VB23	4.90E-3	Stuck Open PORV combined with Loss of Auxiliary Feedwater System	5.54E-3
VB4	4.90E-3	Loss of 4-kV buses F, G, and H; Loss of Auxiliary Saltwater and Component Cooling Water Systems.	8.82E-3

Table 4.4-4. Cable Spreading Room Fire Initiators				
Initiator	Initiator Frequency	Equipment Damage	Geometry Factor	Severity Factor
CSR1	6.70E-3	Loss of Auxiliary Saltwater and Component Cooling Water Systems	0.164	0.5
CSR2	6.70E-3	PORV Control Circuitry Causes Stuck Open PORV	0.276	0.5

4.5 FIRE DETECTION AND SUPPRESSION

4.5.1 FIRE DETECTION AND SUPPRESSION IN THE PLANT

4.5.1.1 Fire Fighting Preplans

Emergency Procedure EP M-6 contains the Fire Fighting Preplan for each fire area in the plant. Fire Fighting Preplans are unique plant and site layout drawings that have been specifically prepared for use as a quick reference guide when combatting a fire, medical or hazardous material emergency. The preplans include a standard set of symbols to locate specific items such as hazards, phones, emergency equipment, command post locations, primary and secondary access routes, and the types of both fixed and portable fire fighting equipment. Included with each drawing is a written description containing the following information:

- Potential Combustibles
- Most Probable Fires
- Access and Egress Routes
- Fire Brigade Staging Area
- Hazardous Materials
- Management of Plant Systems
- Recommendation for Protection of Heat Sensitive Equipment
- Fire Suppression Equipment
- Ventilation
- Communications
- Lighting
- Safety Equipment
- Special Precautions

The preplans include Materials Safety Data Sheets (MSDS) in a separate section. The MSDS that have been identified for immediate reference are the major chemicals that are in use at DCPP on a daily basis.

In addition, the preplans include a section that describe special considerations and actions that should be carried out or considered by the fire brigade for several types of fires at Diablo Canyon, including:

- Wildland Fires
- Turbine Lube Oil Fires
- Electrical Cable Fires
- Energized Electrical Equipment Fires
- Radiological Fires
- Flammable Gas/Liquid Fires

4.5.1.2 Manual Fire Fighting Effectiveness

One of the Fire Risk Scoping Study issues is related to manual fire fighting effectiveness. This Fire Risk Scoping Study is detailed in Section 4.8.3. The Diablo Canyon Fire Protection Program was reviewed against the EPRI "Attributes of Adequate Fire Protection Program" (see Table 4.8-1) from Attachment 10.5 of the EPRI FIVE Methodology Final Report (Reference 4-24). Reviews were independently performed by both a PRA analyst and the Diablo Canyon Fire Marshall. The reviews concluded that the Diablo Canyon Fire Protection Program met all of the attributes of an adequate fire protection program as defined in Section III of Table 4.8-1. The program aspects under review included the composition and training of the fire brigade and the adequacy of the fire brigade equipment inventory.

4.5.1.3 Fire Detection and Suppression in the Control Room

The control room is continuously manned for each plant operation mode. There are 28 smoke detectors inside the electrical control panels of the main control room, and there are 4 room smoke detectors. Of the 28 control panel smoke detectors, 13 are located on the main control boards (vertical boards) and 3 are located on the control console. These detectors are ionization detectors that feed into several different annunciator systems; control board annunciator, "Fire/Smoke Detector," will provide both an audible alarm and a visual window alarm that informs the control room staff of the actuation of a smoke detector. Each of the 13 smoke detectors on the vertical boards also has a red light associated with it (located on top of the vertical boards) that will provide an indication to the operators of the exact location of the fire.

The control room does not have an automatic fire suppression system; however, within the control room, there are seven Halon fire extinguishes available for manual suppression. The fire response procedure (EP M-6) instructs the operators (immediate action) to change the ventilation system to Mode 2 to provide 100 percent outside makeup air during a control room fire. In addition, the control room Fire Fighting Preplan (of EP M-6) instructs the operators to establish additional portable ventilation, if necessary. These actions will ensure the greatest likelihood of maintaining control room habitability.

4.5.1.4 Fire Detection and Suppression in the Cable Spreading Room

There are 15 smoke detectors installed throughout the cable spreading room. The automatic fire suppression feature includes a heat-actuated total flooding CO_2 system, which can also be manually activated from the control room or from immediately outside of the cable spreading room. The manual suppression capability consists of a fire water hose reel and portable extinguishes within the room or in adjacent fire zones at the same elevation.

All Class IE circuits within the cable spreading room are routed in steel conduits or in trays totally enclosed by steel covers. Separation of the enclosed raceway meets the criteria of Institute of Electrical and Electronics Engineers Standard 384 (Reference 4-25).

4.5.2 FIRE DETECTION AND SUPPRESSION IN THE MODEL

As previously discussed in Section 4.3.4, very little credit is taken for fire detection and suppression in modeling the general fire scenarios. In the spatial interactions analysis, no credit is taken as all equipment in the affected zones is assumed to be damaged.

For the control room fire scenarios, credit for suppression of the postulated fires is implicitly considered in the combined geometry and severity factors. This suppression effectiveness is not separable from the severity curves. The severity factor curve is used to describe the conditional frequency of an electrical fire in the control room propagating a given distance or greater.

4.6 ANALYSIS OF PLANT SYSTEMS, SEQUENCES, AND PLANT RESPONSE

4.6.1 EVENT TREES

The fire initiators derived in Section 4.4 are applied to the relevant event trees as detailed in Figure 4.6-1. The fire initiating event sequences can be modeled using three groups of event trees: general plant initiator event trees, control room fire initiators event trees, and cable spreading room fire initiator event trees. These event trees are described in the following sections.

4.6.1.1 General Plant Initiator Event Trees

The general plant initiators (FS1 through FS8) are applied to the following DCPRA event trees in order:

- Electrical Support System Event Tree (Figure 4.6-2)
- Mechanical Support System Event Tree (Figure 4.6-3)
- General Transient Event Tree (Figure 4.6-4)
- Late Tree (Figure 4.6-5)
- Recovery Event Tree (Figure 4.6-6)

These event trees are identical to the general transient event trees described in the IPE Report (Reference 4-6), except the event trees have been modified to reflect the current as-built condition of the plant (For a discussion of plant changes see Section 4.2.2 on Configuration Management). The top event descriptions for the event trees are also contained in that report.

4.6.1.2 Control Room Fire Initiator Event Trees

The control room fire initiators are applied to specialized event trees. The top event definitions are provided below.

- **Top Event FEF - Extinguish Fire Before Equipment Fails.** Top event FEF represents the fraction of fires that are extinguished before the target equipment is damaged or fires of insufficient size to fail the postulated equipment. The value of the split fraction value is the integrated geometry and severity factors developed for each control room fire scenario. Note that top event FEF is included in the event trees as a modeling tool; top event FEF is actually a modifier on initiating event frequency.

- **Top Event FCR - Control Room Remains Habitable.** This top event models the probability the control room remains habitable given a control room fire has occurred.

- **Top Event FEB - Equipment Fails Before Evacuation.** This top event models the timing of equipment failure in relation to control room evacuation. The state

points resulting from this top events and top event FCR influence the assumptions about later top events.

- **Top Event FTP - RCPs Tripped Before Seals Damaged.** This top event applies only to initiating events VB1 (Failure of Auxiliary Saltwater and Component Cooling Water Systems) and VB4 (Loss of 4-kV Vital Buses), and represents the probability that RCPs are not tripped before the RCP seals are damaged, which is assumed to be 10 minutes.

- **Top Event FRE - Recovery of Equipment Prior to LOCA.** This top event represents the probability that ASW and CCW are recovered prior to a seal LOCA occurring.

- **Top Event FML - Mitigation of LOCA or Seal LOCA.** The top event represents the possibility of mitigating a seal LOCA caused by failure of ASW/CCW.

- **Top Event FLT - LOCA Terminated, PORV Closed.** This top event only applies to initiating events VB2A, VB2B, and VB23. The top event represents closing the PORV from the hot shutdown panel.

- **Top Event FHS - Conditional Probability of Hot Short.** This top event represents the probability of a fire-induced hot short, given a fire at an affected circuit (applied to PORVs).

- **Top Event FSU - Probability of Sustained Hot Short.** This top event represents the conditional probability of a sustained hot short, given that a hot short has occurred (applied to PORVs).

- **Top Event FPR - Probability for PORV Fail to Reseat.** This top event represents the probability of a PORV failing to reseat, given that it has failed open following a hot short.

- **Top Event FSH - Secondary Heat Removal Restored.** This top event only applies to fire initiating event VB23. The top event represents restoration of AFW from the hot shutdown panel.

Control room fire initiators VB1 and VB4 are quantified using event tree VB14, shown in Figure 4.6-7. Event tree VB2, shown in Figure 4.6-8, quantifies initiators VB2A and VB2B. Finally, Figure 4.6-9 shows event tree VB23, which quantifies initiator VB23. The split fraction values for the control room fire initiating events are contained in Table 4.6-1.

4.6.1.3 Cable Spreading Room Fire Initiator Event Trees

The cable spreading room fire initiators, CSR1 and CSR2 are quantified using event trees CSR1 and CSR2, shown in Figures 4.6-10 and 4.6-11. The top event descriptions for these event trees are provided below.

- **Top Event FEF - Combined Geometry and Severity Factor.** This top event represents the combined geometry and severity factor for the cable spreading room fire initiating events CSR1 (Loss of Auxiliary Saltwater and Component Cooling Water Systems) and CSR2 (Hot Short Fails Open PORV).

- **Top Event FRE - Human Actions for Recovery.** This top event represents the human actions for recovery. For initiating event CSR1, the top event represents the human actions to trip the RCPs prior to seal damage, and also the human action to restore CCW and ASW from the hot shutdown panel. For initiating event CSR2, the top event represents the human actions to close the PORVs from the hot shutdown panel.

- **Top Event FHS - Conditional Probability of Hot Short.** This top event represents the probability of a fire induced hot short, given a fire at an affected circuit (applied to PORVs).

- **Top Event FSU - Probability of Sustained Hot Short.** This top event represents the conditional probability of a sustained hot short, given that a hot short has occurred (applied to PORVs).

- **Top Event FPR - Probability for PORV Fail to Reseat.** This top event represents the probability of a PORV failing to reseat, given that it has failed open following a hot short.

The split fraction values used for the top events for the cable spreading room fire are listed in Table 4.6-2.

4.6.2 FIRE PRA RESULTS

The IPEEE 1993 Fire PRA sequence quantification resulted in a fire core damage frequency (CDF) point estimate of 2.7E-5 per year. To account for uncertainties in the initiating event frequencies, the component failure rates, and the equipment maintenance unavailabilities, the uncertainty in the fire initiating events core damage frequency was also calculated using the RISKMAN software. The IPEEE 1993 Fire PRA core damage frequency has the following characteristics:

Point Estimate = 2.726E-5

Monte Carlo Mean = 2.74E-5
5^{th} Percentile = 5.18E-6
Median = 1.87E-5
95^{th} Percentile = 7.57E-5

This section presents the results of the IPEEE 1993 Fire PRA portion of the DCPRA-1993 and discusses the important contributors to the core damage frequency resulting from fire initiating events.

In general, more significant figures are presented in the results and data tables than the accuracy of the PRA provides. However, the additional significant figures are important and useful when performing sensitivity studies and also allow for traceability and reproducibility of results.

4.6.2.1 Core Damage Sequences

The IPEEE reporting guidelines provided in NUREG-1407 suggests using the core damage sequence selection criteria provided in NUREG-1335 (Reference 4-26). Some of the NUREG-1335 sequence selection criteria apply to the evaluation of containment performance. An evaluation of the impact of fire initiating events on containment performance is described in Section 4.7; as such, this section only uses the NUREG-1335 selection criteria for core damage. The IPEEE 1993 Fire PRA provides the results in terms of systemic sequences, as opposed to functional sequences. The reporting guidelines for systemic sequences are as follows:

1. Any systemic sequence that contributes 1E-7 or more per reactor year to core damage.
2. Any systemic sequence within the upper 95 percent of the core damage frequency.
3. Any other systemic sequence that the utility determines to be important to core damage frequency.

Table 4.6-3 presents the 32 highest frequency systemic core damage sequences, representing 95 percent of the IPEEE 1993 Fire PRA core damage frequency. The lowest ranking sequence presented has a core damage frequency of 3.7E-8 per year. Although many sequences are reported that are lower than the reporting criteria requires (1E-7 per year), these sequences are reported so the upper 95 percent of the core damage frequency can be provided.

4.6.2.2 Fire PRA Results Evaluation

Section 4.6.2.1 provided an evaluation of the IPEEE 1993 Fire PRA results by examining the dominant sequences contributing to the IPEEE 1993 Fire PRA core damage frequency. This section interprets the results by examining the contributors to core damage from several vantage points. These vantage points include studying the results at the initiating event level, the top event or system level, and by the accident type.

Table 4.6-4 lists the contributions of the IPEEE 1993 Fire PRA initiating events to the IPEEE 1993 Fire PRA core damage frequency. In Table 4.6-4, it can be seen that two control room and two cable spreading room initiating events combine to contribute approximately 70 percent of the total fire-induced core damage frequency. The dominant control room and cable spreading room fires initiating events are:

- **VB1** - A fire in control room vertical board VB1, leading to failure of the ASW and/or the CCW systems.

- **VB4** - A fire in the control room vertical board VB4, leading to failure of 4-kV vital buses F, G, and H.

- **CSR1** - A fire in the cable spreading room circuitry affecting both the ASW and the CCW systems.

- **CSR2** - A fire in the cable spreading room leading to a hot short in the PORV circuitry, resulting in a failed open PORV.

Also shown in Table 4.6-4, only three fire initiating events outside of the control room and the cable spreading room contribute more than 1 percent to the IPEEE 1993 Fire PRA core damage frequency. These three initiating events are as follows:

- **FS1** - A fire that disables both motor-driven AFW pumps. As shown in Table 4.4-2, a variety of fire scenarios contribute to this fire PRA initiating event.

- **FS5** - A fire that leads to loss of both ASW pumps. As shown in Table 4.4-2, three fire scenarios initiating in three different fire areas contribute to this fire PRA initiating event.

- **FS6** - A fire that leads to the loss of vital 4-kV buses F and G. As shown in Table 4.4-2, several fire scenarios contribute to this fire PRA initiating event.

Top event and split fraction importance rankings can provide insights into how various independent failures contribute to fire PRA core damage, given that a fire initiating event has occurred. Table 4.6-5 provides the top event importance rankings for the IPEEE 1993 Fire PRA initiating events. The top events are ranked according to their percentage contribution to the total IPEEE 1993 Fire PRA core damage frequency, i.e., the percentage of the fire core damage frequency attributable to sequences that include failure of these events. Sequences that involve guaranteed failures or dependent top event failures are not counted in the Table 4.6-5 ranking.

The causes of system failure can be better understood by determining the importance of split fractions in the IPEEE 1993 Fire PRA model. The importance evaluation of the nonguaranteed failure split fractions is summarized in Table 4.6-6 (for the top 15 split fractions). For this group of split fractions, three different importance measures are shown as described below:

- the percentage contribution to core damage frequency of the sequences including the given split fraction failed (importance);

- the factor increase in the core damage frequency, resulting when the split fraction is arbitrarily assigned a failure frequency of 1.0 (risk achievement worth);

- the fraction of the core damage frequency, resulting when the split fraction is arbitrarily assigned a failure frequency of 0.0 (risk reduction).

4.6.2.3 Timing of Major Core Damage Accident Sequences

The dominant fire initiated core damage functional sequences are the following:

- Loss of CCW/ASW leading to a RCP seal LOCA
- PORV LOCA
- Loss of Secondary Heat Sink (AFW)
- Station blackout with AFW initially available
- Station blackout with no AFW available

The timing of accident sequences leading to core damage associated with these fire initiated events is similar to the IPE results for internal events. Section 4.6.1 of the IPE report (Reference 4-6) discussed the MAAP simulations of these sequences. The approximate timing of core damage for these sequences is shown in Table 4.7-7.

4.6.2.4 Fire PRA Vulnerability Evaluation

Based on the results presented in this study and the previous findings of the DCPRA-1988, no fundamental vulnerabilities with regard to fire-induced core damage exist at DCPP. The NUMARC Report 91-04, "Severe Accident Issue Closure Guideline," Reference 4-27, provides vulnerability screening guidelines for core damage sequences. These guidelines are summarized in Table 4.6-8. Based on the guidelines, a vulnerability refers to any component, system, operator action, or accident sequence that contributes more than 50 percent to the core damage frequency or has a frequency that exceeds 1E-4 per year. The IPEEE 1993 Fire PRA core damage frequency of 2.7E-5 per year is sufficiently low so as to preclude any vulnerabilities based solely on frequency.

Table 4.6-1. Split Fraction Values and Descriptions for the Control Room Event Trees			
Split Fraction	Top Event	Value	Description
FCR1	FCR	5.0E-02	CONTROL ROOM EVACUATION FREQUENCY
FEB1	FEB	5.0E-01	CONTROL ROOM EQUIP FAILURE OCCURS POST EVACUATION
FEF1	FEF	2.46E-02	CR-VB1 GEOM/SEVERITY FACTOR
FEF2	FEF	8.82e-3	CR-VB4 GEOM/SEVERITY FACTOR
FEF3	FEF	4.41E-02	CR-VB2A GEOM/SEVERITY FACTOR
FEF4	FEF	2.21E-3	CR-VB2B GEOM/SEVERITY FACTOR
FEF5	FEF	5.54E-3	CR-VB23 GEOM/SEVERITY FACTOR
FHSF	FHS	1.0	Hot Short Factor Not Applicable
FLT1	FLT	1.9002E-3	CR-VB2 ZHEF21 Op Act Close PORV HSDP - LOCA Term
FLT2	FLT	1.7001E-3	CR-VB23 ZHEF31 Op Act Close PORV HSDP - LOCA Term
FML1	FML	5E-1	CR-VB14 (J) LOCA Mitigation
FML2	FML	1E-2	CR-VB2A,VB23 (J) ZHEF22 LOCA Mitigation
FML3	FML	2.2187E-2	CR-VB23 (OB1) LOCA Mitigation
FMLF	FML	1.0	CR-LOCA Mitigation Cannot Succeed
FRE1	FRE	2.7002E-3	CR-VB1 ZHEF12 Op Act Recovery of ASW/CCW
FRE2	FRE	3.2004E-3	CR-VB4 ZHEF42 Op Act Recovery of Emergency AC and ASW/CCW
FREF	FRE	1.0	Recovery Not Credited.
FSH1	FSH	1.3001E-2	CR-VB23 ZHEF32 Op Act Recovery of Secondary Heat Removal
FTP1	FTP	2.3003E-3	CR-VB1 ZHEF11 Op Act Trip RCPs Prior To Seal Damage
FTP2	FTP	3.5003E-3	CR-VB4 ZHEF41 Op Act Trip RCPs Prior to Seal Damage
FTPF	FTP	1.0	CR-VB14 No Credit Assumed for Tripping RCPs

Table 4.6-2. Split Fraction Values and Descriptions for the Cable Spreading Room Fire Event Trees			
Split Fraction	Top Event	Value	Description
FEF6	FEF	8.2E-2	CSR1 GEOM/SEVERITY FACTOR
FEF7	FEF	1.38E-1	CSR2 Geom/Severity Factor
FHS1	FHS	2.9E-1	CSR2 Conditional Probability of Hot Short (Inst+Sust)
FHSF	FHS	1.0	Hot Short Factor Not Applicable
FPR1	FPR	1.58E-2	CSR2 (ZTV3RS)
FRE3	FRE	1.0291E-2	CSR1 (J) Op Act - Trip RCPs, Restore ASW/CCW
FRE4	FRE	2.6004E-3	CSR2 (J) Op Act - ZHEF61 - Close PORVs at HSDP
FREF	FRE	1.0	Recovery Not Credited.
FSU1	FSU	1.3793E-1	CSR2 Cond Prob of a Hot Short being Sustained

Table 4.6-3. Top Ranking Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
1	CSR FIRE 1 - LOSS OF ASW/CCW. - HUMAN ACTIONS FOR RECOVERY		COREMELT	5.65E-06	20.74
2	CSR FIRE 2 - PORV INDUCED LOCA. - CONDITIONAL PROBABILITY OF HOT SHORT - PROBABILITY FOR PORV FAIL TO RESEAT		COREMELT	3.65E-06	13.40
3	CONTROL ROOM FIRE AT VB-1 - CONTROL ROOM REMAINS HABITABLE - EQUIPMENT FAILS BEFORE EVACUATION		COREMELT	3.01E-06	11.06
4	CONTROL ROOM FIRE AT VB-1 - CONTROL ROOM REMAINS HABITABLE	- RCPS TRIPPED BEFORE SEALS DAMAGED	COREMELT	3.01E-06	11.06
5	FS6: LOSS OF BUSES HF & HG - RCPS IN OPERATION - AUXILIARY SALTWATER FROM UNIT 2	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	1.92E-06	7.06
6	FS6: LOSS OF BUSES HF & HG - RCS PRESSURE RELIEF AND PORV RECLOSURE	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	1.79E-06	6.57
7	FS5: LOSS OF ASW - RCPS IN OPERATION - RCP SEAL INTEGRITY	- AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	1.48E-06	5.44
8	CONTROL ROOM FIRE AT VB-4 - CONTROL ROOM REMAINS HABITABLE - EQUIPMENT FAILS BEFORE EVACUATION		COREMELT	1.08E-06	3.96
9	CONTROL ROOM FIRE AT VB-4 - CONTROL ROOM REMAINS HABITABLE	- RCPS TRIPPED BEFORE SEALS DAMAGED	COREMELT	1.08E-06	3.96
10	CSR FIRE 2 - PORV INDUCED LOCA. - CONDITIONAL PROBABILITY OF HOT SHORT - PROBABILITY OF SUSTAINED HOT SHORT - PROBABILITY FOR PORV FAIL TO RESEAT		COREMELT	5.83E-07	2.14
11	FS1: LOSS OF BOTH AFW PUMPS - AUXILIARY FEEDWATER SYSTEM - OPERATOR INITIATES FEED AND BLEED COOLING	- RCPS IN OPERATION	COREMELT	4.35E-07	1.59
12	FS1: LOSS OF BOTH AFW PUMPS - AUXILIARY FEEDWATER SYSTEM - REACTOR VESSEL INTEGRITY	- RCPS IN OPERATION	COREMELT	4.21E-07	1.55

Table 4.6-3. Top Ranking Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
13	CONTROL ROOM FIRE AT VB-1 - RECOVERY OF EQUIPMENT PRIOR TO LOCA	- MITIGATION OF SEAL LOCA	COREMELT	3.08E-07	1.13
14	FS1: LOSS OF BOTH AFW PUMPS - 125V DC POWER BUS G	- VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 12 - 120V VITAL INSTRUMENT AC CHANNEL II - AUXILIARY FEEDWATER SYSTEM - OPERATOR INITIATES FEED AND BLEED COOLING - RHR PUMP TRAIN A - CONTAINMENT SUMP VALVE A	COREMELT	2.40E-07	.88
15	FS5: LOSS OF ASW	- AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	1.49E-07	.55
16	CONTROL ROOM FIRE AT VB-1 - RCPS TRIPPED BEFORE SEALS DAMAGED - MITIGATION OF SEAL LOCA		COREMELT	1.32E-07	.48
17	CONTROL ROOM FIRE AT VB-4 - RECOVERY OF EQUIPMENT PRIOR TO LOCA	- MITIGATION OF SEAL LOCA	COREMELT	1.31E-07	.48
18	CSR FIRE 2 - PORV INDUCED LOCA. - CONDITIONAL PROBABILITY OF HOT SHORT - PROBABILITY OF SUSTAINED HOT SHORT - HUMAN ACTION TO CLOSE PORV AT HSDP		COREMELT	9.62E-08	.35
19	CONTROL ROOM FIRE AT VB-4 - RCPS TRIPPED BEFORE SEALS DAMAGED - MITIGATION OF SEAL LOCA		COREMELT	7.19E-08	.26
20	FS5: LOSS OF ASW - 125V DC POWER BUS G	- VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 12 - 120V VITAL INSTRUMENT AC CHANNEL II - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	7.00E-08	.26
21	FS5: LOSS OF ASW - 125V DC POWER BUS H	- VITAL AC 4KV BUS H - 480V SWITCHGEAR BUS H - DIESEL GENERATOR 11 - 120V VITAL INSTRUMENT AC CHANNEL III - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	6.99E-08	.26
22	FS1: LOSS OF BOTH AFW PUMPS - AUXILIARY FEEDWATER SYSTEM - RHR PUMP TRAIN A - RHR PUMP TRAIN B	- RCPS IN OPERATION	COREMELT	4.87E-08	.18
23	CR FIRE VB2 - (B) PORV LOCA + CH PMP FAIL - LOCA TERMINATED, PORV BLOCK VALVE CLOSED	- MITIGATION OF LOCA	COREMELT	4.79E-08	.18

Table 4.6-3. Top Ranking Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
24	FS6: LOSS OF BUSES HF & HG - UNIT 2 125V DC 22, 480V 2G & 4KV HG - UNIT 2 125V DC 23, 480V 2H & 4KV HH - RCPS IN OPERATION - AUXILIARY SALTWATER FROM UNIT 2	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	4.79E-08	.18
25	FS6: LOSS OF BUSES HF & HG - UNIT 2 125V DC 21, 480V 2F & 4KV HF - UNIT 2 125V DC 22, 480V 2G & 4KV HG - RCPS IN OPERATION - AUXILIARY SALTWATER FROM UNIT 2	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	4.55E-08	.17
26	FS6: LOSS OF BUSES HF & HG - UNIT 2 125V DC 21, 480V 2F & 4KV HF - UNIT 2 125V DC 23, 480V 2H & 4KV HH - RCPS IN OPERATION - AUXILIARY SALTWATER FROM UNIT 2	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	4.55E-08	.17
27	FS6: LOSS OF BUSES HF & HG - UNIT 2 125V DC 22, 480V 2G & 4KV HG - UNIT 2 125V DC 23, 480V 2H & 4KV HH - RCS PRESSURE RELIEF AND PORV RECLOSURE	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	4.46E-08	.16

Table 4.6-3. Top Ranking Core Damage Sequences

Rank No.	Sequence Description	Guaranteed Events/Comments	End State	Frequency (per year)	Percent
28	FS6: LOSS OF BUSES HF & HG - UNIT 2 125V DC 22, 480V 2G & 4KV HG - RCPS IN OPERATION - AUXILIARY SALTWATER FROM UNIT 2	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RCP SEAL INTEGRITY - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	4.25E-08	.16
29	FS6: LOSS OF BUSES HF & HG - UNIT 2 125V DC 21, 480V 2F & 4KV HF - UNIT 2 125V DC 22, 480V 2G & 4KV HG - RCS PRESSURE RELIEF AND PORV RECLOSURE	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	4.23E-08	.16
30	FS6: LOSS OF BUSES HF & HG - UNIT 2 125V DC 21, 480V 2F & 4KV HF - UNIT 2 125V DC 23, 480V 2H & 4KV HH - RCS PRESSURE RELIEF AND PORV RECLOSURE	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	4.23E-08	.16
31	FS6: LOSS OF BUSES HF & HG - UNIT 2 125V DC 22, 480V 2G & 4KV HG - RCS PRESSURE RELIEF AND PORV RECLOSURE	- VITAL AC 4KV BUS F - VITAL AC 4KV BUS G - 480V SWITCHGEAR BUS F - 480V SWITCHGEAR BUS G - DIESEL GENERATOR 13 - DIESEL GENERATOR 12 - AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	3.95E-08	.15
32	FS5: LOSS OF ASW - UNIT 2 125V DC 22, 480V 2G & 4KV HG - UNIT 2 125V DC 23, 480V 2H & 4KV HH - RCPS IN OPERATION - RCP SEAL INTEGRITY	- AUXILIARY SALT WATER SYSTEM - COMPONENT COOLING WATER SYSTEM - CENTRIFUGAL CHARGING PUMPS - SAFETY INJECTION PUMPS - RHR PUMP TRAIN A - RHR PUMP TRAIN B - CONTAINMENT FAN COOLERS - OPERATOR SWITCH TO CONT SUMP RECIRCULATION	COREMELT	3.69E-08	.14

Table 4.6-4. Internal Fire Initiated Core Damage Frequency

Initiator	Initiator Frequency	Equipment Damage	Core Damage Frequency	Rank	CDF % Contribution
FS1	4.356E-4	Loss of Both Motor-Driven Auxiliary Feedwater Pumps	1.530E-6	7	5.6
FS2	2.967E-3	Loss of All Charging Pumps	1.702E-8	13	< 1
FS3	1.659E-6	Loss of All CCW Pumps	2.985E-8	12	< 1
FS4	1.832E-3	Loss of Control Room Ventilation	1.037E-8	15	< 1
FS5	1.174E-4	Loss of Both ASW Pumps	2.122E-6	6	7.8
FS6	6.933E-5	Loss of Buses F and G	4.444E-6	3	16
FS7	6.460E-5	Loss of Buses G and H	1.304E-7	8	< 1
FS8	1.401E-8	Delayed Failure of 4-kV Vital Buses F, G, and H	1.401E-8	14	< 1
VB1	4.90E-3	Failure of Auxiliary Saltwater and Component Cooling Water Systems	6.467E-6	1	24
VB2A	4.90E-3	Hot Short Causes Failed Open PORV	5.987E-8	9	< 1
VB2B	4.90E-3	Failed Open PORV combined with failure of Charging Pumps	5.043E-8	10	< 1
VB23	4.90E-3	Failed Open PORV combined with Loss of Auxiliary Feedwater System	3.250E-8	11	< 1
VB4	4.90E-3	Loss of Auxiliary Saltwater and Component Cooling Water Systems from Loss of 4-kV Vital Buses	2.364E-6	5	8.7
CSR1	6.70E-3	Loss of Auxiliary Saltwater and Component Cooling Water Systems	5.654E-6	2	21
CSR2	6.70E-3	Hot Short Causes Failed Open PORV	4.331E-6	4	16
TOTAL			2.726E-5		

Table 4.6-5. Fire PRA Top Event Core Damage Frequency Importance Ranking - Nonguaranteed Failures			
Ranking	Top Event ID	Percentage of Fire CDF	Event Description
1	FEF (*)	70	Extinguish Fire Before Equipment Fails in Control Room or Cable Spreading Room
2	FCR	30	Control Room Remains Habitable
3	FRE	23	Human Actions for Recovery
4	FHS	16	Conditional Probability of Hot Short
5	FPR	16	Probability for PORV to Fail to Reseat
6	RP	15	Trip RCPs Prior to Seal Damage
7	FEB	15	Equipment Fails Before Evacuation
8	RA	8.5	Cross-tie to Unit 2 ASW
9	PR	8.1	PORV Fails to Reseat For non-control room or cable spreading room fire
10	SE	6.6	Seal Integrity Maintained
11	AW	4.7	Auxiliary Feedwater
12	FSU	2.6	Probability of Sustained Hot Short
13	OB	2.1	Bleed and Feed
14	VI	2.0	Vessel Integrity Maintained
15	DG	1.5	DC Bus G
16	FML	.8	Mitigation of LOCA or Seal LOCA
17	FTP	.7	RCPs Tripped Before Seals Damaged
18	DH	.4	DC Bus H
19	SB	.3	SSPS Train B
20	SA	.3	SSPS Train A
21	LB	.3	RHR Train B
22	RF	.3	Cold Leg Recirculation
23	LA	.3	RHR Train A
24	CC	.3	Component Cooling Water
* Note that FEF is a modeling tool only. As such, this top event does not represent an area of investigation for improving plant safety.			

Table 4.6-6. Importance Evaluation of Nonguaranteed Failure Event Tree Split Fractions

Rank	Split Fraction		Importance Measures			
	Name	Description of Item Failed (with Boundary Conditions)	Reference Split Fraction Value	Percentage of CDF w/Split Fraction	Risk Achievement Worth	Risk Reduction Worth
1	FCR1	Control Room Evacuation Frequency	5.0E-2	30	6.7	0.70
2	FEF1 (*)	Control Room - VB1 Geometry/Severity Factor	2.46E-2	24	10.4	0.76
3	FEF6 (*)	Cable Spreading Room 1 Geometry/Severity Factor	8.2E-2	21	3.3	0.79
4	FRE3	Cable Spreading Room 1 - Operator Action to trip RCPs, restore ASW/CCW	1.03E-2	21	21.0	0.79
5	FHS1	Cable Spreading Room 2 - Conditional Probability of Hot Short	2.9E-1	16	1.4	0.84
6	FPR1	Cable Spreading Room 2 - Probability of PORV Failing to Reseat	1.58E-2	16	10.9	0.84
7	FEF7 (*)	Cable Spreading Room 2 - Geometry/Severity Fire	1.38E-1	16	2.0	0.84
8	FEB1	Control Room Equipment Failure Occurs Post Evacuation	5.0E-1	15	1.0	1.0
9	FEF2 (*)	Control Room VB4 - Geometry/Severity Factor	8.8E-3	8.7	10.7	0.91
10	RA4	ASW Unit 2 Crosstie Following Loss of Unit 1 ASW	3.4E-2	8.5	3.4	0.91
11	PRD	PORV Challenged and Fails to Reseat	3.09E-2	8.0	3.4	0.92
12	SE1	Fire Water Hookup to Charging Pumps Following Loss of CCW	1.49E-2	6.5	5.3	0.94
13	AW4	Auxiliary Feedwater with Motor Driven Pumps Unavailable	5.9E-2	4.5	1.7	0.96
14	FSU1	Control Room 2 - Conditional Probability of a Hot Short being Sustained	1.4E-1	2.6	1.0	1.0
15	OB1	Bleed and Feed	2.2E-2	2.1	1.9	0.98

* Note that split fractions related to top event FEF are a modeling tool only. As such, these split fractions do not represent areas of investigation for improving plant safety.

Table 4.6-7. Timing of Fire-Induced Core Damage Sequences	
Functional Sequence Type	Core Damage Timing (Hours)
Loss of CCW/ASW Leading to an RCP Seal LOCA	9 - 202
PORV LOCA	13
Loss of Secondary Heat Sink (AFW)	20
Pressurized Thermal Shock	1.4
Station Blackout with AFW Initially Available	12
Station Blackout with AFW Unavailable	2

Table 4.6-8. Core Damage Vulnerability Screening Guidelines	
Core Damage Frequency Per Group (Per Reactor Year)	Recommended Action
Less than 1E-6	No action required
1E-5 to 1E-6	Establish Severe Accident Management Guideline, with emphasis on preventing core damage, vessel failure, and containment failure.
1E-4 to 1E-5 or 20% to 50% of CDF	Make change in EOPs, other plant procedures, or make minor hardware change, with emphasis on prevention of core damage; or establish Severe Accident Management Guideline.
Greater than 1E-4 or Greater than 50% of CDF	**VULNERABILITY** - Make plant administrative, procedural or hardware modification, with emphasis on reducing the likelihood of the sequence initiator; or make change in plant procedures with emphasis on prevention of core damage; or establish Severe Accident Management Guideline.

Figure 4.6-1. Event Trees Used for Fire PRA

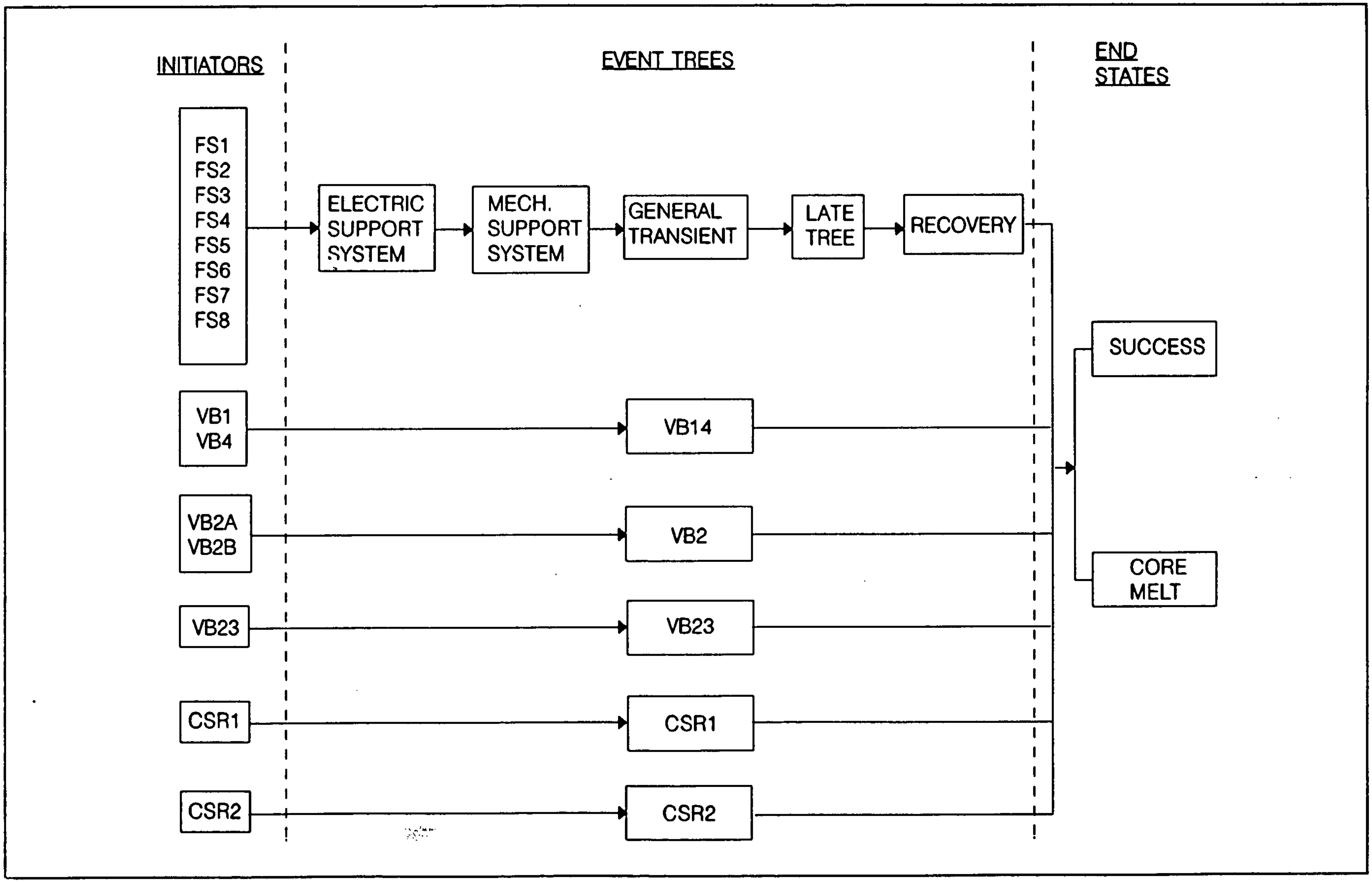


Figure 4.6-2. ELECPWR Electric Power Support Event Tree

Page No. 1

Event Tree: ELECPWR

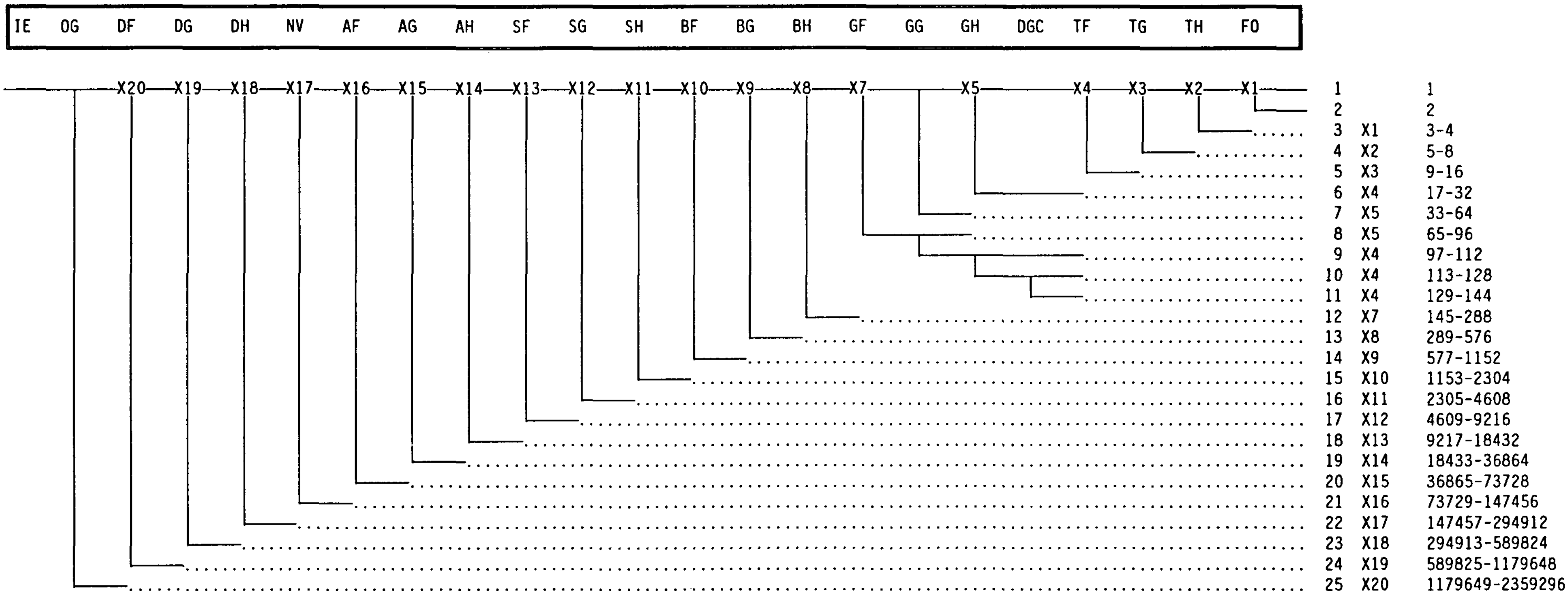


Figure 4.6-2 ELECPWR Electric Power Support Event Tree

Top Event Legend for Tree: ELECPWR

Page No. 2

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
OG	OFFSITE GRID
DF	125V DC POWER BUS F
DG	125V DC POWER BUS G
DH	125V DC POWER BUS H
NV	NONVITAL 4KV POWER
AF	VITAL AC 4KV BUS F
AG	VITAL AC 4KV BUS G
AH	VITAL AC 4KV BUS H
SF	480V SWITCHGEAR BUS F
SG	480V SWITCHGEAR BUS G
SH	480V SWITCHGEAR BUS H
BF	UNIT 2 125V DC 21, 480V 2F & 4KV HF
BG	UNIT 2 125V DC 22, 480V 2G & 4KV HG
BH	UNIT 2 125V DC 23, 480V 2H & 4KV HH
GF	DIESEL GENERATOR 13
GG	DIESEL GENERATOR 12
GH	DIESEL GENERATOR 11
DGC	DIESEL GENERATOR UNIT 1/2 COUPLING
TF	UNIT 2 DIESEL GENERATOR 23
TG	UNIT 2 DIESEL GENERATOR 22
TH	UNIT 2 DIESEL GENERATOR 21
FO	DIESEL FUEL OIL TRANSFER SYSTEM

Figure 4.6-3. MECHSP Mechanical Support Tree

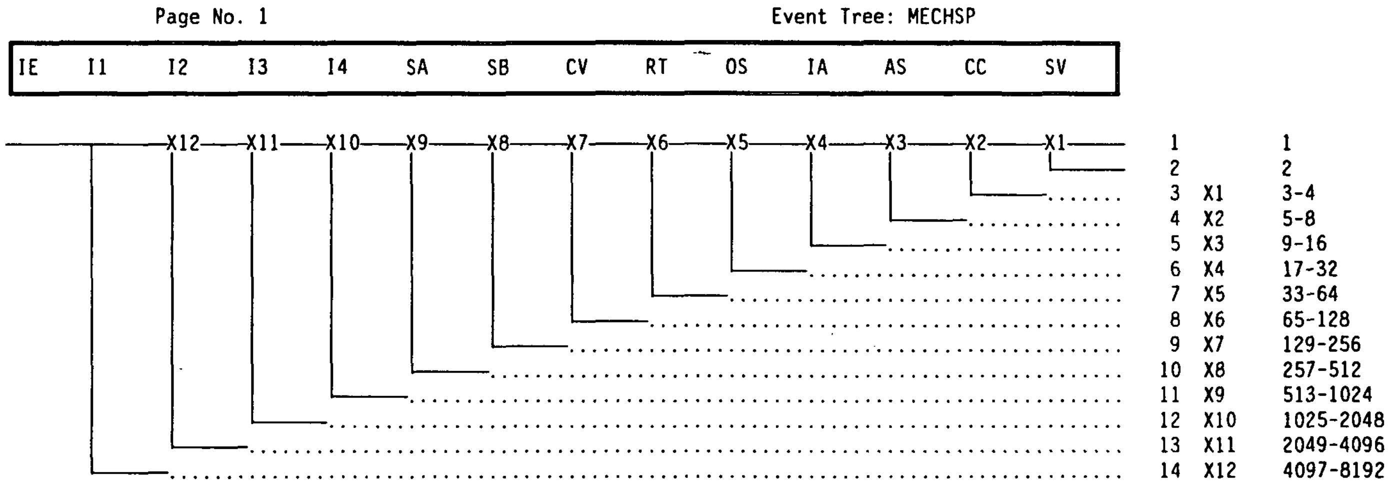


Figure 4.6-3 MECHSP Mechanical Support Tree

Page No. 2

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
I1	120V VITAL INSTRUMENT AC CHANNEL I
I2	120V VITAL INSTRUMENT AC CHANNEL II
I3	120V VITAL INSTRUMENT AC CHANNEL III
I4	120V VITAL INSTRUMENT AC CHANNEL IV
SA	SSPS TRAIN A
SB	SSPS TRAIN B
CV	CONTROL ROOM HVAC SYSTEM
RT	REACTOR PROTECTION SYSTEM
OS	MANUAL SI ACTUATION
IA	INSTRUMENT AIR
AS	AUXILIARY SALT WATER SYSTEM
CC	COMPONENT COOLING WATER SYSTEM
SV	480V SWITCHGEAR VENTILATION SYSTEM

Figure 4.6-4. GENTRN General Transients Event Tree

Page No. 1

Event Tree: GENTRN

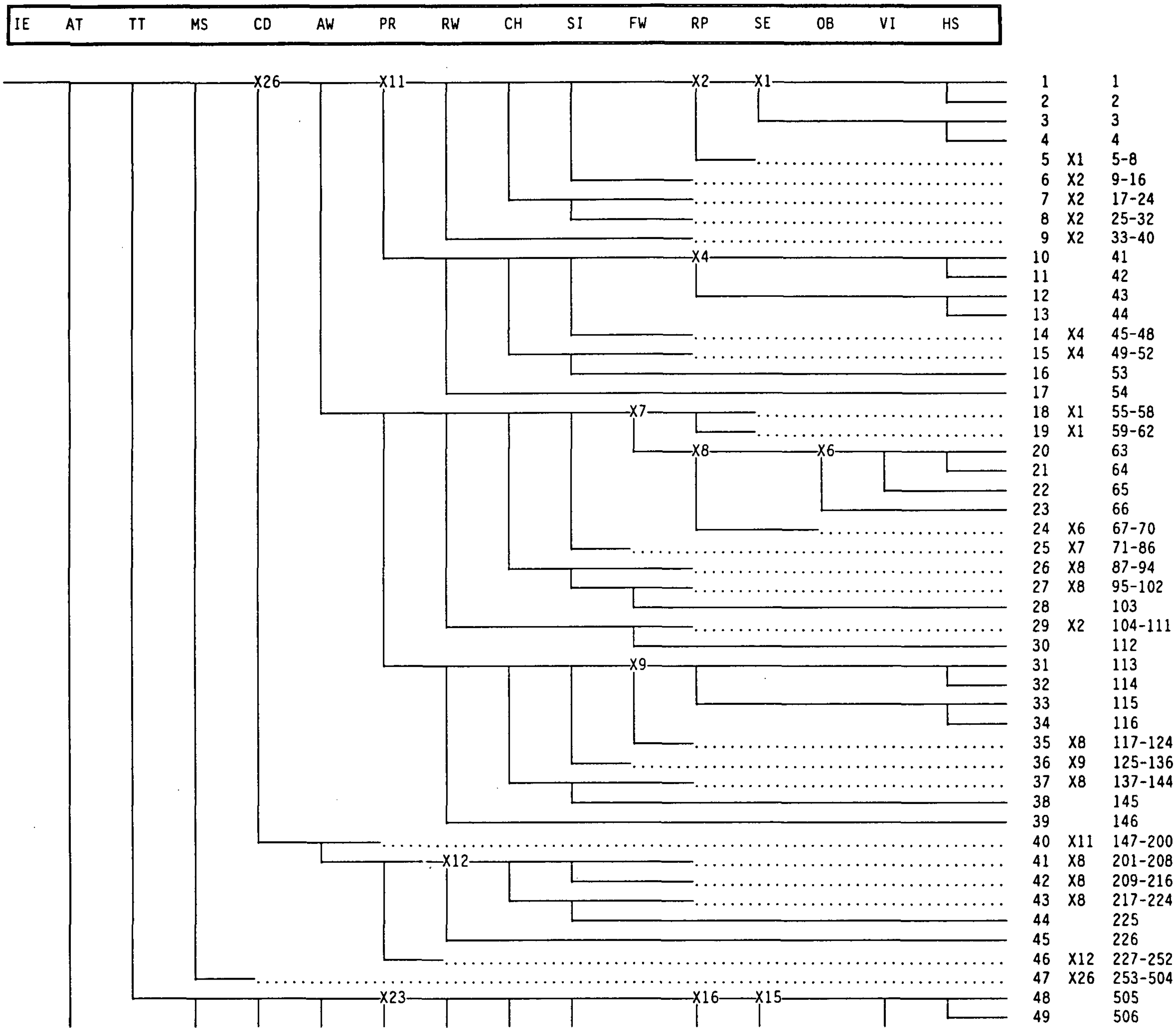
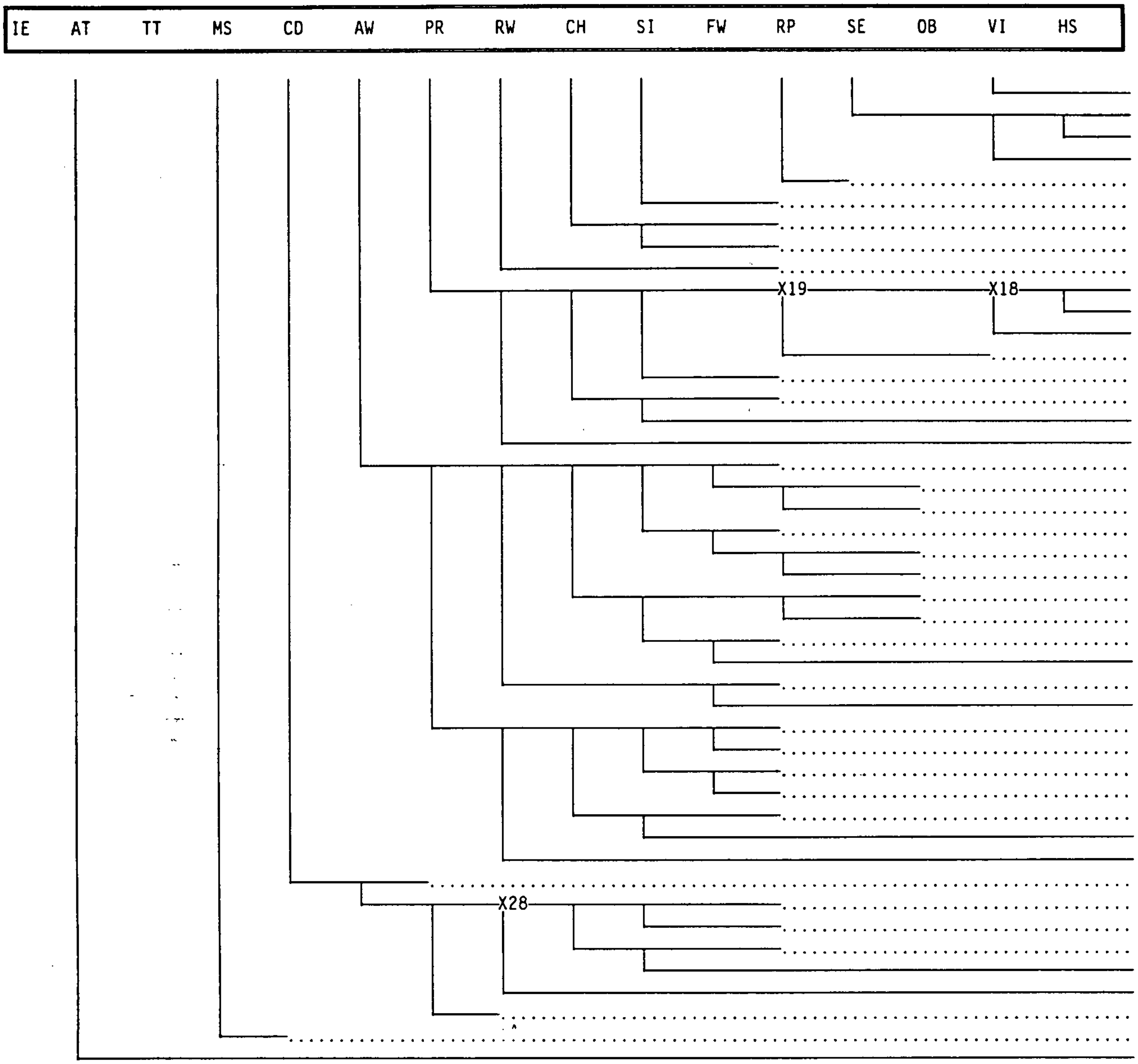


Figure 4.6-4. GENTRN General Transients Event Tree

Page No. 2

Event Tree: GENTRN



50		507
51		508
52		509
53		510
54	X15	511-516
55	X16	517-528
56	X16	529-540
57	X2	541-548
58	X2	549-556
59		557
60		558
61		559
62	X18	560-562
63	X19	563-568
64	X19	569-574
65		575
66		576
67	X16	577-588
68	X6	589-592
69	X6	593-596
70	X16	597-608
71	X6	609-612
72	X6	613-616
73	X6	617-620
74	X6	621-624
75	X2	625-632
76		633
77	X2	634-641
78		642
79	X19	643-648
80	X8	649-656
81	X19	657-662
82	X8	663-670
83	X8	671-678
84		679
85		680
86	X23	681-752
87	X8	753-760
88	X8	761-768
89	X8	769-776
90		777
91		778
92	X28	779-804
93	X26	805-1056
94		1057

Figure 4.6-4 GENTRN General Transients Event Tree

Top Event Legend for Tree: GENTRN

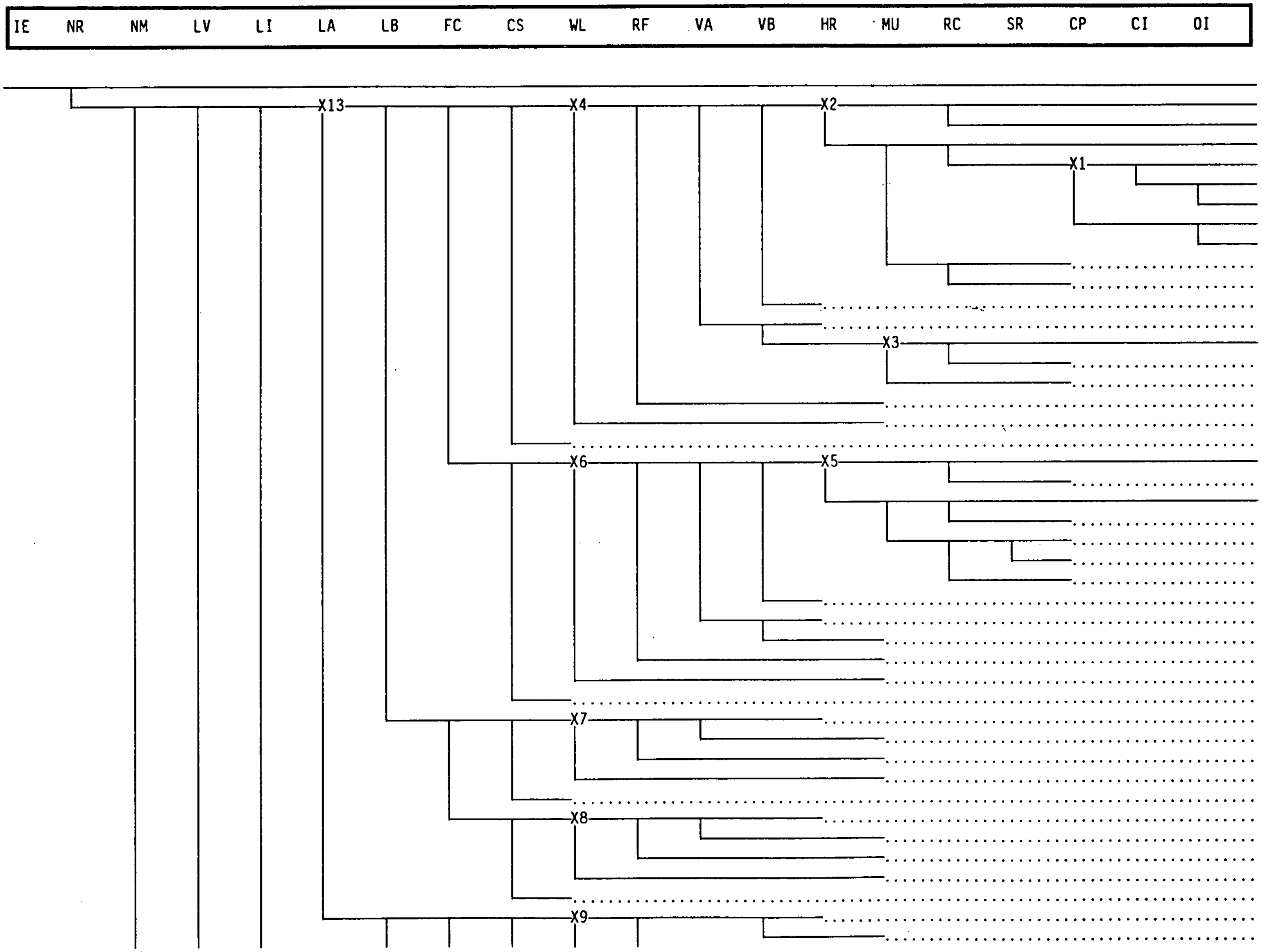
Page 3

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
AT	REACTOR TRIP SUCCESSFUL
TT	TURBINE TRIP
MS	MAIN STEAM LINES ISOLATION VALVES REMAIN OPEN
CD	CONDENSER AND CONDENSATE SYSTEM
AW	AUXILIARY FEEDWATER SYSTEM
PR	RCS PRESSURE RELIEF AND PORV RECLOSURE
RW	REFUELING WATER STORAGE TANK
CH	CENTRIFUGAL CHARGING PUMPS
SI	SAFETY INJECTION PUMPS
FW	MAIN FEEDWATER AND CONDENSATE SYSTEM
RP	RCPS IN OPERATION
SE	RCP SEAL INTEGRITY
OB	OPERATOR INITIATES FEED AND BLEED COOLING
VI	REACTOR VESSEL INTEGRITY
HS	CONTROL ROOM INDICATIONS AND PLANT CONTROL

Figure 4.6-5. LTREE Late Event Tree

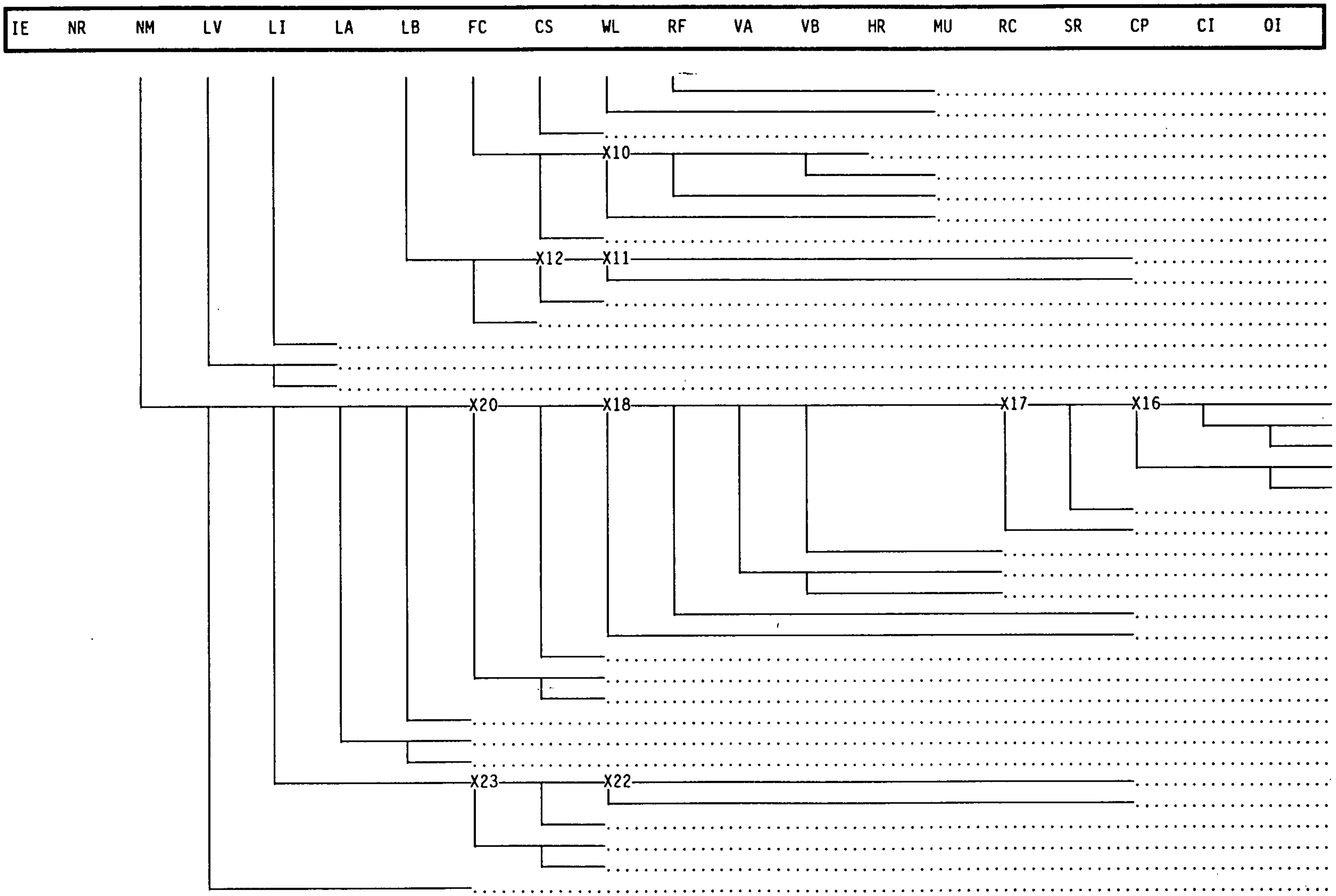
Page No. 1

Event Tree: LTREE



1		1
2		2
3		3
4		4
5		5
6		6
7		7
8		8
9		9
10	X1	10-14
11	X1	15-19
12	X2	20-37
13	X2	38-55
14		56
15	X1	57-61
16	X1	62-66
17	X3	67-77
18	X3	78-88
19	X4	89-175
20		176
21	X1	177-181
22		182
23	X1	183-187
24	X1	188-192
25	X1	193-197
26	X1	198-202
27	X5	203-229
28	X5	230-256
29	X3	257-267
30	X3	268-278
31	X3	279-289
32	X6	290-403
33	X2	404-421
34	X3	422-432
35	X3	433-443
36	X3	444-454
37	X7	455-505
38	X5	506-532
39	X3	533-543
40	X3	544-554
41	X3	555-565
42	X8	566-625
43	X2	626-643
44	X3	644-654

Figure 4.6-5. LTREE Late Event Tree



45	X3	655-665
46	X3	666-676
47	X9	677-727
48	X5	728-754
49	X3	755-765
50	X3	766-776
51	X3	777-787
52	X10	788-847
53	X1	848-852
54	X1	853-857
55	X11	858-867
56	X12	868-887
57	X13	888-1773
58	X13	1774-2659
59	X13	2660-3545
60		3546
61		3547
62		3548
63		3549
64		3550
65	X16	3551-3555
66	X16	3556-3560
67	X17	3561-3575
68	X17	3576-3590
69	X17	3591-3605
70	X16	3606-3610
71	X16	3611-3615
72	X18	3616-3685
73	X18	3686-3755
74	X18	3756-3825
75	X20	3826-4105
76	X20	4106-4385
77	X20	4386-4665
78	X16	4666-4670
79	X16	4671-4675
80	X22	4676-4685
81	X22	4686-4695
82	X22	4696-4705
83	X23	4706-4745

Figure 4.6-5. LTREE Late Event Tree

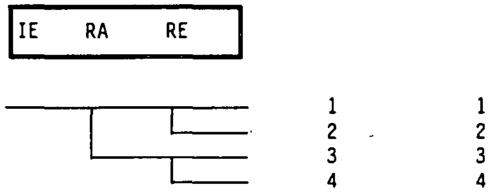
Top Event Legend for Tree: LTREE

Page 3

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
NR	NO RECIRCULATION
NM	NO CORE DAMAGE
LV	RHR SUCTION FROM RWST
LI	COLD LEG INJECTION LINES
LA	RHR PUMP TRAIN A
LB	RHR PUMP TRAIN B
FC	CONTAINMENT FAN COOLERS
CS	CONTAINMENT SPRAY
WL	WATER LEVEL FOR SUMP RECIRCULATION
RF	OPERATOR SWITCH TO CONT SUMP RECIRCULATION
VA	CONTAINMENT SUMP VALVE A
VB	CONTAINMENT SUMP VALVE B
HR	HIGH PRESSURE RECIRCULATION
MU	MAKEUP TO RWST/HOT LEG SUCTION
RC	CCW COOLING TO RHR HEAT EXCHANGERS
SR	CONTAINMENT SPRAY RECIRCULATION
CP	CONTAINMENT ISOLATION > 3 INCHES
CI	CONTAINMENT ISOLATION < 3 INCHES
OI	OPERATOR ACTION TO ISOLATE CONTAINMENT

Figure 4.6-6. RECV Recovery Event Tree

Page No. 1 Event Tree: RECV

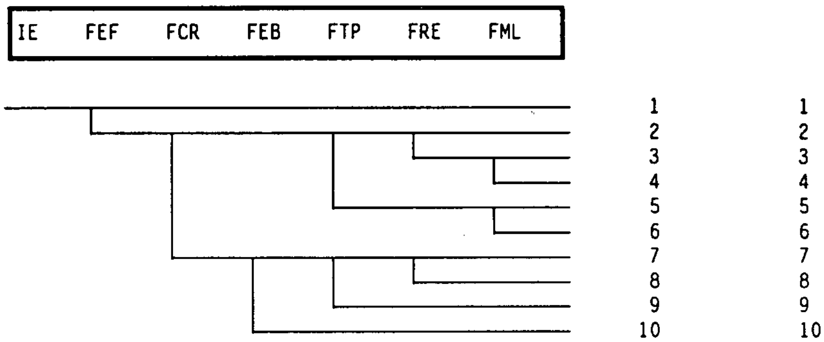


Top Event Legend for Tree: RECV

Top Event Designator.....	Top Event Description............................
IE	SEQUENCES TO BE RECOVERED
RA	AUXILIARY SALTWATER FROM UNIT 2
RE	RECOVERY ACTION PRIOR TO CORE UNCOVERY

Figure 4.6-7. VB14 Control Room Fire Event Tree

Page No. 1 Event Tree: VB14

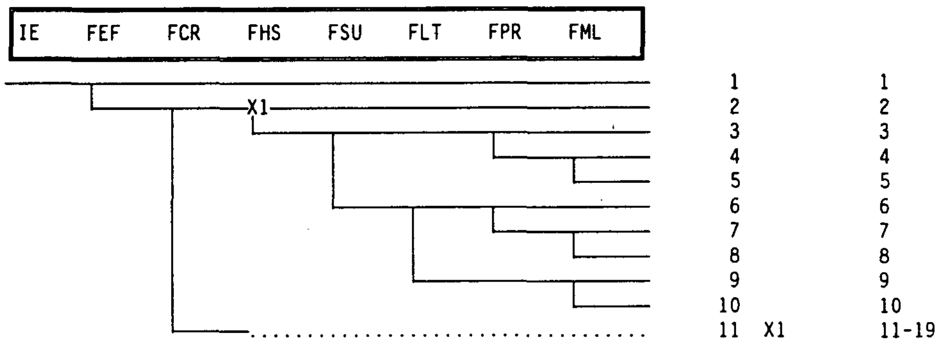


Top Event Legend for Tree: VB14

Top Event Designator.....	Top Event Description............................
IE	CONTROL ROOM FIRE SCENARIOS VB-1 AND VB-4
FEF	EXTINGUISH FIRE BEFORE EQUIPMENT FAILS
FCR	CONTROL ROOM REMAINS HABITABLE
FEB	EQUIPMENT FAILS BEFORE EVACUATION
FTP	RCPS TRIPPED BEFORE SEALS DAMAGED
FRE	RECOVERY OF EQUIPMENT PRIOR TO LOCA
FML	MITIGATION OF SEAL LOCA

Figure 4.6-8 VB2 Control Room Fire Event Tree

Event Tree: VB2

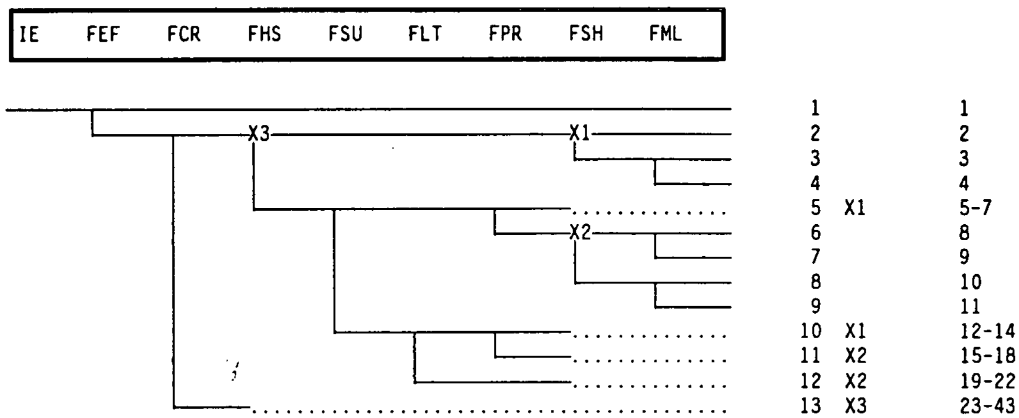


Top Event Legend for Tree: VB2

Top Event Designator.....	Top Event Description............................
IE	Initiating Event
FEF	EXTINGUISH FIRE BEFORE EQUIPMENT FAILS
FCR	CONTROL REMAINS HABITABLE
FHS	COND PROB OF HOT SHORT
FSU	COND PROB OF SUSTAINED HOT SHORT
FLT	LOCA TERMINATED FROM HSDP
FPR	PROB FOR PORV TO FAIL TO RESEAT
FML	MITIGATION OF LOCA

Figure 4.6-9. VB23 Control Room Fire Event Tree

Event Tree: VB23



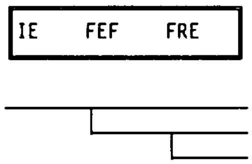
Top Event Legend for Tree: VB23

Top Event Designator.....	Top Event Description............................
IE	CONTROL ROOM FIRE SCENARIOS VB-2/3
FEF	EXTINGUISH FIRE BEFORE EQUIPMENT FAILS
FCR	CONTROL ROOM REMAINS HABITABLE
FHS	COND PROB OF HOT SHORT
FSU	COND PROB OF SUSTAINED HOT SHORT
FLT	LOCA TERMINATED, PORV CLOSED AT HSDP
FPR	PROB PORV FAIL TO RESEAT
FSH	SECONDARY HEAT REMOVAL RESTORED
FML	MITIGATION OF LOCA AND/OR LOSS OF HEAT REMOVAL

Figure 4.6-10. CSR1 Cable Spreading Room Fire Event Tree

Page No. 1

Event Tree: CSR1



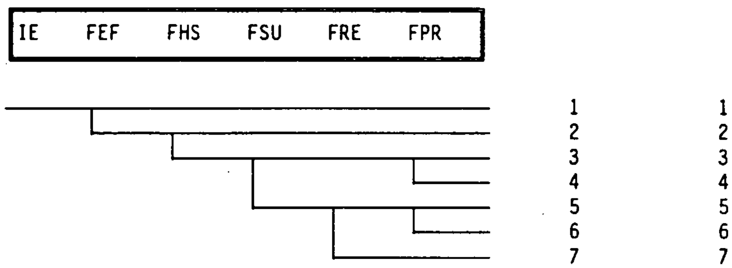
1	1
2	2
3	3

Top Event Legend for Tree: CSR1

Top Event Designator.....	Top Event Description............................
IE	Initiating Event
FEF	COMBINED GEOMETRY AND SEVERITY FACTOR
FRE	HUMAN ACTIONS FOR RECOVERY

Figure 4.6-11. CSR2 Cable Spreading Room Fire Event Tree

Page No. 1 Event Tree: CSR2



Top Event Legend for Tree: CSR2

Top Event Designator.....	Top Event Description....................................
IE	Initiating Event
FEF	COMBINED GEOMETRY/SEVERITY FACTORS
FHS	CONDITIONAL PROBABILITY OF HOT SHORT
FSU	PROBABILITY OF SUSTAINED HOT SHORT
FRE	HUMAN ACTION TO CLOSE PORV AT HSDP
FPR	PROBABILITY FOR PORV FAIL TO RESEAT

4.7 ANALYSIS OF CONTAINMENT PERFORMANCE

GL 88-20, Supplement 4, Section 4.2 directs the following:

> "Containment performance should be assessed to determine if vulnerabilities stemming from sequences that involve containment failure modes distinctly different from those obtained in the internal events analysis are predicted."

In Appendix 2 of GL 88-20, Supplement 4, the following aspects of this assessment are listed.

- "Identify mechanisms that could lead to containment bypass"
- "Identify mechanisms that could cause failure of the containment to isolate"
- "Determine the availability and performance of the containment systems under the external hazard to see if they are different from those evaluated under the internal event evaluation"

The potential for containment bypass is addressed in Section 4.7.1. The potential impact of fire on containment isolation MOVs and AOVs is addressed in Section 4.7.2. The results of an evaluation of the potential impact of fires on containment systems performance is discussed in Section 4.7.3.

Fire-induced electrical faults such as open circuits, short circuits to ground, or hot shorts to de-energized conductors can affect containment performance (containment isolation, containment bypass, or containment heat removal). Thus, it is important to assess the failure modes of containment isolation valves and containment systems in response to fire-induced electrical faults. Fire scenarios which lead to open circuits in the power cable of an MOV would cause the valve to fail "as is". Similarly, shorts to ground or open circuits of certain 120-V AC control circuits for MOVs could cause the valve to fail in the "as is" position. A hot short to a de-energized 120 V AC control circuit (or a three phase hot short to the power cables) could cause an MOV to change position.

Open circuits or shorts to ground would typically cause an AOV to move to its fail position. Hot shorts and multiple "smart" shorts to ground on AOV control circuits could result in inadvertent valve movement. Such movements can be terminated by removing power to the affected control circuit.

Additionally, in the event that a hot short were to occur, its duration is expected to be short. According to NUREG/CR-4840, "Procedures for the External Event Core Damage Frequency Analysis for NUREG-1150,"

> "Even if (fire-induced) spurious actuations were to occur, it is known from past fires (such as Brown's Ferry) that within approximately one-half hour spurious actuations terminate in open circuits."

Reference 4-28 contains estimated conditional probabilities for hot shorts given a cable fire. A mean frequency of 0.25 is assumed for a momentary hot short, possibly resulting in a spurious signal. A mean frequency of 0.04 is assumed for a sustained hot short. Depending on the specific failure mode of the affected circuit, one or both of the hot short conditional probabilities may be applied to the scenario quantification.

4.7.1 CONTAINMENT BYPASS POTENTIAL

A previous evaluation (Reference 4-29) quantifying the potential for Interfacing System LOCA (ISLOCA) was reviewed for the potential of fire induced containment bypass scenarios. An Interfacing System LOCA is an event that involves the failure of the valves that isolate the high pressure RCS from low pressure systems, such as the RHR system. If the ISLOCA leads to failure in low pressure system piping that simultaneously causes a loss of high pressure reactor coolant and disables all or part of the ECCS, it is normally referred to as a V-sequence. It is characterized by reactor coolant discharge outside of the containment and, if core melt occurs, the potential for radioactivity release without the benefit of containment.

The calculation surveyed 11 high-to-low-pressure piping interfaces that could serve as potential ISLOCA pathways. Of the 11 potential pathways, seven were qualitatively screened as being less likely or bounded in consequence when screened against the following three screening criteria:

1. The interface between the RCS and the lower pressure piping in an interfacing system contains any combination of at least three check and/or motor-operated valves rated at RCS pressure (as well as intervening piping).

2. The "lower pressure" piping and components are able to withstand the incursion of reactor coolant.

3. The interface reduces reactor coolant pressure to that of low pressure by passive means (such as an orifice built into a pipe), and failure to reduce pressure initiates automatic actions such as relief valve actuation and automatic isolation valve closure.

The majority (four) of these seven pathways are protected by check valves that are not vulnerable to fire-induced circuit faults or damage. Thus, internal fires do not represent a new mechanism for containment bypass. Protection of the remaining three pathways relies on either the letdown orifice (fire does not constitute a new failure mode) or on the normally closed MOVs in the RHR suction line discussed below.

The ISLOCA analysis deemed the following two pathways to be most significant:

1. RCS Piping (Loop 4 Hot Leg) to RHR Suction Line

2. RCS Piping (Loops 1-4 Cold Leg) to RHR Discharge Lines

The RHR suction line piping is isolated from the RCS Loop 4 Hot Leg by two normally closed and leak-tested, motor-operated, double-disk, gate valves (8701 and 8702) in series. These MOVs are interlocked to prevent inadvertent opening when RCS pressure exceeds 390 psig. One of these valves (8701) also has an interlock to prevent inadvertent opening with RCS temperature (as measured in the pressurizer vapor space) greater than 475 degrees F. Furthermore, the Technical Specifications require that power be removed from these valves so that they cannot be opened from the control room during operation in Modes 1 through 3. While a single three-phase hot short to the power cable of one of these MOVs might be hypothesized, it is not considered credible to simultaneously have such a circuit fault to both. Thus, this pathway is judged not to be vulnerable to fire-induced circuit faults.

Each of the four RHR cold leg discharge lines is isolated from the RCS by two check valves, which are also leak tested. These check valves are not vulnerable to fire scenario induced damage.

Finally, two additional interfaces between the RCS and low pressure systems that communicate outside containment were considered in the ISLOCA calculation:

1. between the reactor coolant pump (RCP) thermal barrier and the CCW system, and

2. through the number one RCP seal and the RCP seal return line outside containment

The first of these two pathways requires the rupture of an RCP thermal barrier cooling tube. Otherwise, the CCW system is not in direct contact with the RCS. Similarly, several valves provide isolation at containment penetrations for the CCW supply (FCV-356) to and return from the reactor vessel support coolers and RCP lube oil coolers (FCV-749 and FCV-363). While these valves are normally open, fail "as is", motor-operated valves, the CCW system is physically a closed system, not in communication with the RCS or the containment atmosphere. The probability of fire-induced damage to these isolation valves, coincident with a failure of the RCP thermal barrier or one of the other heat exchangers, is considered to be negligibly small. Thus, fire-induced damage to the control or power circuitry of these CCW isolation valves is not judged to pose a threat of radioactive material escaping the containment. It is not likely that a fire scenario would represent a new failure mechanism for this pathway.

The last potential containment bypass pathway examined is the RCP seal return line. At the containment penetration, two motor-operated isolation valves (8100 and 8112) are provided for containment isolation. During normal operation, this line is open between the RCPs and the CVCS outside containment. RCS flow through the system is limited to a very low rate (about 3 gpm per pump) by the number one RCP seal in each pump. The concern would be when the number one seal in one of more RCPs fails, allowing a higher flow into the seal return line. In addition to the two MOVs in the common return line, each RCP return line is equipped with a fail open, air-operated valve (AOV). The common return line is also equipped with a relief valve set at 150 psig with a relief

capacity of 225 gpm diverted to the pressurizer relief tank. If power is available to all the seal return isolation valves, each of the four return paths may be dismissed from further consideration based on screening Criterion 3. It is credible that fire scenarios might make power unavailable to the isolation valves. These scenarios are estimated in the following section on containment isolation.

4.7.2 CONTAINMENT ISOLATION ANALYSIS

Containment isolation systems' response to internal events was analyzed in the PRA as part of the IPE. This analysis is documented in Reference 4-30. The containment isolation function is represented by top events WL, CP, CI, and IV in the late frontline event trees. Each of these top events questions the ability of one or more penetrations to automatically isolate and remain isolated for 24 hours following an initiating event. The analysis included a qualitative review to evaluate each penetration in terms of the potential for radioactive release. The primary screening criteria for containment penetrations in the analysis are described in the following:

- Containment penetrations that are not used and are closed during power operation are not included. Exceptions include certain large containment lines (e.g., containment purge lines) and steam generator blowdown and blowdown sample lines. These lines are periodically opened during plant operation. In addition, pre-existing containment leaks are also considered in this analysis.

- Penetration lines that have no direct contact with the RCS or the containment atmosphere are not included.

- Penetration lines that are required for safety functions, and are therefore not isolated, are not included.

- Penetration lines that have a negligible failure frequency; e.g., multiple failures of three or more valves are required to fail the isolation of a penetration.

The containment penetrations selected for inclusion in the analysis were grouped into different top events according to the impacts of their failures. Since the steam generator blowdown and blowdown sample lines provide no contact with the RCS or containment atmosphere except in the event of steam generator tube rupture (SGTR), these penetration lines are modeled only for the SGTR initiating event.

Table 4.7-1 lists the valves associated with the containment isolation model in the PRA for the valves grouped into top events WL, CP, and CI. The table includes the associated support systems and failure modes. Automatic isolation of the common drain line of the reactor cavity sump and containment sump was modeled by top event WL. The automatic isolation of the containment pressure and vacuum relief line and the containment purge lines was modelled by top event CP. All valves included in top events WL and CP fail closed on a loss of power. Top event CI models the automatic isolation

of the containment penetration lines that connect to the RCS or containment atmosphere, that are permitted to be open during power operation, that have a flow area less than an equivalent 3-inch diameter hole, and that are not already included in top event WL.

A review of the valves in top event CI reveals only the following two valves pertinent to containment isolation failure concerns:

RCP Seal Water Return Isolation Valves -
CVCS 1-8100 / MOV / Inside Containment
CVCS 1-8112 / MOV / Outside Containment

The RCP seal water return line valves are normally open, fail "as is", motor-operated valves, isolating a line in communication with the RCS. The containment isolation failure analysis focused on core damage sequences initiated by fire scenarios with the potential to impact both the inboard and outboard RCP seal water return line isolation valves. A conservative bounding estimate of fire-induced containment isolation failure due to the RCP seal water return line is quantified in Section 4.7.2.1.

Additionally, fire-induced hot shorts in containment isolation AOV circuits might lead to containment isolation failure. A bounding estimate of the likelihood of containment isolation failure by this mechanism is quantified in Section 4.7.2.2. In both cases, the estimate relies on the conditional probability of core damage given a fire initiator in an identified area. The fire initiator dependent conditional probability is defined as the core damage frequency divided by the initiator frequency (see Table 4.6-1) for a given fire initiator. Individual scenario contributions to core damage frequency may then be computed by multiplying the core damage conditional probability by the scenario frequency.

4.7.2.1 Containment Isolation MOV Failure Probability

To quantify a bounding estimate of core damage with containment isolation failure due to the RCP seal water return line isolation valves, power and control circuits for the two valves were traced through the plant. Those fire areas where both the inboard and outboard isolation valve could potentially be impacted were selected for evaluation. These potentially affected fire areas were the control room, the cable spreading room, the penetration areas, and propagation scenarios affecting the relevant 480 V vital switchgear rooms. Previously screened fire scenarios that affected the fire areas identified in the circuit routing were also reevaluated for the potential to contribute to core damage with containment isolation failure. In addition to the fire areas identified by the circuit routing, two fire initiators (FS7 and FS8) involve loss of the 4-kV electric power support (through the resultant loss of 480 V power) to both isolation valves. Additionally, control room scenario VB4 includes loss of control of 4-kV electric power support from the control room. The fire scenarios that contribute to core damage initiators in the identified fire areas form the basis of this bounding analysis of containment isolation failure via the RCP seal water return line.

A human action failure rate, ZHEOI3, to manually close the outboard RCP seal water return stop valve, evaluated as part of the IPE, was applied to the quantification of fire initiated core damage sequences with containment isolation failure. This human action requires an entry into the penetration area (3-BB-100). Although the Appendix R combustible loading translates to a fire duration of no more than eleven minutes, the analysis conservatively assumes no credit for the human action for fire scenarios involving the penetration area.

A rigorous analysis of these fire scenarios for containment isolation failure would involve detailed cable routing within each fire zone and estimation of geometric factors and severity factors for the circuitry of both RCP seal water return stop valves, combined with the previously determined factors for the targets leading to core damage. Alternatively, for this bounding analysis, it was conservatively assumed that any fire leading to core damage within the fire zones identified, also disables both seal water return stop valves (equivalent to a geometric factor and severity factor of 1.0 for the RCP seal water return stop valve circuits). The use of such a conservative assumption ensures that the results will represent an upper bound estimate of core damage with containment isolation failure and account for all uncertainty associated with intrazone circuit routing.

Based on the proximity of the RCP seal water return stop valve controls to the PORV controls on Vertical Board 2 in the control room, the specific PORV-related fire scenarios were selected for quantification of core damage with containment isolation failure in the control room (VB2A, VB2B, and VB23) and in the cable spreading room (CSR2).

The contribution to core damage with containment isolation failure for each of the selected fire scenarios is computed in Table 4.7-2. Table 4.7-1 shows the RCP seal water return line isolation valves modeled as components of top event CI. Top event CI containment failures contribute to the "small, early containment failure" release category of the IPE. The bounding analysis result of 3.9E-7 in Table 4.7-2 is approximately 5 percent of the IPE result of 7.6E-6.

4.7.2.2 Containment Isolation AOV Failure Probability

Table 4.7-1 lists eight sets of containment isolation AOVs. These containment isolation AOVs fail closed on a loss of power. Two penetrations include a check valve in series with an AOV. The remaining six sets of AOVs are potentially vulnerable to fire-induced hot shorts. Simultaneous hot shorts on both the inboard and outboard AOVs could represent a containment isolation failure.

Specific circuit tracing on these AOVs was not performed. The 125 VDC system provides control power to the AOVs. Vital train separation is maintained in the battery/inverter rooms and the 480 VAC switchgear rooms. Therefore, circuits in these fire areas would only be vulnerable to propagation fire scenarios. Redundant valve circuits are coincident in the control room and the cable spreading room. Circuits most likely traverse the raceways in the Chemistry Lab and Access Control areas (4-A and 4-B) in routing from the 480 VAC switchgear rooms to the penetration area (3-BB).

To quantify a bounding estimate on the likelihood of core damage with containment isolation failure resulting from hot shorts in AOVs, fire scenarios leading to core damage in the fire areas containing these AOV control power circuits were identified. As in the case of the RCP seal water return isolation MOVs, a geometric factor and severity factor of 1.0 is assumed for the AOV circuits. The postulated AOV containment failure mechanism requires a hot short in each of the control circuits for the redundant isolation valves. As a conservative assumption, the probability of a sustained hot short (Reference 4-28) was applied to the frequency quantification only once. Due to the proximity of containment isolation valve controls on Vertical Board 1 in the control room, fire scenario VB-1 was selected to quantify the control room fire contribution. Due to the number of containment isolation valve pairs (6) and the uncertainty of cable routing through the cable spreading room, both cable spreading room fire scenarios were included in this quantification.

Table 4.7-3 presents the contributions to core damage with containment isolation failure for each location where fire-induced hot shorts in the DC control power circuits for containment isolation AOVs represent a potential failure mechanism. Table 4.7-1 shows AOVs modeled as components of top event CP, CI and WL. Top event CP containment failures contribute to the "large, early containment failure" release category of the IPE. It is conservative to assume that any AOV containment isolation failure will result in a "large, early containment failure". The bounding analysis result of 7.59E-7 is approximately one-third of the IPE result of 2.45E-6.

The results of the bounding containment isolation failure analyses for the RCP seal water return line and the containment isolation AOVs are summarized in Table 4.7-4. The total containment isolation failure due to both types of valves serves as an estimate of the frequency for the "small, early containment failure" and the "large, early containment failure" release categories. The total of 1.15E-6 is approximately 10 percent of the sum of the IPE result for early containment failures, 1.005E-5.

4.7.3 CONTAINMENT SYSTEMS PERFORMANCE

The response of containment systems performance to internal events was estimated in the IPE. An evaluation of the potential impact of fires on the CFCUs and the CS system was made to address the containment systems performance issue for the 1993 IPEEE Fire PRA. These containment systems were not explicitly modeled in the development of fire initiators. An evaluation was performed of the potential for fires to impact containment systems performance, as different from the impact of internal events. The PRA success criteria for the containment systems, following a general transient, are either of the following:

- One of two CS pumps, or
- Two of five CFCUs.

Circuit routing information was obtained for the three trains of CFCUs and both trains of the CS system. The fire zones containing circuits for these components was compared

against the fire zones that contribute to the fire initiator frequency for pertinent fire initiators. Pertinent initiators for the CS system and the CFCUs were determined based on the availability of support systems and the assumption in the model about containment systems status. Thus, the effort focused on initiators where the model assumes success of the containment systems. Initiators that result in the fire-induced failure of a support system (such as ASW and CCW, or electric power support for ASW and CCW) necessary for containment systems are not pertinent. Such failures are not different than the internal events generated guaranteed failure of CFCUs or CS.

The success criteria is ensured by the absence of any critical fire scenarios for the pertinent initiators which disable both trains of CS. Additionally, no critical fire scenarios for the initiators pertinent to CFCUs disable more than three CFCUs. Thus, there is no fire impact different from the impact of internal events on the performance of containment systems.

Table 4.7-1. List of Components - Containment Isolation System							
Top Event	Block ID	Major Components (Name and ID No.)	Failure Mode	Functional and Environmental Support Systems	Actuated Position	Initial Component State	Loss of Power Position
CI	A1	Check Valve CVCS 1-8109	Failure to close on demand (ZTVCOD). Gross leakage during operation (ZTVCOL).	—	Closed	Open	N/A
CI	A2	Motor-operated valve CVCS 1-8112	Failure to close on demand (ZTVMOD). Transfer open during operation (ZTVMOT).	480V AC Bus 1H SSPS CI ph A Train A	Closed	Open	As Is
CI	A3	Motor-operated valve CVCS 1-8100	Failure to close on demand (ZTVMOD). Transfer open during operation (ZTVMOT).	480V AC Bus 1G SSPS CI ph A Train B	Closed	Open	As Is
CI	C1	Air-operated valve LWS 1-FCV-253	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 11 (FC)* Instrument Air (FC)* SSPS CI ph A Train A	Closed	Open	Closed
CI	C2	Air-operated valve LWS 1-FCV-254	Failure to Operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 12 (FC)* Instrument Air (FC)* SSPS CI ph A Train B	Closed	Open	Closed
CI	D1	Air-operated valve LWS 1-FCV-255	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 11 (FC)* Instrument air (FC)* SSPS CI ph A Train A	Closed	Open	Closed
CI	D2	Air-operated valve LWS 1-FCV-256	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 12 (FC)* Instrument Air (FC)* SSPS CI ph A Train B	Closed	Open	Closed
CI	J1	Check valve LWS 1-60	Failure to close on demand (ZTVCOD). Gross leakage during operation (ZTVCOL).	—	Closed	Open	N/A
CI	J2	Air-operated valve LWS 1-FCV-260	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 12 (FC)* Instrument Air (FC)* SSPS CI ph A Train B	Closed	Open	Closed
CI	K1	Check Valve RCS 1-8047	Failure to close on demand. (ZTVCOD). Gross leakage during operation. (ZTVCOL).	--	Closed	Open	N/A
CI	K2	Air-operated valve RCS 1-8045	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 12 (FC)* Instrument Air (FC)* SSPS CI ph A Train B	Closed	Open	Closed
CP	E1	Air-operated valve VAC 1-FCV-662	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 11 (FC)* Instrument Air (FC)* SSPS CVI Train A	Closed	Closed	Closed

Table 4.7-1. **List of Components - Containment Isolation System**

Top Event	Block ID	Major Components (Name and ID No.)	Failure Mode	Functional and Environmental Support Systems	Actuated Position	Initial Component State	Loss of Power Position
CP	E2	Air-operated valve VAC 1-FCV-663	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 12 (FC)* Instrument Air (FC)* SSPS CVI Train B	Closed	Closed	Closed
CP	E3	Air-operated valve VAC FCV-664	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 12 (FC)* Instrument Air (FC)* SSPS CVI Train B	Closed	Closed	Closed
CP	L1	Air-operated valve VAC 1-FCV-660	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 11 (FC)* Instrument Air (FC)* SSPS CVI Train A	Closed	Closed	Closed
CP	L2	Air-operated valve VAC 1-FCV-661	Failure to operate on demand (ZTVAOD). Transfer open during Operation (ZTVAOT).	125V DC Bus 12 (FC)* Instrument Air (FC)* SSPS CVI Train B	Closed	Closed	Closed
CP	M1	Air-operated valve VAC 1-RCV-11	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 11 (FC)* Instrument Air (FC)* SSPS CVI Train A	Closed	Closed	Closed
CP	M2	Air-operated valve VAC 1-RCV-12	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 12 (FC)* Instrument Air (FC)* SSPS CVI Train B	Closed	Closed	Closed
WL	B1	Air-operated valve LWS 1-FCV-500	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC bus 11 (FC)* Instrument Air (FC)* SSPS CI ph A Train A	Closed	Open	Closed
WL	B2	Air-operated valve LWS 1-FCV-501	Failure to operate on demand (ZTVAOD). Transfer open during operation (ZTVAOT).	125V DC Bus 12 (FC)* Instrument Air (FC)* SSPS CI ph A Train B	Closed	Open	Closed

* Support system not required for WL, CI, or CP success, since the valve fails safe (closed) on failure of support system.

Table 4.7-2. Quantification of Fire Induced Core Damage with Containment Isolation Failure from RCP Seal Water Return Line Isolation MOV						
Location	Initiator	Core Damage Conditional Probability (a)	Scenario	Scenario Frequency	Failure of Human Action ZHEOI3 (b)	Core Damage with Containment Isolation Failure
Control Room	VB2A	1.2219E-5	VB2A	4.9E-3	9.301E-3	5.569E-10
Control Room	VB2B	1.0291E-5	VB2B	4.9E-3	9.301E-3	4.690E-10
Control Room	VB23	6.6318E-6	VB23	4.9E-3	9.301E-3	3.022E-10
Control Room	VB4	4.8239E-4	VB4	4.9E-3	9.301E-3	2.1985E-8
Cable Spreading Room	CSR2	6.4645E-4	CSR2	6.7E-3	9.301E-3	4.028E-8
Penetration Area	FS1	3.5126E-3	3-BB-115-FS-1 (c)	7.690E-5	1.0	2.7012E-7
Penetration Area	FS1	3.5126E-3	3-BB-100-FS-1 (c)	1.538E-5	1.0	5.402E-8
480 V Switchgear Rooms	FS7	2.0186E-3	5-A-2-FS-4	8.909E-7	9.301E-3	1.6727E-11
480 V Switchgear Rooms	FS7	2.0186E-3	5-A-3-FS-3	8.909E-7	9.301E-3	1.6727E-11
4-kV Switchgear Rooms	FS7	2.0186E-3	13-C-FS-2 (d)	3.180E-5	9.301E-3	5.9704E-10
4-kV Switchgear Rooms	FS7	2.0186E-3	13-B-FS-3 (d)	2.960E-5	9.301E-3	5.5574E-10
Turbine Deck	FS8	1.0	14-D-FS-3 (d)	1.401E-8	9.301E-3	1.303E-10
TOTAL						3.891E-07

NOTES:

(a) Core damage conditional probability is computed as the ratio of core damage frequency from a specific initiator to the initiator frequency.

(b) Human action, ZHEOI3 represents the failure rate of an operator action to manually close 8100, RCP Seal Water Return Stop Valve, Outside Containment. This action requires entry into fire zone 3-BB-100. Assuming a value of 1.0 for this failure rate serves to take no credit for this action for scenarios involving the penetration area.

(c) This fire scenario was screened out in the fire risk assessment, but has been reintroduced to consider its contribution to core damage with containment bypass.

(d) This fire scenario does not impact RCP seal water return line isolation valve circuitry. This scenario fails electric power support to both valve operator motors.

Table 4.7-3. **Quantification of Fire-Induced Core Damage with Containment Isolation Failure from Hot Shorts in Containment Isolation AOVs**

Area	Initiator	Core Damage Conditional Probability	Scenario	Scenario Frequency	Sustained Hot Short Probability	Core Damage with Containment Isolation Failure
Control Room	VB1	1.320E-3	VB1	4.9E-3	.04	2.59E-7
Cable Spreading Room	CSR1	8.439E-4	CSR1	6.7E-3	.04	2.26E-7
	CSR2	6.4645E-4	CSR2	6.7E-3	.04	1.73E-7
Battery/Inverter	FS6	6.409E-2	6A1/6A2 Propagation	2*1.83E-6 = 3.66E-6	.04	9.39E-9
480 VAC SWGR	FS6	6.409E-2	5A1/5A2 Propagation	2*8.909E-7 = 1.782E-6	.04	4.57E-9
Chem Lab / Access Control	FS6	6.409E-2	4-A(1A)	6.93E-6	.04	1.78E-8
	FS5	1.808E-2	4-A(1B)	2.772E-5	.04	2.00E-8
	FS5	1.808E-2	4-B	6.517Ee-5	.04	4.71E-8
Penetration Area	FS1	3.513E-3	3-BB-115	7.690E-5	.04	1.08E-8
	FS1	3.513E-3	3-BB-100	1.538E-5	.04	2.16E-9
	FS1	3.513E-3	3-BB-85	1.923E-4	.04	2.70E-8
TOTAL						7.97E-7

Table 4.7-4. Summary of Quantification of Fire-Induced Core Damage with Containment Isolation Failure		
Plant Area	RCP Seal Water Return Stop Valves	Containment Isolation AOVs
Control Room	2.331E-8	2.59E-7
Cable Spreading Room	4.028E-8	3.14E-7
Battery / Inverter	-----	9.39E-9
480 VAC Switchgear	3.35E-11	4.57E-9
Access Control Area	-----	8.49E-8
Penetration Area	3.24E-7	4.00E-8
4-kV Switchgear	1.15E-9	-----
Turbine Deck	1.30E-10	-----
Subtotals	3.89E-7	7.59E-7
TOTAL	1.15E-6	

4.8 TREATMENT OF FIRE RISK SCOPING STUDY ISSUES

4.8.0 EPRI Response to Fire Risk Scoping Study Issues

This section follows the EPRI guidance on the Sandia Fire Risk Scoping Study Evaluation provided as Attachment 10.5 of the "Fire-Induced Vulnerability Evaluation (FIVE)" final report (Reference 4-24). The EPRI table is reproduced here in Table 4.8-1. The following sections address each Fire Risk Scoping Study issue:

4.8.1 Seismic/Fire Interactions
4.8.2 Fire Barrier Qualifications
4.8.3 Manual Firefighting Effectiveness
4.8.4 Total Environment Equipment Survival
4.8.5 Control Systems Interactions

4.8.1 Seismic/Fire Interactions

The EPRI-suggested response to the Sandia Fire Risk Scoping Study issue related to seismic/fire interactions consists of the following three aspects:

1. Seismically Induced Fires
2. Seismic Actuation of Fire Suppression Systems
3. Seismic Degradation of Fire Suppression Systems

The IPEEE fire walkdown discussed in Section 4.2 included a seismic/fire component. This portion of the walkdown activities verified, through visual examination, the pertinent details in identified fire areas relevant to each of the three aspects identified above.

4.8.1.1 Seismically Induced Fires

The seismically induced fires aspect of the IPEEE fire walkdown focused on the potential hazards of flammable liquids or gases during a seismic event. The IPEEE walkdown team considered flammable liquids or gases within tanks, vessels, piping, cylinders, or other storage vessels that might be subject to leakage or failure.

Plant operating procedures specify that after a seismic event, a thorough inspection of all plant areas be conducted to assess and, if possible, remedy any damage that might have occurred to plant components as a result of the seismic event. Any earthquake-induced fire or potential fire hazard created by the earthquake would be identified during this inspection and, if necessary, the plant fire brigade would be dispatched to the fire. This inspection of all plant areas would be completed within two hours after a seismic event.

Bulk gas storage is not permitted inside structures housing safety-related equipment. A separate chemical and gaseous storage vault is provided for storage of hydrogen. Bulk hydrogen storage tanks are located outside, east of the auxiliary building. At the hydrogen bulk storage vault, the hydrogen system is equipped with excess flow automatic

shutoff valves which will shut off hydrogen supply if system demand exceeds 50 cfm. The valves are inherently reliable passive elements, and the light weight internals and housings would not be vulnerable to damage from a seismic event since the valves are rigidly supported in place. These valves would provide protection against and indication of a significant hydrogen leak.

To further minimize hazards from a hydrogen explosion, hydrogen lines are enclosed within a guard pipe where it runs in areas containing safety related equipment. The guard pipe is vented to the outdoors and has been pressure tested to verify that it is leak tight. The guard pipe is constructed of carbon steel piping and fittings. Hydrogen leakage in safety related areas would require failure of both the hydrogen piping and the guard piping. This could be postulated to occur only in the event of complete collapse of the piping system. If this were to occur, hydrogen flow would be sufficient to trip the excess flow valves. A small hydrogen leak (insufficient to close the trip valves) is unlikely to occur in safety related areas and, as described below, would not create a hazard.

Fire zones containing safety related equipment in which the hydrogen piping (within guard pipe) runs are as follows:

1. Fire Pump Room (fire zone 3-R) elevation 115'.
2. Penetration Areas (fire zone 3-BB), elevations 115' and 100'.
3. Auxiliary Building, elevation 100' (fire zone 3-X).

In summary, large hydrogen leaks in safety-related areas are unlikely, and if such a leak occurred, the excess flow trip valves would prevent hydrogen build up. Small hydrogen leaks in safety-related areas cannot be reasonably expected to occur, and even if such a leak were to occur, an explosive concentration of hydrogen could not build up since these areas are properly ventilated to purge any hydrogen leakage.

The IPEEE fire walkdown also focused on the storage and use of compressed gas cylinders. Procedure AP C-763, "Compressed Gas Cylinder Control," includes the following requirement for storage of flammable gases:

> "Flammable gases (hydrogen, butane, propane, acetylene, etc.) and oxygen cylinders in storage shall be separated from each other by 20 feet or by an approved 5-foot high barrier that has a 1-hour fire rating."
>
> "When cylinders are used outside of an approved designated storage location, the cylinders shall be secured in an upright position to a structural member. Cylinders shall be secured to structural members in two places (e.g., top and bottom) using an approved strap specifically made for this or 1/2" thick rope minimum."

A complete welding and open flame permit system exists and is governed by the referenced administrative procedure. Oxygen and acetylene are stored in the hot shop and warehouse areas. Fuel gases are also used routinely in the machine shop area and hot shop. The fire hazards analyses of these areas considered the contribution of fuel

gases to the overall combustible loading. Safety-related equipment is not present in any of these areas. The warehouse area and machine shop are protected by hose reels and backed up by portable fire extinguishers. Permits are required whenever welding or cutting is done outside established shop areas.

Flammable liquids containers are stored in flammable materials storage cabinets that meet the intent of NFPA 30, "Flammable and Combustible Liquids Code." Procedure OM8.ID1, "Fire Loss Prevention," includes instructions on controls for flammable liquids and temporary storage locations.

In addition to the controls and mitigating features applied to flammable gases and liquids, electrical switchgear is secured and supported such that it is unlikely to represent a seismic-induced fire hazard.

4.8.1.2 Seismic Actuation of Fire Suppression Systems

At the October 10, 1991, Advisory Committee on Reactor Safeguards (ACRS) meeting on the Long Term Seismic Program, several questions were raised regarding the PRA and the risk impact of inadvertent actuation of fire suppression systems during a seismic event. On October 11, 1991, PG&E and NRC Staff met with ACRS members to answer these questions. All questions were answered to the satisfaction of the ACRS members.

The ACRS questioned whether seismically induced inadvertent actuation of fire suppression systems had been considered in the PRA for fire water wet-pipe systems and fire water deluge systems. These systems had been considered in the PRA in the spatial interactions analysis and during the seismic walkdowns. As a result of the seismic walkdowns, it was concluded that seismically induced inadvertent actuation of these systems was not a significant contributor to risk.

In areas with safety-related equipment, wet-pipe systems are used with sprinkler heads. A fragility was developed for the sprinkler heads of the fire water wet-pipe system and was found to have a median seismic capacity greater than 10 g spectral acceleration. Therefore, it was concluded that it would be very unlikely for the sprinklers to actuate during a seismic event. Additionally, if the sprinklers did actuate, no single sprinkler can affect more than one train of safety-related equipment because of physical separation (either the trains are too far apart or located in separate compartments). Sprinkler heads deliver water at a rate of 20-30 gpm, covering an area 10-12 feet in diameter. Each sprinkler head actuates individually. Also see Section 4.3.5 for a discussion of fire water suppression system induced equipment damage.

Fire water deluge systems are used in a few select places in the plant. Specifically, DCPP Unit 1 has 11 (10 in Unit 2) deluge valves for protection of the following systems:

- turbine bearings,
- hydrogen seal oil unit,
- main feedwater pump turbines,
- lube oil reservoir,

- main and startup transformers.

These systems are actuated in a variety of ways, such as mechanical linkage or by a control system. The only PRA equipment in locations where deluge systems are used, is the startup transformer. While the startup transformer is a source of offsite power, loss of the startup transformer alone would not result in core damage. Therefore, it was concluded that the fire water deluge systems would not contribute to risk. It is judged that even if there were an inadvertent actuation, the quantity of water would not be enough to affect other equipment. If one of these deluge systems were to actuate, it could result in an initiating event, but the frequency of occurrence would be small compared to the regular initiating event frequencies for these initiators.

One goal of the IPEEE fire walkdown team was to visually verify, where possible, the train separation for safety-related equipment with respect to wet-pipe sprinkler coverage. The walkdown also served to verify the absence of fire water deluge system impact on safety-related equipment. The walkdown also served to verify the presence and distribution of drains in relation to both types of fire water systems.

4.8.1.3 Seismic Degradation of Fire Suppression Systems

NRC Regulatory Guide 1.29, "Seismic Design Classification" provides guidance for identifying and classifying structures, systems, and components which should be seismically qualified. Among other things, this Regulatory Guide specifies that non-safety-related structures, systems, and components should be seismically designed if their failure could jeopardize the functioning of safety-related components in a seismic event. Many nuclear power plants comply with Regulatory Guide 1.29 by virtue of a "Seismic II over I" program; i.e., Class II (nonsafety-related) components above Class I (safety-related) components are installed with seismic design considerations.

As a condition for the issuance of an Operating License (OL) for Diablo Canyon, PG&E implemented the Seismically Induced Systems Interaction Program (SISIP) to address this issue. During the pre-OL SISIP, extensive walkdowns of the Diablo Canyon Units 1 and 2 were performed to identify postulated seismically induced interactions created by non-safety-related sources that could potentially jeopardize safety-related targets. Approximately 3800 seismically induced interactions were identified as a result of these walkdowns. Approximately one-third of these postulated interactions were of an inconsequential nature and were documented and dispositioned by the walkdown team with no further action required. Another one-third of the postulated interactions were resolved by various engineering analyses. In some instances, the analysis was a detailed seismic qualification; in other instances, the analysis might have been an evaluation of the consequences of the postulated interaction. The remaining one-third of the identified interactions were resolved by plant modification. The modification usually provided seismic qualification to the identified source. In some instances, targets were relocated or shielded. In a few instances, interactions were resolved by revising plant operating procedures.

The results of the Program conducted prior to the issuance of OLs for Units 1 and 2, including a computer database printout listing the 3,800 postulated interactions, are documented in the ten-volume SISIP Final Report (Reference 4-31). The SISIP Final Report includes the results of the pre-OL SISIP, the program manual for the pre-OL SISIP (Appendix B), and the interactions documented by the pre-OL SISIP (Attachment 13 for the Unit 1, Attachment 16 for Unit 2).

To ensure that the objective of the SISIP is met on an ongoing basis, plant modifications and housekeeping and maintenance activities are reviewed for their potential to create seismically induced systems interactions (SISI). The SISI Manual (Reference 4-32) provides the technical guidance to design engineers to perform SISI evaluations. Specifically, fire protection system modifications are evaluated to ensure that in the event of a seismic disturbance, the fire protection system as modified will not adversely affect safety-related equipment. This may be accomplished by supporting fire system components such that the components will not endanger safety-related equipment in the area.

The SISIP defines component failure as a failure of connections, structural members, and non-structural members (the component undergoes failure as opposed to failure of the components' supports). Connections to evaluate include welded or bolted connections. Failure of structural members can result from tensile loads, and compressive loads that can cause buckling, bending, shearing, torsion, or combined loads. Failure of non-structural members includes failure of component accessories or appurtenances, equipment panels, or casings. If component failures occur, falling and/or gross deflections need to be considered. Degraded operation of the source component is not a concern unless environmental effects result from such degraded operation.

In most instances, the evaluation of failure potential component's also requires a concurrent evaluation of the component's support capability. The SISIP defines a component to include equipment, piping, ducting, raceways, tanks, panels and cabinets, and architectural features.

In addition to the failure of a component or its supports, deflection of piping is also considered in the SISIP analysis. Finally, if a component is assumed to fail or rupture, the following possible environmental effects are considered in the SISIP process:

- line break causing flooding or jet impingement
- steam line break creating a high temperature, high humidity environment
- chemical spills, such as acid, caustic or hydrazine
- hydrogen explosion
- toxic gas release
- oil spills and resulting fire
- switchgear failure resulting in explosion or fire

Fire suppression capability after a safe shutdown earthquake consists of manual hose reels and portable extinguishers. Hose reels have been provided throughout the plant so that all areas of the plant containing safety-related equipment are accessible by at least

one hose stream. Portions of the fire water system have been seismically qualified so that all hose reels in safety-related areas of the plant, with the exception of the intake structure where the safety-related equipment is enclosed within fire barriers, will be available following a safe shutdown earthquake. The qualified system consists of the 300,000 gallon fire water tank, two motor-driven fire pumps, and fire mains and piping required to provide water to the hose reel stations in safety-related areas of the plant. Cross-ties exist between the auxiliary building and the turbine building so that the fire pumps can supply water to any fire system component within the plant without the use of the yard loop. Check valves in the six yard loop feeder lines into the plant prevent water loss out of the yard loop (which could conceivably be damaged as a result of an earthquake). The check valves have normally closed, manual by-passes to ensure availability of a backup water supply for the transformer deluge systems.

The seismically qualified portion of the fire water system can be readily isolated from the rest of the fire water system. Procedure EP M-4 (Reference 4-33) instructs operators as follows:

> "Within two hours following an earthquake >0.02g, inspect all zones listed in Technical Specifications Table 3.3-11 for fires. If a portion of the Fire Protection System is earthquake damaged as ascertained by visual inspection or flow annunciator, isolate that portion from the remainder of the system (refer to Appendix 7.5 for Post-Earthquake Fire System Isolation Valve numbers and locations)."

The existing turbine building sprinkler systems can be isolated from the rest of the system by closing two valves per unit. Reactor coolant pump sprinklers can be isolated from the seismically qualified portion of the fire system by closing valves in the lines to the sprinkler systems or by closing containment fire system isolation valves inside or outside of containment. The existing auxiliary building sprinklers can be isolated by closing one valve. All sprinkler systems have flow alarms to provide control room annunciation of system actuation and/or leakage. Sufficient fire water would be available for multiple hose streams even considering the water that could be lost from breaks in nonqualified sprinkler piping prior to plant operators isolating the leaks. Backup fire protection capability is provided by three 250 gpm, portable, engine-driven fire pumps. Connections are available from the ASW (at the CCW heat exchanger) to provide suction to the portable pumps. The pump discharge can be tied into a fire main to resupply the fire water tank or to pressurize the fire system for long-term fire fighting. The portable pumps are stored in a suitable area to ensure that they will not be affected by a seismic event.

All buildings in which the qualified fire system piping is run have been reevaluated for the Hosgri earthquake and, where necessary, were strengthened as a result of the analysis. The qualified fire system piping runs in the vicinity of some non-Class I equipment in the turbine building; however, as a part of the Hosgri reevaluation, supports for major non-Class I equipment were reanalyzed and modified where necessary.

The IPEEE fire walkdown team observed examples of the following degrees of support for fire water suppression systems:

- seismically qualified
- seismically supported
- installed in accordance with NFPA 13

The IPEEE fire walkdown team verified that the various fire suppression systems installations throughout the plant did not introduce new seismically induced vulnerabilities to safety-related equipment.

4.8.2 FIRE BARRIER QUALIFICATIONS

The operability of fire barriers and barrier penetrations ensures that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment.

4.8.2.1 Fire Barriers

ECG 18.7 (Reference 4-34) and Surveillance Test Procedure (STP) M-70 (Reference 4-35) contain surveillance requirements for fire rated assemblies, which include fire barriers, fire barrier penetration seals, including cable tray, conduit and piping penetrations, sealed hatches, fire doors, fire barriers for electrical raceways, and penetrations for ventilation, including fire and smoke dampers. Fire barriers are identified and inspected in accordance with the latest revision of the fire protection fire barrier drawings. A description of the fire barriers for each fire area/zone is given in Section 9.5A of the FSAR (Reference 4-36). Deviations from the DCPP Fire Protection Program are documented in Fire Hazards Appendix R Evaluations (FHAREs) and are written in accordance with NRC Generic Letter 86-10.

4.8.2.2 Fire Doors

Fire door inspection requirements are outlined in STP M-70 as follows:

- Fire doors shall be intact, normally closed and latched properly. Additionally, their individual automatic closing mechanisms and any protecting directional spray sprinkler heads shall be operable. A fire door is considered functional if it is capable of being latched closed. Nonfunctional doors should be reported to the Fire Protection Specialist (FPS) immediately.

- Fire doors shall be exercised to verify their operability and their integrity. Any fire door without labels or signs shall be noted. Unlabeled fire doors additionally protected by directional spray sprinkler heads shall have those heads inspected for proper alignment and integrity. Automatic rolling fire doors are to be operated by activating their release mechanism. They shall remain intact and in good working order and be restored to the armed

condition following the test. Fire doors shall be documented by door number. Nonfunctional doors shall be reported to the FPS immediately.

4.8.2.3 Penetration Seal Assemblies

The penetration seal design criteria are documented in DCM S-98 (Reference 4-37) and controlled by ECG 18.7.

Penetration seal inspection requirements are outlined in STP M-70 as follows:

- Fire barrier penetrations shall be intact and be sealed with the appropriate fire retardant seal per Penetration Seal Calculations. The calculation file identifies the types of penetration seals installed at DCPP. The fire barrier penetration seal program ensures that a qualified seal is used in each penetration.

- Penetration seals shall be inspected once every 18 months from the primary fire area side for the following:

  - Degradation - shrinking, loss of adhesion to the surface, cracking, embrittlement and cell structure breakdown to a powdery substance for foam. Chipped plaster or pyrocrete. Loose damming boards or metal straps on boots.

  - Damage - gouged foam, cut boots, punctures, missing clamps or damming boards.

- Small gap formation or concave surface may be observed on the silicone foam as a result of pressure relief after completion of the foaming and thermal contraction. Air gap formations may be observed in silicone foam as the result of pressure relief after the foam is solidified. Foam seals with large air gaps are repaired in accordance with appropriate maintenance procedures and reinspected prior to acceptance. However, if gaps are deeper, then repair work is necessary. Repairs will be completed in accordance with the appropriate maintenance procedure and reinspected prior to acceptance.

In response to the concerns in NRC Information Notice 88-04, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals," (Reference 4-38), a penetration seal evaluation program was developed for DCPP units 1 and 2. All fire barrier penetration seals have been documented in penetration seal calculations. A non-conformance (Reference 4-39) has recently been initiated to address problems associated with silicone foam penetration seals.

4.8.2.4 Fire Dampers

Fire damper inspection requirements are outlined in STP M-70 as follows:

- Fire dampers shall be intact, normally open and functional, and capable of restricting air flow when tripped.

- Fire dampers are listed by number, room, duct, wall, position switch etc. All fire dampers shall be tested by disconnecting their link or actuating device, and shall be restored to the armed condition following damper testing. Where practical, the damper should be dropped with normal ventilation flow in the ductwork. Any fire damper that fails to close automatically shall be immediately reported to the FPS. Independent verification shall be performed to ensure that the damper has been properly restored.

NRC Information Notice 89-52, "Potential Fire Damper Operational Problems," (Reference 4-40) expressed concerns about the closing reliability of Ruskin curtain-type fire dampers under ventilation system operational air flow conditions. The IE Notice referenced a 10 CFR Part 21 notification to the NRC issued on November 6, 1984 by Ruskin.

In response to the 10 CFR Part 21 notification, PG&E issued design changes that included a closure test under flow conditions of those specific models of Ruskin fire dampers, identified by the vendor as being susceptible to the "closure under air flow" concern. Those dampers that failed the test were provided with closure springs and retested.

IEN 89-52 addressed the need to evaluate other curtain-type fire dampers (other than Ruskin) for the same concerns with closure under air flow. An evaluation found that there are 108 curtain-type dampers at DCPP, of which 69 are not manufactured by Ruskin. Thirteen of these 69 dampers are tested in an environment that does not simulate actual flow conditions. Therefore, procedure STP M-70, "Inspection of Fire Barrier Penetrations," was modified to simulate the actual flow conditions when the 13 curtain-type fire dampers are tested.

NRC Information Notice 83-69, "Improperly Installed Fire Dampers at Nuclear Power Plants," (Reference 4-41) addressed concerns regarding proper installation and ratings of fire dampers. An Appendix R team inspection was conducted in 1986 by the NRC Staff at DCPP to review cases where fire damper assemblies were not installed in accordance with the manufacturer's design. The Appendix R team agreed with PG&E's assessment that the subject fire damper assemblies provide an adequate level of fire safety for the areas in which they are installed.

4.8.3 MANUAL FIRE FIGHTING EFFECTIVENESS

Section III of Table 4.8-1 lists detailed attributes of an adequate fire protection program related to manual fire fighting effectiveness as determined by the EPRI evaluation for the Fire Risk Scoping Study. The scope of the evaluation includes the following:

- reporting fires
- fire brigade (makeup and equipment)

- fire brigade training
- fire brigade practice
- drills
- records

A detailed review of the Diablo Canyon Fire Protection Program against this list of attributes was performed by both the Diablo Canyon Fire Marshall and a PRA analyst. This review concluded that the Diablo Canyon Fire Protection Program meets or exceeds all attributes listed in the table. The procedure or other applicable reference for demonstrating DCPP programmatic compliance with each of these attributes is summarized in Table 4.8-2.

4.8.4 TOTAL ENVIRONMENT EQUIPMENT SURVIVAL

4.8.4.1 Potential Adverse Effects on Plant Equipment by Combustion Products

The Diablo Canyon ventilation exhaust system has been evaluated to determine the capability of removing smoke and products of combustion in the event of a fire. To enhance manual fire-fighting effectiveness, the ventilation systems either supply fresh outside air to rooms or exhaust air from rooms into a closed duct system. Ventilation exhaust capability, either manual or automatic, exists in all plant areas.

In addition to the plant ventilation exhaust system, selected doors can be opened and portable fans used to provide circulation for smoke removal. Ten electric-powered fans and one gasoline-powered fan are available in the turbine building as follows:

- Two 24-inch electric fans (10,000 cfm capacity each) and four 16-inch electric fans (5,200 cfm capacity each) are stationed at the fire brigade locker room on the 140-foot elevation of the turbine building.

- Two 24-inch electric fans, two 16-inch electric fans, and one gasoline-powered, positive pressure ventilation fan are located in the Unit 2 west buttress area.

The electric fans can be powered by any 110-V outlet. Two 9-kW gasoline-powered electric generators are available as a backup power source. The portable generators along with a selection of heavy gauge extension cords are staged in the Unit 2 west buttress area.

Appendix R fire areas or zones at DCPP are separated to ensure post-fire safe shutdown capability. For example, vital switchgear rooms are separated by train and are located in separate fire areas. Design features of the room that prevent the spread of fire to adjacent rooms (e.g. sealed penetrations, fire/smoke dampers, fire doors with limited gaps) also help prevent the spread of smoke to adjacent compartments.

Stairwells are designed to minimize smoke infiltration. They are located to provide escape and access routes for fire fighting and are enclosed by 2-hour fire walls and fire doors.

As part of the supporting analysis (Reference 4-21) for Appendix R, an evaluation of manual operator actions that may be required in a fire area (or that may require transit through an area) where a fire has occurred was performed. The analysis evaluated plant areas to determine if entering these areas was feasible, based on existing fire hazards, location of operator actions within the area, availability of detection and suppression systems, separation of redundant safe shutdown circuitry, and primary and alternative routes to an area. The evaluation determined that the manual operator actions that may need to be performed for post-fire safe shutdown (Appendix R) could be performed successfully following a postulated fire in an area that could require entry.

4.8.4.2 Spurious or Inadvertent Fire Suppression Activation

NRC Information Notice 83-41, "Actuation of Fire Suppression Systems Causing Inoperability of Safety-Related Equipment" (Reference 4-42), expressed several concerns over the inadvertent actuation of suppression systems that could damage equipment credited for safe shutdown. The conclusions of the 83-41 analysis are as follows:

- Floor drains that remove the expected amount of fire-fighting water are provided in all areas where sprinklers are located and in most areas where hose reels would be used. Due to the presence of floor drains and the drainage of water under doors and down stairwells, significant water accumulation and leakage through floor penetration seals would be minimal.

- It has been evaluated that the use of fire hoses will not cause a significant water accumulation in rooms that have floor mounted penetration seals but do not have floor drains To extinguish a fire within a room, at least one door must be open. Since water from the hose would flow through the open doorway, water accumulation within the room would be negligible.

- All water suppression systems at Diablo Canyon that protect safety-related equipment are wet pipe sprinklers. Wet pipe sprinkler systems are very reliable because they operate only after the sprinkler head fusible link has melted. Sprinkler protection is complemented by smoke detectors or flame detectors to initiate operator response. Heat detectors, which are not as susceptible to false alarms as smoke detectors or flame detectors, are used to actuate other automatic suppression systems at Diablo Canyon.

- Dry pipe deluge systems are only used on non safety-related equipment. Therefore, inadvertent fire suppression activation is not of concern.

4.8.4.3 Operator Action Effectiveness

The principal procedures governing operator actions in response to a fire are described below.

Casualty Procedure EP M-10, "Fire Protection of Safe Shutdown Equipment," (Reference 4-43), provides analyzed corrective actions to take following a fire in any plant area

containing safe shutdown equipment. This procedure provides guidance on component losses and potential manual operator actions on a fire area basis.

Procedure OP AP-8A, "Control Room Inaccessibility - Establishing Hot Standby," (Reference 4-23), provides instructions on how to achieve and maintain hot standby when operation from the control room is no longer possible due to fire, smoke, heat, ammonia, high radioactivity, explosion, credible security threat or other occurrences that make the control room uninhabitable as determined by the Shift Supervisor. This procedure would also be utilized in the event of a severe cable spreading room fire, since potential circuit damage from a cable spreading room fire could prevent control of safe shutdown components from the control room.

Procedure OP AP-8B, "Control Room Inaccessibility - Hot Standby to Cold Shutdown," (Reference 4-44), provides instructions on the transition from hot standby to cold shutdown when operation from the control room is no longer possible.

DCPP operators are periodically trained on the above procedures. Operators are also notified of revisions to procedures at training sessions that they attend every five weeks.

Self-contained breathing apparatus are provided and stored in the control room. Fire brigade personnel receive training on the proper use of this equipment. Extra air bottles and the capability to provide at least a 6-hour supply of air are available on-site.

As mentioned in Section 4.8.4.1 above, manual operator actions that may be required in a fire area where a fire has occurred have been evaluated for each fire area. The evaluation determined that the operator actions could be performed successfully.

4.8.5 CONTROL SYSTEMS INTERACTIONS

A thorough review of the DCPP safe shutdown analysis was performed in 1991 and 1992 under the Appendix R Documentation Enhancement Project. The review included a detailed review of the impact of a control room or cable spreading room fire on the ability to safely shutdown. The review identified a nonconformance with Appendix R requirements in the design of the control circuitry for the 4-kV pumps and the DGs.

The Appendix R analysis review determined that a fire in the control room or the cable spreading room for DCPP Units 1 and 2 could damage DC control circuitry for 4-kV pumps, potentially resulting in the loss of electrical control of the pumps from the hot shutdown panel or the 4-kV switchgear. A similar concern was identified with the control circuitry for the DGs, in that a postulated control room or cable spreading room fire could disable local starting and loading of the DGs. These non-conformances were reported under Licensee Event Report (LER) 2-92-001 (Reference 4-45).

Design changes were issued and implemented during refueling outages 1R5 and 2R5 to provide the necessary circuit isolation to ensure that control of 4-kV pumps and DGs remained available at their respective remote control stations following a postulated control room/cable spreading room fire.

As a corrective action of the non-conformance, an additional detailed review was performed to determine if similar conditions existed for other plant components that could adversely impact the ability to safely shutdown following a postulated fire. This review determined that there were no similar circuit isolation deficiencies that could adversely impact post-fire safe shutdown. The post-fire safe shutdown analysis that documents the ability to safely shutdown following a postulated fire in the control room/cable spreading room is documented in Calculation M-928 (Reference 4-21).

NRC Information Notice 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire" (Reference 4-46), addresses problems that could arise if a control room fire forced reactor operators to evacuate the control room. It was found that a fire in the control room may cause hot shorts, which when combined with the absence of thermal overload protection, could result in valve damage before the operator is able to shift control of the valves to the remote/hot shutdown panel. Control of the valves was lost because their thermal overload protection bypassed, and their torque and limit switches are wired upstream of the control room/remote shutdown panel.

PG&E investigated the effects of a control room fire at Diablo Canyon. It was determined that DCPP does not have similar problems with MOVs as described in NRC Notice 92-18. This is because the subject valves have torque and limit switches in their control circuits that are wired downstream of the transfer relay contacts in the motor control center. Therefore, fire induced faults including a hot short will not circumvent the function of the torque and limit switches or prevent valve operation subsequent to the valve control transfer to the hot shutdown panel.

Table 4.8-1. Attributes of Adequate Fire Protection Program

SANDIA FIRE RISK SCOPING STUDY EVALUATION

I. SEISMIC/FIRE INTERACTIONS

1. Seismically Induced Fires:

   As part of the seismic assessment walkdown, verify hydrogen or other flammable gas or liquid storage vessels in areas with seismic safe shutdown or safety-related equipment are not subject to leakage under seismic conditions. Examples would be improperly anchored hydrogen or oxygen bottles, hydrogen tanks used for primary coolant chemistry control, etc.

2. Seismic Actuation of Fire Suppression Systems:

   As part of the seismic assessment, verify that the design of water suppression system considers the effects, if appropriate, of inadvertent suppression system actuation and discharge on that equipment credited as part of the seismic safe shutdown path in a margins assessment that was not previously reviewed relative to the internal flooding analysis or concerns such as discussed in NRC I & E Information Notice 83-41.

3. Seismic Degradation of Fire Suppression Systems

   As part of the Seismic assessment walkdown, verify fire suppression systems have been structurally installed in accordance with good industrial practice and reviewed for seismic considerations such that suppression system piping and components will not fall and damage safe shutdown path components nor is it likely that leaking or cascading of the suppressant will result.

II. FIRE BARRIER QUALIFICATIONS

Fire Barriers

1. Fire barriers and components such as fire dampers, fire penetration seals and fire doors for fire barriers considered in the FIVE Methodology are included in the plant surveillance program.

Fire Doors

2. A fire door inspection and maintenance program.

Penetration Seal Assemblies

3. A penetration seal inspection and surveillance program.

4. Fire barrier penetration seals have been installed and maintained to address concerns such as those identified in NRC Information Notice No. 88-04

Fire Dampers

5. An inspection and maintenance program for fire dampers

6. Damper installations address concerns such as those identified in NRC Information Notice No. 89-52, "Potential Fire Damper Operational Problems," dated June 8, 1989 and NRC Information Notice No. 83-69, "Improperly Installed Fire Dampers at Nuclear Power Plants," dated October 21, 1983.

Table 4.8-1. Attributes of Adequate Fire Protection Program

SANDIA FIRE RISK SCOPING STUDY EVALUATION

III. MANUAL FIRE FIGHTING EFFECTIVENESS

Reporting Fires

1. Appropriate plant personnel knowledgeable in the use of portable fire extinguishers.
2. Portable extinguishers located throughout the plant.
3. A plant procedure for reporting fires in the plant.
4. A plant communication system that includes contact to the control room.

Fire Brigade

1. A fire brigade made up of at least 5 trained people on each shift?
2. The brigade leader and at least two other brigade members on each brigade shift are knowledgeable in plant systems and operations?
3. Each Brigade member receives an annual review of physical condition to evaluate his ability to perform fire fighting activities?
4. Minimum equipment provided for the brigade includes the following:
    a. Personal protective equipment such as SCBA, turnout coats, boots, gloves, and hard hats.
    b. Emergency communications equipment.
    c. Portable lights.
    d. Portable ventilation equipment
    e. Portable extinguishers

Fire Brigade Training

5. Brigade members receive an initial classroom instruction program consisting of the following:
    a. Review of the plant fire fighting plan and identification of each individual's responsibilities.
    b. Identification of typical fire hazards and associated types of fires that may occur in the plant.
    c. Identification of the location of fire fighting equipment and familiarization with the layout of the plant including access and egress routes.
    d. The proper use of available fire fighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays and fires involving flammable and combustible liquids and gases.
    e. The proper use of communication, lighting, ventilation, and emergency breathing equipment.
    f. Fighting fires inside buildings and confined spaces.
    g. Review of fire fighting strategies and procedures.

Table 4.8-1. Attributes of Adequate Fire Protection Program

SANDIA FIRE RISK SCOPING STUDY EVALUATION

<u>Practice</u>

6.	Fire brigade members receive hands-on structural fire fighting training at least once per year to provide experience in actual extinguishment and the use of emergency breathing apparatus.
<u>Drills</u>	
7.	Fire brigade drills are performed in the plant so that each fire brigade shift can practice as a team.
8.	Drills performed at regular intervals for each shift fire brigade.
9.	At least one unannounced fire drill for each shift fire brigade performed per year.
10.	At least one drill per year performed on a "backshift" for each shift fire brigade.
11.	Drills pre-planned to establish training objectives and critiqued to determine how well the training objectives have been met?
12.	At least triennially, and unannounced drill is performed for and critiqued by qualified individuals independent of the licensee's staff.
13.	Pre-fire plans are developed for safety-related areas of the plant (as a minimum).
14.	The pre-fire plans are updated and used as part of the brigade training.
15.	Fire brigade equipment is maintained.
<u>Records</u>	
16.	Records are provided for each fire brigade member demonstrating the minimum level of training and refresher training has been provided.

Table 4.8-1. Attributes of Adequate Fire Protection Program

SANDIA FIRE RISK SCOPING STUDY EVALUATION

IV. TOTAL ENVIRONMENT EQUIPMENT SURVIVAL

Potential Adverse Effects on Plant Equipment by Combustion Products

1. The FIVE methodology does not currently provide for an evaluation of non-thermal environmental effects of smoke on equipment see Section 4.2.2.

2. However, be aware of and sensitive to potential impact of smoke and products of combustion on human performance in safe shutdown operations in application of FIVE.

Spurious or Inadvertent Fire Suppression Activation

1. Verify that the design of fire suppression systems considers the effects, if appropriate, of inadvertent, suppression system actuation and discharge on equipment credited for safe shutdown for concerns such as those discussed in NRC I & E Information Notice, 83-41.

Operator Action Effectiveness

1. There are safe shutdown procedures identifying the steps for planned shutdown when necessary in the event of a fire.

2. Operators receive training on these procedures.

3. If in performance of these procedures operators are expected to pass through or perform manual actions in areas that may contain fire or smoke suitable SCBA equipment and other protective equipment are available for operators to perform their function.

V. CONTROL SYSTEMS INTERACTIONS

1. Safe shutdown circuits are physically independent of, or can be isolated from, the control room for a fire in the control room fire area.

Table 4.8-2. Manual Fire Fighting Effectiveness

EPRI Response Question	Applicable DCPP Reference
1. Appropriate plant personnel knowledgeable in the use of portable fire extinguishers.	<u>AP B-51</u>, Appendix A contains a list of courses which constitute the Fire Brigade Training Program. Several of these courses apply to the use of portable fire extinguishers: EFD 811 Fire Chemistry, Classification & Extinguishing Agents. EFD 813 Water, Gas, and Foam Application Techniques EFD 815 Dry Chemical Portable Extinguisher Application Techniques <u>NPAP B-13</u>, describes initial and requal training for members of the Shift Fire Brigade to include: 4.4.1 a. Training shall be given to new fire brigade members to familiarize them with the location and operation of fire protection and suppression equipment... 4.4.2 a. Use of different types of fire protection, fire fighting, and rescue equipment provided. b. Actual operation of portable fire extinguishers, fire hoses, and breathing apparatus. Further, 4.5 Members of the Assistant Fire Brigade, and other employees shall receive periodic training/retraining in fire protection. As a minimum, this training should include: 4.5.2 The instruction relating to fire prevention and suppression. 4.5.3 The location and use of plant fire fighting equipment and the limitations placed upon this equipment. Further, 4.8 ...Fire Watch training shall include classroom instruction and hands on use of portable fire extinguishers. Specifically, the "Fire Watch" is responsible for knowing: 4.8.3 The proper extinguisher to use as applicable to the fire loading.
2. Portable extinguishers located throughout the plant.	Emergency Procedure, <u>EP M-6</u>, "Fire," includes Fire Fighting Preplans for each area of the plant. Locations of portable fire extinguishers as well as hose reels and sprinklers are detailed under the heading, "Fire Suppression Equipment" for each Preplan.

Table 4.8-2. Manual Fire Fighting Effectiveness	
EPRI Response Question	Applicable DCPP Reference
3. A plant procedure for reporting fires in the plant.	IDAP OM8.ID1 addresses fire reporting responsibilities of all plant personnel in Step 4.6 as follows: "All remaining plant personnel shall be instructed to report any fire to the control room describing the extent of the fire, its location and the potential for further damage. Only after reporting the fire should the individual attempt to extinguish it or limit its spread and then only if the individual has been trained in fire fighting." FSAR Appendix 9.5H 5. Fire Detection by Plant Personnel a. Reporting of fires takes precedence over fighting a fire. Only personnel who are trained in the use of fire fighting equipment may attempt to suppress a fire. b. The fire alarm signal system is the normal way to report a fire. A direct call by telephone or radio to the control room may be utilized in some instances. General Employee Training, GET All employees with site access or protected area access receive General Employee Training through courses GAAA-100, "Site Access Handbook," or GPAA 100, "Protected Area Access," respectively. Each of these classes contain a segment devoted to Industrial Safety and Fire Protection. Administrative responsibilities for plant personnel are delineated to include the following: "It is the responsibility of all plant employees to immediately report all fires, assist fire brigade personnel as directed in a fire fighting effort, report all fire hazards to their supervisor, work safely and in such a manner as not to create a fire hazard and to be acquainted with the "Fire Protection Plan" and the use and location of emergency equipment."

Table 4.8-2. Manual Fire Fighting Effectiveness	
EPRI Response Question	Applicable DCPP Reference
	This course material further details the actions involved in Fire Detection by Personnel "Reporting of fires should take precedence over fighting a fire. Only personnel who are trained in the use of fire fighting equipment should attempt to suppress a fire." The fire alarm signal system will be the normal way to report a fire. A direct call by telephone or radio to the control room may be utilized in some instances. (The fire alarm signal system can be actuated from any Company telephone in the plant by dialing 779. Report any fire to the Shift Foreman even if you do not activate the fire code. IDAP OM8.ID1 further states (step 4.5) responsibilities of the "Fire Watch" to include the following: 4.5.2 Notifying the "Shift Foreman" of fires, and sounding the fire alarm if necessary. 4.5.2 Extinguishing fires when obviously within the capability of the equipment available and his training. IDAP OM8.ID1 "Fire Loss Prevention," in addressing responsibilities of the Fire Marshal includes Step 3.4.8 as follows: Documenting fires and other related incidences as to, but not limited to: a. Actual or proximate causes, if determinable. b. Impact of fire or other related incident on the plant at the time of the incident. c. Effectiveness of fire prevention methods and fire protection systems and equipment provided and available at the time of the incident (i.e. - sprinklers extinguished, extinguisher or fire hose used to extinguish fire). d. Effectiveness of the Fire Brigade teams responding.
4. A plant communication system that includes contact to the control room.	Emergency Procedure, EP M-6, "Fire," includes Fire Fighting Preplans for each area of the plant. For each preplan, information is provided, detailing available communications equipment in the area. EP M-6, "Fire," Section 2.0, ("SYMPTOM OR ENTRY CONDITION") addresses specific operator responses to the reporting of a fire. EP M-6 contains detailed instructions on the use of the Control Room Fire Phone in Attachment 5.4.

Table 4.8-2. Manual Fire Fighting Effectiveness	
EPRI Response Question	Applicable DCPP Reference
1. A fire brigade made up of at least 5 trained people on each shift.	FSAR Appendix 9.5H B. Fire Brigade Organization and Responsibilities "There is one Fire Brigade on each shift providing continuous response capability. The fire brigade personnel may have no other fire emergency responsibilities that would prevent them from performing fire brigade duties. (1) Leader Fire Brigade Trained (4) Crew Member Assigned Operator or Trained Member NOTE: The Fire Brigade Leader is a member of the Fire Brigade that is a trained Fire Brigade Leader and has been designated as such by the Plant Fire Marshall. A licensed Operator will accompany the Fire Brigade Leader in all Emergency Responses, unless the Licensed Operator is the Fire Brigade Leader.
2. The brigade leader and at least two other brigade members on each brigade shift are knowledgeable in plant systems and operations.	All fire brigade members are Operators. The fire brigade leader is a Senior Control Operator. The fire brigade is supplemented by an Industrial Fire Officer. The IFO is a professional career fireman dedicated to each shift.
3. Each brigade member receives an annual review of physical condition to evaluate his ability to perform fire fighting activities.	IDAP OM14.ID2, "Medical Examinations," details specific requirements (including a reference to NFPA 600) for annual medical examinations for members of fire brigades.

Table 4.8-2. Manual Fire Fighting Effectiveness	
EPRI Response Question	Applicable DCPP Reference
4. Minimum equipment provided for the brigade includes the following: a. Personal protective equipment such as SCBA, turnout coats, boots, gloves, and hard hats. b. Emergency communications equipment. c. Portable lights. d. Portable ventilation equipment. e. Portable extinguishers.	(Government Industrial Safety Order Title 8 Article 10.1) (NFPA 600, FSAR Appendix 9.5H) <u>NFPA 600</u> - Chapter 6 Equipment (a) Portable Fire Extinguishers. (b) Hose and Hose Accessories. (c) Portable Lighting Equipment. (d) Forcible Entry Tools. (e) Ladders (f) Salvage and Overhaul (g) Respiratory Protective Equipment. (h) Rescue and First Aid Equipment (i) Special Purpose Equipment (j) Personnel Protective Equipment

Table 4.8-2. Manual Fire Fighting Effectiveness	
EPRI Response Question	Applicable DCPP Reference
5. Brigade members receive an initial classroom instruction program consisting of the following: a. Review of the plant fire fighting plan and identification of each individual's responsibilities. b. Identification of typical fire hazards and associated types of fires that may occur in the plant. c. Identification of the location of fire fighting equipment and familiarization with the layout of the plant including access and egress routes. d. The proper use of available fire fighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays, and fires involving flammable and combustible liquids and gases. e. The proper use of communication, lighting, ventilation, and emergency breathing equipment. f. Fighting fires inside buildings and confined spaces. g. Review of fire fighting strategies and procedures.	("Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance", NFPA Standard 600, FSAR App 9.5H, AP B-51) <u>FSAR Appendix 9.5H</u> Fire Brigade Training - Plant Technical Specifications, and State Regulations require periodic training of Fire Brigade members. The training program utilizes classroom instruction, practice in fighting typical fires, and fire drills. Training is conducted on a continuing basis. Training sessions are designed such that all areas are completed every two years for Fire Brigade members. FRSS training scope items a-g are listed in the FSAR as follows: FRSS a. = FSAR d. FRSS b. = FSAR a. FRSS c. = FSAR b. FRSS d. = FSAR c. FRSS e. = FSAR k. FRSS f. = FSAR g. FRSS g. = FSAR h. <u>AP B-51</u> - APPENDIX A - Fire Brigade Training Program EFD 811 Fire Chemistry, Classification & Extinguishing Agents EFD 813 Water, Gas, and Foam Application Techniques EFD 815 Dry Chemical Portable Fire Extinguisher Application Techniques EFD 821 Fire Prevention, Detection and Personnel Safety Equipment (SCBA) EFD 824 Suppression of Gas and Oil Fires EFD 825 Suppression of Electrical and Wildland Fires EFD 826 Limiting Fire Damage and Suppression of Radiological Fires EFD 831 DCPP Fire Protection Plan, Fire Brigade and Offsite Fire Response EFD 840 Fire Brigade Leader Training EFD 850 Programmed Practical Fire Fighting EFD 900 Fire Drill Participation

Table 4.8-2. Manual Fire Fighting Effectiveness	
EPRI Response Question	Applicable DCPP Reference
6. Fire brigade members receive hands-on structural fire fighting training at least once per year to provide experience in actual fire extinguishment and the use of emergency breathing apparatus.	Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance". 2.0 Practice Practice sessions should be held for fire brigade members on the proper method of fighting various types of fires of similar magnitude, complexity, and difficulty as those which could occur in a nuclear power plant. These sessions should provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions. These practice sessions should be provided at regular intervals but not to exceed 1 year for each fire brigade member. FSAR Appendix 9.5H 4. PRACTICE A practice programmed fire shall be initiated annually to provide experience in the art of bringing fires under control and the use of fire fighting equipment and self-contained breathing apparatus under strenuous fire fighting conditions. AP B-51 - "Industrial Safety and Fire Protection Training" D - SCHEDULE FOR FIRE BRIGADE TRAINING 3. Practical Fire Fighting EFD 850 a. Initial training for fire brigade members, shall be an 8 hour hands on training session that includes proper method of fighting various types of fires of similar magnitude, complexity and difficulty as those which could occur at DCPP. b. Requal training sessions shall be scheduled once a year to allow attendance by all fire brigade members. E - Fire Brigade Member Requalification 3. Practical Fire Fighting a. All fire brigade members are required to participate in the once a year practical fire fighting session. The Fire Marshal may give a special training session to fire brigade members who become deficient in this annual training. The fire marshal may waive a brigade member's participation in one practical session if the member has participated in at least two previous annual sessions and has demonstrated proficiency by an oral examination. The waiver must be documented. b. Annual is defined in this procedure as no later than 16 months after the last training was accomplished.

<table>
<tr><th colspan="2">Table 4.8-2. Manual Fire Fighting Effectiveness</th></tr>
<tr><th>EPRI Response Question</th><th>Applicable DCPP Reference</th></tr>
<tr><td>7.

Fire brigade drills are performed in the plant so that each fire brigade can practice as a team.

8.

Drills performed at regular intervals for each shift fire brigade.

9.

At least one unannounced fire drill for each shift fire brigade performed per year.

10.

At least one drill per year performed on a "backshift" for each shift fire brigade.</td><td>("Nuclear Plant Fire Protection Functional Responsibilities..."
(FSAR Appendix 9.5H)

<u>FRAC-QA</u>
3.0 Drills
Fire Brigade drills should be performed in the plant so that the fire brigade can practice as a team. Drills should include the following:
a. Assessment of fire alarm effectiveness, time required to notify and assemble fire brigade, and selection, placement and use of equipment.
b. Assess each brigade member's knowledge of his role in the fire fighting strategy for the area assumed to contain the fire. Assess the brigade members conformance with established plant fire fighting procedures and use of fire fighting equipment, including self-contained breathing apparatus, communication

<u>AP B-51</u> - Industrial Safety and Fire Protection Training
D - SCHEDULE FOR FIRE BRIGADE TRAINING
2. Fire drills EFD 900 shall be conducted at least quarterly for each shift fire brigade. Each fire brigade member shall participate in two drills per year. There shall be at least one series of unannounced drills per year and/or at least one series of backshift drills per year.
E - FIRE BRIGADE MEMBER REQUALIFICATION
2. Drills
a. All fire brigade members that are required to participate in quarterly drills shall, within six months from the time of the last drill, requalify.
b. All fire brigade shifts that are required to participate in the back shift and/or unannounced drills shall not exceed 18 months from the last requalification.

<u>FSAR Appendix 9.5H</u>
3. FIRE BRIGADE DRILLS
a. Fire Brigade drills are conducted quarterly.
b. The drills are conducted on the plant site in areas containing significant fire hazards where similar fires of that type, size and arrangement could reasonably occur.
c. Drills are conducted so that each Fire Brigade member can participate. Each Brigade member should participate in at least one drill per year.
d. At least one drill per year is performed on a back-shift.
e. At least one drill per year for each Fire Brigade is unannounced.
f. Drills will be observed by supervisory personnel to:
1) Assess the effectiveness of the notification systems and times for the response of the Fire Brigade and their selection and use of equipment.
2) Assess the individual Fire Brigade member's knowledge of his responsibilities, conformance with established procedures and the use of fire fighting and other emergency equipment to the extent practicable.
3) Assess the Fire Brigade Leader's effectiveness in direction of the fire fighting effort.
4) Assess the overall effectiveness of the drill to determine if the training objectives are being met.
g. Two drills per year shall involve a coordinated response involving the plant Fire Brigade and offsite fire protection agencies.</td></tr>
</table>

Table 4.8-2. Manual Fire Fighting Effectiveness	
EPRI Response Question	Applicable DCPP Reference
11. Drills pre-planned to establish training objectives and critiques to determine how well the training objectives have been met?	(AP B-51) AP B-51 EFD 900, Active Participation in Fire or Fire Drill At the completion of the drill, the Fire Marshal should prepare a "Training Session Record" form including an evaluation report for entry into the training records system. FSAR Appendix 9.5H 3. FIRE BRIGADE DRILLS a. Fire Brigade drills are conducted quarterly. b. The drills are conducted on the plant site in areas containing significant fire hazards where similar fires of that type, size and arrangement could reasonably occur. c. Drills are conducted so that each Fire Brigade member can participate. Each Brigade member should participate in at least one drill per year. d. At least one drill per year is performed on a back-shift. e. At least one drill per year for each Fire Brigade is unannounced. f. Drills will be observed by supervisory personnel to: 1) Assess the effectiveness of the notification systems and times for the response of the Fire Brigade and their selection and use of equipment. 2) Assess the individual Fire Brigade member's knowledge of his responsibilities, conformance with established procedures and the use of fire fighting and other emergency equipment to the extent practicable. 3) Assess the Fire Brigade Leader's effectiveness in direction of the fire fighting effort. 4) Assess the overall effectiveness of the drill to determine if the training objectives are being met. g. Two drills per year shall involve a coordinated response involving the plant Fire Brigade and offsite fire protection agencies.
12. At least triennially, an unannounced drill is performed for and critiques by qualified individuals independent of the licensee's staff.	(NRC Generic Letter 82-21) Generic Letter 82-21 provides guidance on meeting the Technical Specification requirements for an independent triennial fire protection audit. Enclosure 3 lists minimum elements of such an audit. Item 7 of the scope section reads as follows: 7. (The audit should verify that) Plant response to fire emergencies is adequate by analyzing incident records and witnessing an unplanned fire drill.
13. Pre-fire plans are developed for safety-related areas of the plant (as a minimum).	(EP M-6) Attachment 5.1 of EP M-6 contains Fire Fighting Preplans for each area of the plant.

Table 4.8-2. Manual Fire Fighting Effectiveness	
EPRI Response Question	Applicable DCPP Reference
14. The pre-fire plans are updated and used as part of the brigade training. (EP M-6, FSAR Appendix 9.5H)	Attachment 5.1 of EP M-6 contains Fire Fighting Preplans for each area of the plant. Each Preplan carries a revision number to reflect updates. FSAR Appendix 9.5H details classroom instructions for Fire Brigade Training to include the following: g. The proper method for fighting fires in various plant locations including confined spaces. h. Fire fighting procedures and strategies including recent changes. i. Plant modifications that have a significant impact on fire protection.
15. Fire brigade equipment is maintained.	FSAR Appendix 9.5H FIRE EQUIPMENT INSPECTION AND MAINTENANCE To meet the plant license requirements, CAL-OSHA and Nuclear Mutual Limited requirements, fire equipment for the plant is inspected and maintained on a routine basis. IDAP OM8.ID1 includes in the responsibilities of the Fire Marshal and/or Fire Protection Specialist: 3.4.11 Verifying that all fire suppression equipment available for emergency response is properly maintained per NFPA Standards and Cal-OSHA requirements.
16. Records are provided for each fire brigade member demonstrating the minimum level of training and refresher training has been provided.	IDAP OM8.ID1 includes the following instruction: 4.1.1 A Fire Brigade Training Program shall be performed, and the training program maintained, in accordance with Reference 8.16 of this procedure. This training program shall be reviewed by the PSRC and approved by the Plant Manager. No reduction of the requirements of the training program shall be made unless they are specifically reviewed by the PSRC. NPAP B-13, "Qualification and Training Requirements of Plant Personnel Specifically Concerned With Fire Loss Prevention", Step 4.10: Fire protection training records, including critiques of drills and hands on training sessions, shall be maintained and processed as described in Nuclear Plant Administrative Procedure NPAP B-2 and supplements. Critiques of training sessions, such as drills and hands on training, shall be included with the fire protection training records.

4.9 USI A-45 and other Safety Issues

4.9.1 Safety Issues Background

NUREG-1407 describes NRC programs related to internal fires. These programs are summarized below.

- USI A-45, "Shutdown Decay Heat Removal Requirements," was initiated to determine if the decay heat removal function at power plants is adequate, and if cost-beneficial improvements could be identified. USI A-45 was subsumed in the IPE Reference 4-6); the adequacy of the decay heat removal system should be addressed as part of the fire IPEEE evaluation. The evaluation is described below.

- GI-57, "Effects of Fire Protection System Actuation of Safety-Related Equipment," assesses the impact of inadvertent actuation of fire protection systems on safety systems; it is one of the issues identified in the Fire Risk Scoping Study. An examination of the effects of fire protection actuation on safety-related equipment is discussed in Section 4.3.5 and 4.8.4 of the IPEEE report.

- NUREG/CR-5088, "Fire Risk Scoping Study," identifies several fire issues that may not have been addressed in previous fire PRAs. Section 4.8 of this report addresses the Fire Risk Scoping Issues.

4.9.2 Decay Heat Removal Evaluation

As part of the Diablo Canyon IPE study, the adequacy of the decay removal capabilities at DCPP for internal initiating events was demonstrated. The NRC Staff Evaluation (Reference 4-46) concluded that

> "Based on the licensee's IPE process used to search for DHR vulnerabilities, and review of Diablo Canyon plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI A-45 (Decay Heat Removal Reliability) resolution."

This section assesses the decay heat removal capabilities at DCPP in response to fire events. This evaluation considers the systems and operator actions required to remove decay heat in the 24-hour mission time following a fire initiating event during full-power operation. At DCPP, decay heat is removed by several different systems and methods, depending on the type of transient.

Fire initiating events result in general transients or small LOCAs; fire-initiated medium LOCAs or large LOCAs are not deemed credible. During general transients and small LOCAs (whether fire-initiated or not), the main feedwater system and the AFW system supply water to the steam generators, thereby providing cooling to the RCS. After reactor trip, the AFW system provides the normal secondary heat sink with recirculation back to the condenser through the 40 percent steam dump valves, or to the atmosphere through the 10 percent atmospheric steam dump valves. If the AFW system fails to supply the

secondary heat sink, the operators are instructed to attempt to restore the secondary heat sink; either by reestablishing main feedwater, or by depressurizing the steam generators and providing feedwater from the condensate system. If the secondary heat sink is lost, decay heat can be removed from the RCS with bleed and feed cooling (i.e., feeding with centrifugal charging pumps and safety injection pumps, bleeding through the PORVs). When the RCS has cooled and pressure has decreased sufficiently, the RHR system can remove decay heat through the RHR heat exchangers, which transfer heat to the CCW system.

The AFW system is equipped with three independent trains, each capable of satisfying 100 percent of the decay heat removal needs after reactor trip (Reference 4-47). The auxiliary feedwater system contains two motor-driven pumps and one turbine-driven pump. The two motor-driven pumps are contained in one fire area; the impact of fires failing both motor-driven AFW pumps is represented by the initiating event FS1 (Table 4.6-4), with a core damage frequency of 1.5E-6 per year, or 5.6 percent of the total fire core damage frequency. If the two motor-driven AFW pumps fail to run as a result of an internal fire, the turbine-driven AFW pump is still available to provide decay heat removal capability. The failure of the auxiliary feedwater system from all causes, including loss of all support, contributes approximately 6 percent to the IPEEE 1993 Fire PRA core damage frequency, as shown in Table 4.9-1.

The main feedwater system and the condensate system are backup sources of feedwater to provide a secondary heat sink. Each unit contains two turbine-driven main feedwater pumps and three motor-driven condensate pump sets. The Diablo Canyon PRA does not take credit for their operation (for either internal or external events). If the main feedwater and condensate systems were credited in the model, the importance of the AFW system would be decreased.

The operator actions required to initiate and maintain bleed and feed cooling are proceduralized in DCPP functional response procedure FR-H.1. Table 4.9-1 indicates that the failure of bleed and feed cooling, from all causes, contributes approximately 4 percent to the IPEEE 1993 Fire PRA core damage frequency. The PRA model is conservative, in that the procedural actions to open the reactor vessel head vents and depressurize one steam generator to atmospheric pressure are not credited.

The RHR system contains two 100 percent capacity, independent trains, each containing a motor-driven pump and heat exchanger. The RHR pumps can be aligned to take water from the refueling water storage tank, the hot leg of RCS loop 4, or the containment recirculation sump. Table 4.9-1 shows that for fire initiating events, the failure of the systems or operator actions involved in operating the RHR system do not contribute significantly to the core damage frequency.

Decay heat removal insights were discussed in Appendix 5 of NRC Generic Letter 88-20 (Reference 4-48). The decay heat removal systems at DCPP do not exhibit any vulnerabilities with respect to these concerns as a result of fire initiating events. As discussed previously, equipment redundancy is clearly evident in each of the decay heat removal systems. The motor-driven AFW pumps are located in a separate fire zone from

the turbine-driven AFW pump room. The centrifugal charging pump room is spatially separated from the safety injection pump room. Each RHR pump is located in a separate room. Additionally, the electrical power support systems (switchgear, batteries, diesel generators, etc.) are spatially separated on a train basis.

In summary, the results of the IPEEE 1993 Fire PRA indicate no vulnerabilities at DCPP with regard to decay heat removal.

Table 4.9-1. Importance Evaluation for the Fire PRA Decay Heat Removal Evaluation			
System or Function	Top Event	Description/Comments	Percentage of CDF in which Event is Failed
Auxiliary Feedwater	AW	Auxiliary Feedwater - Losses from all causes AW4 - Support for MDPs Unavailable AWF - No support available FSH1 - Human action to control AFW from hot shutdown panel following control room fire	5.8 4.5 1.3 < .1
Bleed and Feed Cooling and High Pressure Injection	OB	Failure of Bleed and Feed Cooling - Losses from All Causes OB1 - Support Available OBF - No Support Available FML3 - Bleed and Feed Capability following control room fire	3.6 2.0 1.5 < .1
	CH	Charging Pumps - Support Available	< .1
	SI	Safety Injection Pumps - Support Available	< .1
Residual Heat Removal	RF	Operator Switches to Recirculation Mode - Support Available	.3
	LA	RHR Pump Train A - Support Available	.3
	LB	RHR Pump Train B - Support Available	.3
	LV	RHR Suction from RWST	< .1
	RW	RWST	< .1

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5. HIGH WINDS, FLOODS, AND OTHERS

5.0 INTRODUCTION

A probabilistic evaluation of the impact of "other" external initiating events (i.e., hazards other than fire and seismic events) on the Diablo Canyon Power Plant (DCPP) was performed as part of the Diablo Canyon Probabilistic Risk Assessment (DCPRA, Appendix F, Reference 5-1). Other external events, which were analyzed in Reference 5-1, included such causes as external flood, aircraft crashes, hazardous chemical, high winds, etc. The methodology utilized for the IPEEE for "other" external events was based on the analysis documented in Reference 5-1. The "other" external events analysis was updated for the IPEEE (Reference 5-2); included in the update was a review of the NRC Standard Review Plan (SRP) Compliance Checklist for DCPP (Reference 5-3), a review of other plant documentation, design changes, design criteria memoranda, calculations and a confirmatory walkdown.

The DCPRA was completed in July 1988 and included an evaluation of a number of external event hazards. The approach was to determine a conservative estimate of possible hazard sources and occurrence frequencies from the Diablo Canyon Power Plant Final Safety Analysis Report (FSAR) Update (Reference 5-5) or other data. Then, the specific plant facilities and components that may be subject to damage were identified and assigned a conditional failure probability. Finally, the hazard and impact were combined with other random failures in the plant model to lead to core damage and/or offsite releases. If the core damage frequency was considerably less than other internal or external events, the event was screened out.

A number of "other" external events have also been considered in the DCPRA. Most of the events came from a compiled list found in the PRA Procedures Guide (Reference 5-4). However, many hazards from that list were judged to be of little significance or relevance to Diablo Canyon and, therefore, were not analyzed further. Table 5-1 gives this list of hazards and summarizes the reasons for including or excluding them from this analysis.

For the Diablo Canyon IPEEE, to evaluate these hazards, a review of other plant design documentation was performed. The plant documentation included the FSAR Update, NUREG-0675, Supplement No. 34 - Safety Evaluation Report for DCPP (Reference 5-6), NUREG-1407 (Reference 5-7), the SRP (NUREG-0800, Reference 5-8), NRC SRP Compliance Checklist for DCPP, and Diablo Canyon Design Criteria Memoranda (DCMs). Diablo Canyon Design Change Notices (DCNs) between 4/1/88 and 4/1/93 were also reviewed to identify any changes (since the completion of the original DCPRA) that could adversely effect the capability of the plant to withstand these "other" external events.

A plant walkdown was conducted to identify any changes that might impact plant vulnerabilities to these external events; additionally, the purpose of the walkdown was to assure that all conceivable external plant hazards were systematically considered. The conclusion of the walkdown is that no significant changes which would degrade the ability of the plant to withstand these "other" hazards, have occurred since the operating license

was issued. One observation from the walkdown was that a new chemical ethanolamine (ETA), which is less hazardous than the chemicals that had been analyzed in the PRA, was found in use to replace the existing ammonia hydroxide. The detailed description is listed in Section 5.4.1.

Based on the above review and walkdown, it is concluded that no plant vulnerabilities exist and that the IPEEE screening criteria are met for these external events. The reviews of the plant design bases for these hazards are briefly described in Sections 5.1 to 5.3 for high winds, floods, and transportation/nearby facilities, respectively. Also, the findings from the walkdown are summarized in Reference 5-9 and Table 5-2.

5.1 HIGH WINDS

A review of the analysis in Reference 5-5 shows that the critical concrete structures at DCPP can withstand at least a 200-mph wind without major damage (such as collapse of a wall or overturning of a structure). The annual frequency of excessive tornado wind (>= 200 mph) on the structures was calculated to be less than 3.2E-7 per year (Reference 5-1). The annual frequency of excessive hurricane wind (>= 150 mph) on the structures was calculated to be 3.2E-12 per year (Reference 5-1). With such low initiator frequencies, it is judged that tornado wind-initiated scenarios and hurricane-initiated scenarios are insignificant contributors to the overall core damage frequency and, therefore, there are no plant vulnerabilities to high winds.

The site design basis for high winds and tornadoes was reviewed (as documented in the FSAR Update, Sections 3.3 and other pertinent licensing and design information). The appropriate SRP criteria for high winds and tornadoes were then compared to the plant's current design basis (DCPP SRP Compliance Checklist). The review concluded that DCPP conforms to the SRP criteria.

5.2 EXTERNAL FLOODING

The external flooding hazard was evaluated by reviewing the FSAR Update, Sections 2.4, 3.4 (Reference 5-5), and the appropriate SRP criteria for the external flooding (Reference 5-2). This includes flooding from a maximum probable hurricane, tsunami, high tide, storm waves, probable maximum precipitation (PMP), and a severely degraded breakwater. The review concluded that DCPP conforms to the SRP criteria; therefore, there are no vulnerabilities. Reference 5-5 also shows that heavy rains will not cause sufficient ponding on the plant site to flood safety-related buildings; nor will it cause the only stream near the site (Diablo Creek) to overfill. The roofs of the safety-related buildings are designed to handle a PMP of 4 inches per hour. If the rainfall intensity should exceed this drain capacity, overflow scuppers will still prevent ponding on the roof. Yard areas around safety-related buildings are also sloped to keep water away from the buildings.

Another possible flooding source considered in Reference 5-1 is the raw water reservoirs located on the hill behind the plant at Elevation 310 feet. There are two reservoirs, each holding about 2.25 million gallons. Each reservoir is roughly egg-shaped, with major and

minor dimensions of approximately 270 feet and 190 feet. It is unlikely that the reservoirs can fail in such a way to pose a threat to the plant. However, a worst case scenario was evaluated in Reference 5-1 and the study concluded that the depth of flooding is not expected to cause serious damage to the plant. In addition, the flood will only be temporary and not sustained.

The other issue for external flooding (i.e., Section 2.4 of Reference 5-7) is Generic Letter 89-22 (Reference 5-10), in which the NRC adopted the latest National Weather Service (NWS) Probable Maximum Precipitation (PMP) criteria for future plants. It was indicated in the letter from the National Weather Service that the PMP for California, which is presently defined by Hydrometeorological Report (HMR) # 36 (Reference 5-11), is still valid. Thus, the present analysis in DCPRA for external flooding, which is based on HMR # 36, is still valid.

The only safety-related equipment needing special protection from external flooding are the auxiliary saltwater (ASW) pumps located within the intake structure. There are two ASW pumps per unit. Each pump is housed in its own room. Each room is equipped with a normally closed watertight door. The pump rooms are equipped with snorkels to allow air in the room to remove heat from the ASW pump motors. These snorkels allow the pump rooms to be waterproof up to +48 feet above the mean lower low water level (MLLW).

Two cases were evaluated for possible flooding of the ASW pump rooms (Reference 5-1). One case considers when one or more pump room doors being left open or failing during a tsunami event. In this case, flooding of the pump rooms will occur if the water level reaches the main deck level of the intake structure, which is at +20 feet MLLW. The other case considers the pump room doors being closed but the combined tsunami-storm wave height exceeding +48 feet MLLW.

The total frequency of flooding all four ASW pumps was calculated to be 5.7E-5 per year (Reference 5-1). However, loss of all ASW pumps does not automatically lead to core damage since there is a possibility of aligning fire water to the charging pumps, thus preventing RCP seal failure. The flood-initiated core damage failure frequency was therefore calculated to be 7.2E-7 per year, which is small compared with other contributors, and less than the 10E-6 per year suggested for screening in NUREG-1407 (Reference 5-7).

No significant items were noted for external flooding during the plant walkdown. Based on the review and walkdown, there have been no significant changes that would adversely affect the external flooding design basis at DCPP since issuance of the operating licenses.

In conclusion, the DCPP design basis for external flooding satisfies the SRP criteria. Also an assessment of Generic Letter 89-22 shows that the revised NWS PMP criteria does not impact DCPP. No potential vulnerabilities were identified with regard to external flooding.

5.3 TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

These hazards were evaluated by reviewing the FSAR Update, DCPP SRP Compliance Checklists, DCNs, and by a plant walkdown to identify and confirm that the original hazard analyses relevant to these external events are still valid and the SRP criteria are met. The events analyzed included aircraft crash, hazardous chemical, external fires, and ship impact.

5.3.1 TRANSPORTATION ACCIDENTS

5.3.1.1 Ship Impact

The plant intake structure, which houses the safety-related ASW pumps, is located on the coastline. The potential hazard to the intake structure and the ASW system from maritime vessels was analyzed in Reference 5-1. Scenarios involving ship breakthrough of the breakwater in its normal state (not degraded by heavy wave action) were shown to be not possible due to the speed required to generate the kinetic energy needed to physically force a passage. Scenarios involving oil spills and other floating debris were also concluded to have no consequence. Analysis scenarios involving a degraded breakwater, therefore greatly increasing the possibility of a ship arriving in the intake cove, resulted in a core damage frequency of 2E-8 per year. Scenarios involving a ship blocking the flow of water into the intake cove result in a core damage frequency of 5E-9 per year. Both of these frequencies are negligible compared to other core damage scenarios. No plant specific vulnerabilities were identified.

5.3.1.2 Aircraft Crash

There is one airport at San Luis Obispo (SLO) and there are several airways in the vicinity of DCPP. The airport is located approximately 12.5 miles east-northeast of DCPP. The airport is a general aviation county airport with some scheduled commercial service. The airways include low-level federal airways used by small general aviation type aircraft, high-level jet routes used by large air carriers, and military training routes used by fighter-type aircraft. An analysis of the risk from aircraft crashes on the DCPP was done in Reference 5-1 and, as part of the update, a review of the change in air traffic was also performed. The percentage of the collapse and perforation frequencies that will lead to core damage was provided by PLG. The result of the analyses indicated that the total damage and impact frequencies from aircraft impact that will lead to core damage are low (7E-7 per year), and, even if it is assumed that any structural damage leads to core damage, the frequency of core damage from aircraft crashes is negligible compared to other core damage scenarios.

5.3.2 NEARBY FACILITY ACCIDENTS

Industry in the vicinity of the DCPP site is mainly light and of a local nature serving the needs of agriculture in the area. Food processing and transportation of oil are the area's major industries, although the numbers employed are not large. Less than 5 percent of the work force in SLO County is engaged in manufacturing. The largest industrial complex is Vandenberg Air Force Base, located about 35 miles south-southeast of the

site. The Port San Luis tanker loading pier and the Point San Luis lighthouse and Coast Guard Reservation are approximately 6-1/2 miles east-southeast of the site.

The closest US Army installation is the Hunter-Liggett Military Reservation approximately 40 miles north of the site. The California National Guard maintains Camp Roberts, located to the east of the Hunter-Liggett Reservation, and Camp San Luis Obispo, located about 8 miles northeast of DCPP.

US Highway 101 is the main arterial road serving the coastal region in this portion of California. It passes about 10 miles to the east of the site, separated from it by the San Luis Mountains. State Route 1 passes 10 miles to the north and carries moderate traffic between SLO and the coast. The nearest public access is by county roads in Clark Valley (5 miles north) and See Canyon (5 miles east). Access to the site is by a private road from Avila Beach.

No products are manufactured, stored, or transported within 5 miles of the DCPP site. Materials manufactured, stored, or transported beyond 5 miles are not likely to be a significant hazard to the plant.

No explosive or combustible materials are stored within 5 miles of the site and no natural gas or other pipelines pass within 5 miles.

On the DCPP site, there are no natural-draft cooling towers or other tall structures that could damage equipment or structures important-to-safety in the event of collapse of such tall structures.

5.3.3 EXTERNAL FIRES

A review of the plant layout shows that the hazard to the plant from external fires is not significant except for the hillside area to the east of the plant (Reference 5-1). There was one instance in 1982 (prior to commercial operation) where nearby brush fire caused a partial loss of offsite power. This event happened before commercial operation of units 1 and 2 and DCPP has implemented a wildland management program that uses a variety of programs and methods to limit the fuels loading characteristics of the vegetation in the area surrounding the plant to keep brush growth down. There has been no reoccurrence with the wildland management program in place. If an external fire of the 1982 type and magnitude were to reoccur at the site, the most likely impact would be limited to a partial or total loss of offsite power. This type of event is considered as a contributor to the loss of offsite power initiating event, which is quantified and analyzed separately in the IPE (Reference 5-12). The loss of offsite power initiating event frequency used in the DCPP IPE is 9.1E-2. A less conservative number of 7.7E-2 value was used in the NRC study NUREG/CR-4550 (Reference 5-13) based on the industry loss of offsite power initiating event frequency report NUREG/CR-5032 (Reference 5-14).

If the loss of offsite power initiating event frequency per year was calculated based on DCPP's plant-specific experience (1 loss of offsite power event in 13 years from 1982 to 1994), the resulting DCPP-specific initiating event frequency would be 7.7E-2 per year.

Again this is less conservative than the loss of offsite power initiating event value used in the DCPP IPE report. Based on the discussion above, it was determined that the loss of offsite power initiating event frequency used in the DCPP IPE is bounding and the hazard from external fires to the plant is included in other core damage scenarios.

5.4 OTHER HAZARDS

5.4.1 HAZARDOUS MATERIAL

The hazard from chemicals stored onsite is dominated by the potential effect of a spill on control room habitability. The initiating event would be a chemical spill or tank rupture. This could be caused, for example, by a handling accident, container failure, or some other accident. After the material is released, to contribute significantly to risk, it must be carried by some mechanism to the control room air intake. It was assumed in the analysis in Reference 5-1 that the plant is at power at the beginning of the accident. When the chemical reaches the control room air intake, there are mitigating factors that prevent the operators from being incapacitated. The core melt scenario frequencies depend on the likelihood of successful operator intervention and on the random equipment failure rate.

The only hazardous chemical on site that may pose a hazard to the control room operators is ammonium hydroxide. DCPP used to have five, one-ton, cylindrical chlorine tanks located at the intake structure. In the DCPRA-1988, these tank were the most significant hazard. Subsequent to the completion of the DCPRA-1988, the chlorine tanks were replaced by a 7,000 gallon sodium hypochlorite tank. The sodium hypochlorite does not pose a direct hazard to the control room operators. This chemical is, therefore, not included in this evaluation (Reference 5-15).

Scenarios involving the control room operators being incapacitated from the spill of the remaining hazardous chemical (ammonium hydroxide) resulted in a core damage frequency of 8E-7 per year at DCPP (Reference 5-3). It is a small contributor to the total core damage frequency compared to the other initiators. During the walkdown (Reference 5-9), the engineers noticed a new chemical, ethanolamine (ETA), which was in the testing stage, to replace the existing ammonia hydroxide. ETA has been tested in DCPP Unit 1 since August 17, 1993, to control the secondary system pH. The field test data show that ETA performance was superior to ammonia. In addition, ETA is analyzed to be less hazardous than ammonia hydroxide (Reference 5-16). An analysis was done to evaluate control room (CR) habitability after a postulated 6,000 gallon, 85 percent ETA spill (Reference 5-17); the result indicated that the storage of 6,000 gallons of 85 percent ETA does not create any hazard to control room operators under normal CR HVAC or emergency CR HVAC operating conditions.

A number of other external events have also been considered in the PRA as mentioned in Section 5.0. Many hazards from Table 5-1 were judged to be of little significance or relevance to Diablo Canyon and, therefore, were not analyzed further. The list of the external events included in this review is provided in Table 5-1.

The basic approach was to review the DCPRA, Appendix F, the FSAR Update, and recent industry/NRC experiences on external events. The intent of this effort was to assess whether these events required further analysis or if the effort conducted in NUREG/CR-5042, Supplement 2 (Reference 5-18) was applicable to DCPP. The results of this review confirmed the reasonableness of the NUREG/CR-5042 conclusions that screened out these events from Diablo Canyon's IPEEE.

In addition, as described previously, a plant walkdown was conducted to confirm that there were no new impacts from other hazards. The results of the walkdown are documented in Reference 5-9.

The conclusion of this evaluation and walkdown is that there are no potential vulnerabilities identified with regard to these other hazards. Table 5-1 summarizes the basis for this conclusion.

5.5 UPDATED PRA RESULTS FOR HIGH WINDS, FLOODS, AND OTHER EXTERNAL EVENTS

The DCPRA (completed in July 1988) included an evaluation of a number of external event hazards. DCPRA, Appendix F quantitatively evaluated these external hazards: For this analysis, some minor changes have been made since 1988 to reflect revisions to the FSAR update, DCNs, and new data related to these hazards. The revised frequencies for high winds, floods, and other events are listed below:

Other External Event	Upper Bound Frequency of Core Damage per Year
• Aircraft crash	7.0E-7
• Ship impact	1.9E-8
• External flooding	7.2E-7
• Hurricane and tornado wind and missile	3.2E-7
• Hazardous chemical	8.0E-7
• External fire	Negligible

The conclusion of this evaluation is that there are no potential vulnerabilities identified with regard to these other hazards.

5.6 REFERENCES

5-1. PLG, Inc., "Diablo Canyon Probabilistic Risk Assessment, Appendix F," prepared for Pacific Gas and Electric Company, PLG-0637, July 1988.

5-2. Pacific Gas and Electric Company Calculation File Number F.5, Revision 0, "Documentation for the Other External Events Report," dated April 27, 1994.

5-3. Compliance Checklists, NRC Standard Review Plan, Diablo Canyon Power Plant, Pacific Gas and Electric Company, Revision 1, February 10, 1981, prepared by Stafco, Inc. Under Contract 5-74-80.

5-4. American Nuclear Society and Institute of Electrical and Electronics Engineers, "PRA Procedures Guide; A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," sponsored by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute, NUREG/CR-2300, April 1983.

5-5. Units 1 and 2, Diablo Canyon Power Plant Final Safety Analysis Report Update, Revision 9, November 1993.

5-6. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report," NUREG-0675, Supplement No. 34, Docket Nos. 50-275 and 50-323, June 1991.

5-7. U.S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, Final Report," NUREG-1407, June 1991.

5-8. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, July 1981.

5-9. Pacific Gas and Electric Company Memo to File, "Other External Events Walkdown Documentation," dated April 13, 1994, Chron. No. 220766.

5-10. U.S. Nuclear Regulatory Commission, Generic Letter 89-22, "Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed by National Weather Service," October 19, 1989.

5-11. Letter from John L. Vogel, Chief, Hydrometeorological Branch of National Oceanic and Atmospheric Administration to Ada Chuang, Pacific Gas and Electric Company, dated October 12, 1993.

5-12. Diablo Canyon Power Plant, Units 1 and 2, Individual Plant Examination Report, dated April 1992.

5-13. NUREG/CR-4550, Volume 3, Revision 1, "Analysis of Core Damage Frequency: Surry, Unit 1 Internal Event," dated April 1990.

5-14. NUREG/CR-5032, "Modelling Time to Recovery an Initiating Event Frequency for Loss of Offsite Power Incidents at Nuclear Power Plants," dated January 1988, by R.L. Iman.

5-15. Pacific Gas and Electric Company, "Control Room Habitability - Onsite Toxic Chemical Analysis," Mechanical (HVAC) Calculation Number 86-13, Diablo Canyon Power Plant, Units 1 and 2 Project, Revision 6, June 8, 1993.

5-16. Pacific Gas and Electric Company, Design Change Package (DCP) N-049047, Revision 0, dated February 18, 1994.

5-17. Letter from Bechtel Project Engineer to Michael R. Tresler, Pacific Gas and Electric Company, dated June 2, 1993, Chron. No. 208129.

5-18. U.S. Nuclear Regulatory Commission, "Evaluation of External Hazards to Nuclear Power Plants in the United States," NUREG/CR-5042, Supplement 2, February 1989.

5-19. Pacific Gas and Electric Company, "Long Term Seismic Program Final Report," PG&E Letter No. DCL-88-192, July 31, 1988.

Table 5-1. Summary Evaluation of Other External Events			
Hazard Type	Source Exists	Included in this Analysis	Remarks
Aircraft Impacts	Yes	Yes	See Section 5.3.
Avalanche	No	No	See landslide hazard type.
Coastal Erosion	Yes	No	Very slow process; long lead time to put plant in cold shutdown.
Drought	No	No	Not applicable, ultimate heat sink is Pacific Ocean.
External Flooding	Yes	Yes	See Section 5.2.
Extreme Winds and Tornadoes	Yes	Yes	See Section 5.1.
Fog	Yes	No	No direct impact, however indirect impact of fog, such as impact on aircraft crash frequency, is addressed as part of specific hazards.
Forest Fires	Yes	Yes	See Section 5.3.
Frost	Yes	No	Impact is negligible. Very unlikely event.
Hail	Yes	No	Impact of hail on offsite power is included in the frequency of loss of offsite power analysis. Contribution to the overall risk is judged to be negligible.
High Tide or High Lake Level	Yes	Yes	See Section 5.2.
High River Stage	No	No	Not applicable, no source exists.
High Summer Temperature	Yes	No	The impact of high temperature environment on equipment performance is included in equipment failure data.
Hurricane	Yes	Yes	See Section 5.1.
Ice Cover	Yes	No	Very unlikely event. Impact is negligible.
Industrial or Military Facility Accident	Yes	Yes	See Section 5.3.
Landslide	Yes	No	Likelihood of occurrence is small due to low profile topography and plant site grading design. Contribution to the overall risk is judged to be negligible.
Lightning	Yes	No	Plant is equipped with lightning protection. Impact on offsite power included in loss of offsite power frequency evaluation. Contribution to the overall risk judged to be negligible.
Low Lake or River Water Level	No	No	Not applicable, no source exists. However, the possibility of a ship blocking the saltwater intake leading to low water level is discussed in Section 5.3.

Table 5-1. Summary Evaluation of Other External Events

Hazard Type	Source Exists	Included in this Analysis	Remarks
Low Winter Temperature	Yes	No	Impact on equipment has been included through component (independent and common cause) failure rates.
Meteorite	Yes	No	Likelihood of occurrence is very small.
On Site Truck Accident	Yes	No	Main access road is distanced from rest of the buildings at DCPP.
Pipeline Accident (gas, etc.)	Yes	Yes	See Section 2.2 of FSAR (Reference 5-5).
Intense Precipitation	Yes	Yes	See Section 5.2.
Release of Chemicals in Onsite Storage	Yes	Yes	See Section 5.3.
River Diversion	No	No	Not applicable, no source exists.
Sandstorm	Yes	No	Judged to be insignificant in occurrence, frequency, and risk.
Seiche	No	No	Not applicable, no source exists.
Snow	No	No	Not applicable, no source exists.
Soil Shrink-Swell Consolidation	Yes	No	Very slow process. Contribution to the overall risk is judged to be negligible.
Storm Surge	Yes	Yes	See Section 5.2.
Transportation Accidents	Yes	Yes	See Section 5.3.
Tsunami	Yes	Yes	See Section 5.2.
Toxic Gas	Yes	Yes	See Section 5.3.
Turbine-Generated Missile	Yes	Yes	Low risk, addressed in DCPP Long Term Seismic Program Final Report (Reference 5-19), not required for IPEEE.
Volcanic Activity	No	No	Not applicable, no active volcanic mountains nearby.
Waves	Yes	Yes	See Section 5.2.

TABLE 5-2. OTHER EXTERNAL EVENTS WALKDOWN MATRIX										
		AREA POTENTIALLY IMPACTED IF HAZARD FREQUENCY NOT ACCEPTABLY LOW								
HAZARD	SCREENED OUT *	TURBINE BUILDING	CONTROL ROOM HVAC INTAKE AREA	CONTAINMENT / AUXILIARY BLDG AREA **	TANKS AREA ***	RESERVOIR AREA	230 KV SWITCH-YARD	500 KV SWITCH-YARD	BREAKWATER AREA	INTAKE AREA
Aircraft Impact	3	X	X	X	X	X	X	X	X	X
Avalanche	1									
Coastal Erosion	1									
Drought	1									
External Flooding	2,3	X	X	X	X				X	X
Extreme Winds and Tornados	3	X	X	X	X	X	X	X	X	X
Fog	1									
Forest Fires	1,4									
Frost	1									
Hail	1									
High Tide, High Lake Level, or High River Stage	3	X	X	X	X				X	X
High Summer Temperature	1									
Hurricane	3	X	X	X	X	X	X	X	X	X
Ice	1,2									
Industrial or Military Facility Accident	2,3								X	X
Landslide	1,2									
Lightning	1									

TABLE 5-2. OTHER EXTERNAL EVENTS WALKDOWN MATRIX

		AREA POTENTIALLY IMPACTED IF HAZARD FREQUENCY NOT ACCEPTABLY LOW								
Low Lake or River Water Level	1									
Low Winter Temperature	1									
Meteorite	1									
Pipeline Accident (gas, etc.)	1									
Intense Precipitation	2	X	X	X	X					
River Diversion	1									
Sandstorm	1									
Seiche	1,2									
Snow	1									
Soil Shrink-Well Consolidation	1									
Storage (Release of Chemicals in Onsite Storage)	2,3	X	X	X	X					
Storm Surge	2,3								X	X
Transportation Accidents	3								X	X
Tsunami	2,3								X	X
Toxic Gas	2,3	X	X	X	X					
Turbine-Generated Missile	3	X	X	X	X					
Volcanic Activity	1									

TABLE 5-2. OTHER EXTERNAL EVENTS WALKDOWN MATRIX										
		AREA POTENTIALLY IMPACTED IF HAZARD FREQUENCY NOT ACCEPTABLY LOW								
Waves	3								X	X

X - denotes an area potentially impacted by the hazard.
* (1) - Screened out using the process depicted in Figure 5.1 of NUREG-1407 (hazard frequency acceptably low).
* (2) - Screened out using the process depicted in Figure 5.1 of NUREG-1407 (plant/facilities design meet 1975 SRP criteria).
* (3) - Screened out using the process depicted in Figure 5.1 of NUREG-1407 (bounding analysis was done).
* (4) - In 1982, a brush fire impacted the 230 and 500 KV switchyard areas. PG&E has implemented brush clearing procedures to minimize chance of reoccurrence.

** This area includes the Containment Building, Auxiliary Building, Fuel Handling Area, and Ventilation Area.
*** This area includes the Primary Water Storage Tank, Condensate Storage Tank, Refueling Water Storage Tank, Transfer Storage Tank, and Fire Water Tank.

6. LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 IPEEE PROGRAM ORGANIZATION

The IPEEE program has been managed by the PG&E PRA Group, which is part of the Nuclear Regulatory Services Department. Contributions to the IPEEE were made by numerous PG&E Departments.

As discussed in Section 2, the DCPRA-1988 was completed in 1988 by a team of PG&E personnel and consultants led by PLG. The PRA Group at PG&E now maintains complete control and responsibility of the DCPRA. As part of the PRA and IPEEE processes, the PRA Group is supported by members of other organizations, as discussed below.

- **Nuclear Engineering Services** - Fire protection personnel provided Appendix R information, cable routing information, and Sandia Fire Risk Scoping Study responses. The department assisted in the review of design changes issued since the DCPRA-1988 for their impact on seismic risk. They provided updated fragility information, where needed. The department also provided information on the "Other External Events" hazard evaluation. Finally, the department participated in the IPEEE walkdowns.

- **Operations** - Operations personnel participated in the human actions operator surveys and also provided information regarding DCPP operation.

- **Emergency Services** - Emergency Services personnel assisted in the Fire PRA, by providing information on Sandia Fire Risk Scoping Study issues, as well as DCPP-specific fire brigade information.

- **Nuclear Regulatory Services** - PG&E licensing personnel in the department provided guidance on the IPEEE and interface with the NRC on IPEEE issues.

- **Training** - The Training Department personnel conducts operator and technical staff training classes on the subject of PRA insights.

- **Systems Engineering** - System Engineers provided information on the operation of their systems.

- **Reliability Engineering** - Reliability Engineers assisted in the collection of the plant-specific failure rate and maintenance data.

- **Independent Safety Engineering Group** - Provided DCPP and industry experience of various events related to the PRA and the IPEEE and participated in the IPEEE walkdowns.

- **Geosciences Department** - Confirmed the validity of the hazard curves developed as part of the DCPP Long Term Seismic Program (LTSP).

Consultant assistance was provided by Robert Kennedy (seismic), Tom Kip (seismic fragility), and PLG (external events PRA).

The individuals who participated in the IPEEE analyses are presented in Table 6-1.

6.2 COMPOSITION OF INDEPENDENT REVIEW TEAM

As discussed earlier, the DCPRA-1988 developed for the Long Term Seismic Program is the basis for the DCPP external events PRA performed for the IPEEE. As such, the internal review of the DCPRA-1988 is an important aspect of the independent review process. The DCPRA-1988 analyses were reviewed by PG&E personnel to ensure the accuracy of the PRA documentation from an independent perspective, and to validate the PRA process and its results. The personnel that participated in the review represented various organizations within PG&E, as summarized in Table 6-2.

Another important aspect of the independent review of the external events portion of the DCPRA is the NRC review of the DCPRA-1988 that began in 1985 and was completed in 1991 with the issuance of SSER No. 34 for NUREG-0675 (Reference 6-1). The review of the DCPRA-1988 was conducted by the NRC and its consultants. The review in the SSER concluded that the PRA analyses were acceptable. The comments made by the NRC and its consultants on aspects of the PRA relating to external events are discussed in Sections 6.3 and 6.4.

During the PRA update process performed for the IPEEE, PRA analyses were prepared and independently verified in accordance with PG&E procedures. The independent verification of PRA analyses was performed by individuals in the PRA Group, or from PLG. The IPEEE report was reviewed by PG&E DCPP organizations, as summarized in Table 6-3.

The IPEEE report was reviewed by consultants experienced in external events PRA. PLG, Robert Kennedy, and Tom Kip reviewed the report. Donald J. Wakefield of PLG performed the major review for PLG.

6.3 AREAS OF REVIEW AND MAJOR COMMENTS

The DCPRA-1988 review by PG&E included review of the external events portion of the PRA. Areas of review are included in Table 6-2, and include reviews of the spatial interactions analysis, external events analysis, event trees, fault trees, and data analysis.

The comments from the DCPRA-1988 review by PG&E were primarily directed toward the accuracy of the PRA models and data, as they reflected design and operating plant

conditions. These comments were resolved between PG&E and PLG before the DCPRA-1988 documentation was submitted to the NRC in the LTSP Final Report and in subsequent submittals.

The DCPRA-1988, including the external events portions, was subjected to a rigorous and comprehensive independent evaluation by the NRC Staff. This evaluation provided additional confirmation of the adequacy of the DCPRA-1988. During the evaluation process, several comments from the NRC Staff and Bookhaven National Laboratory (BNL) were resolved by PG&E. The comments from BNL and the NRC Staff related to the following areas: seismic analysis (sequences, assumptions, human factors, relay chatter, quantification, uncertainties, fragilities), support and front-line system analysis, human actions analysis, data analysis, success criteria, truncation and cutoff frequencies, common cause treatment, external events considered, systems analysis assumptions, and fire analysis. PG&E responded to the questions in submittals to the NRC. Where appropriate, the PRA Group revised the PRA models during the IPE and IPEEE processes to include model modifications, improvements, or sensitivity studies resulting from these comments. PG&E's resolution of the comments applicable to external events is summarized in Section 6.4 and presented in Table 6-4.

The update to the external events portion of the DCPRA-1988 (DCPRA-1993) was subject to PG&E independent review, in accordance with PG&E procedures. Modeling updates and analyses pertaining to the external events analysis were documented and subjected to PG&E reviews.

PG&E personnel reviewed the IPEEE report, as indicated in Table 6-3. The objective of the IPEEE peer review is "the same type of review as requested for the internal event IPE," (Reference 6-2), i.e., "to ensure the accuracy of the documentation packages and to validate both the IPE (IPEEE) process and its results" (Reference 6-3). Most of the review comments were either of an editorial nature or requests for clarification. The major comments provided are presented in Table 6-5. The review did not reveal any PRA modeling inaccuracies or inconsistencies.

The PLG review of the external events portion of the IPEEE found the results to be consistent with those reported to the NRC in 1988, after accounting for design changes and modeling enhancements. PLG's major comments are also included in Table 6-5.

6.4 RESOLUTION OF COMMENTS

Comments generated within PG&E as part of its internal review of the DCPRA-1988 were resolved before the DCPRA-1988 results were submitted to the NRC in the LTSP Final Report. These comments were primarily directed toward the accuracy of the PRA model and data as they reflected plant design and operating conditions. Resolution of the comments was reflected in the documentation submitted to the NRC.

During the NRC review of the LTSP, several comments were made by the NRC Staff and BNL. PG&E responded to these comments during the LTSP review. Additionally, corrections to the DCPRA-1988 and the LTSP report were documented in PG&E's addendum to the LTSP Final Report, which was submitted to the NRC on February 13, 1991 (Reference 6-4). Some of these corrections led to model changes in the IPE and IPEEE models. These are documented in Table 6-4.

The resolution of the major IPEEE comments generated by the PG&E and PLG reviewers are presented in Table 6-5. Most of the comments concerned clarification of items or identification of conservatisms in the report. Editorial comments or areas of clarification were addressed in the version of the report submitted to the NRC. No significant changes were made to the PRA models as a result of the reviews. Areas of conservatism are most helpful in identifying areas where the external events PRA can be enhanced in future updates.

6.5 REFERENCES

6-1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report," NUREG-0675, Supplement No. 34, Docket Nos. 50-275 and 50-323, June 1991.

6-2. U.S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

6-3. U.S. Nuclear Regulatory Commission, "Individual Plant Examination for Severe Accident Vulnerabilities," Generic Letter 88-20, November 23, 1988.

6-4. Pacific Gas and Electric Company, "Addendum to Long Term Seismic Program Final Report," PG&E Letter No. DCL-91-027, February 13, 1991.

Table 6-1. IPEEE Analysis Personnel

PG&E

PRA Analysts

Ada E. Chuang
Randall M. Johnson
Thomas L. Leserman
Douglas E. Naaf
John C. Oliver
Richard C. Ryan
Mark W. Zimmermann

Seismic Assistance

David Ovadia
Mohsin Khan
Eric M. Fujisaki
Lloyd Cluff
Henry Thailer
Operations Personnel

Fire Assistance

Andy Ratchford
Koji Maemura
Russ Leatham
Dave Cosgrove
Carmon Johnson
Eric Mellinger

Other External Events Assistance

Ed Dubost
Carmon Johnson
Richard Potter
Mike Peterson
Francis Ling

Consultants

Don Wakefield - PLG
Wee Tee Loh - PLG
Hal Perla - PLG
Robert Kennedy
Tom Kip - EQE Engineering Consultants

Table 6-2a.PG&E Review of the DCPRA-1988

DCPRA-1988 Review Area	Engineering										
	Auer	Connell	Pellisero	Smith	Beckham	Thierry	Clark	Hermann	Anderson	McCall	Tidrick
Initiating Events	X	X	X		X						
Support System Dependency Tables	X	X									
Support System Analysis	X	X		X	X						
Frontline System Analysis		X	X		X	X					
Event Sequence Diagrams		X			X						
Support Event Trees	X	X		X							
Frontline Event Trees	X	X	X		X						
Special Event Trees	X	X	X		X						
Data Analysis						X					
Human Action Analysis					X						
Spatial Interactions Analysis	X	X			X		X	X			
Plant Damage States		X			X						X
Design and Construction Errors Analysis	X	X			X				X	X	
External Events		X			X						X

Table 6-2b.PG&E Review of the DCPRA-1988

DCPRA-1988 Review Area	Operations			Training			Regulatory Compliance	
	Fridley	Luckett	Fisher	Molden	Martin	Beardon	Grebel	Hinds
Initiating Events	X	X	X	X	X			
Support System Dependency Tables					X			
Support System Analysis					X			X
Frontline System Analysis					X			X
Event Sequence Diagrams	X	X			X	X	X	
Support Event Trees					X			
Frontline Event Trees	X	X		X	X			
Special Event Trees	X	X		X	X			X
Data Analysis								
Human Action Analysis		X		X	X			
Spatial Interactions Analysis								X

Table 6-3. PG&E Departments that have Reviewed the IPEEE Report
Nuclear Engineering Services
Maintenance Services
Reliability Engineering
Operations
Nuclear Quality Services
Training
Nuclear Regulatory Services
Emergency Services
Geosciences
Technical Services

Table 6-4. NRC/BNL Questions and Comments from the DCPRA-1988 Review that Pertain to External Events	
NRC/BNL Question or Comment	**Response/Resolution**
A question was raised pertaining to how the CO_2 system in the DG rooms respond to seismic events. The comment was made that flooding the rooms with CO_2 may inhibit recovery operations. (Ref: BNL Letter Report-07, July 1989)	One recovery action (manual control of the diesel generator level control valves) does require the operator to enter the diesel generator room. With no credit for operator action, the core damage frequency was found to increase only 1%.
A question was raised pertaining to the fraction used (one third) for the area of control room vertical board 1 on which a fire could affect both ASW and CCW control circuits. (Ref: NRC Staff Questions on PRA Fire Analysis, October 1989)	The use of one third of the board area with a severity factor of 1.0 was shown to be conservative.
A question was raised if 500kV switchyard breakers have been replaced with seismic resistant breakers. (Ref: BNL Letter Report-08, December 1989)	No credit is given for the 500kV source of offsite power in either the internal events PRA or the external events PRA, so the question is irrelevant.
Requests were made to provide the seismic block diagram containing the Boolean equations used in the seismic quantification and also provide the document describing the seismic sequences. (Ref: NRC Letter, Enclosure 3, August 1989)	Documentation was provided.
Requests were made for additional documentation and justification for the seismic analysis, including information on the following: RHR heat exchanger frequency calculation, nozzle loads on RHR heat exchanger, basis for Beta values and factors in fragility calculations, basis for seal LOCA assumptions, relay fragility calculations, turbine building nonlinear analysis, seal water heat exchanger fragility, RWST low-level relay lock-in capability, 24 hour offsite power recovery assumption, seismic human actions credited, and hazard/fragility interface. (Ref: NRC Letter, Enclosure 4, August 1989)	Information was provided.
A request for information was made on the fire analysis pertaining to the basis for 10 minutes before onset of RCP seal damage after loss of CCW. (Ref: NRC Letter, March 1990)	Justification was provided.
A request for information was made on the control room fires on the following: independence between the control room and the hot shutdown panel, LOCA mitigation capability from the hot shutdown panel, and procedures for tripping the RCPs from outside the control room when component cooling water is lost. (Ref: NRC Letter, March 1990)	Information was provided and the DCPRA Appendix F was revised.
A request for information was made on the methods for uncertainty calculation for the non-seismic dominant sequences, the fire scenarios, and the seismic model. (Ref: NRC Meeting April 3 and 4, 1990)	Information was provided.

Table 6-4. NRC/BNL Questions and Comments from the DCPRA-1988 Review that Pertain to External Events	
NRC/BNL Question or Comment	**Response/Resolution**
It was requested that PG&E provide the contribution to core damage frequency of fire initiating events FS2, FS3, FS4, and FS7. (Ref: NRC Meeting April 3 and 4, 1990)	Information was provided.
It was requested that PG&E clarify why fire scenario 13-A-FS1 was not used in the dominant sequence model. (Ref: NRC Meeting April 3 and 4, 1990).	Information was provided demonstrating that a fire in that fire area does not have a significant impact.
There was a request for additional information on the treatment of turbine building fires, particularly the impact of turbine building fires on 4kV switchgear rooms. (Ref: NRC Verbal Request, June 1990)	Information was provided on severe turbine building fires. For the IPEEE, the initiating event frequency of turbine deck fires was updated, based on industry experience.

Table 6-5. Major PG&E and PLG IPE Report Review Comments	
Comment	**Resolution**
Since the Safety Injection System (SI), is not an Appendix R system, it was questioned how the cable routing was determined for use in the fire analysis?	The SI cable routing was determined using a combination of engineering drawings, walkdown information, and engineering judgement. To determine the impact of sensitivity of PRA results to SI cable routing information, the risk achievement worth of the SI system for fire sequences was reviewed; it was shown that the fire PRA core damage frequency increases by less than one percent, even if no credit is given for the SI system.
Since offsite power is not an Appendix R system, is it properly considered in the fire analysis?	Engineering provided the actual cable routing of all of the offsite power distribution systems. The offsite power routing information validates the existing screening used in the fire analysis.
Since not all of the RHR system is credited in Appendix R, it was questioned how the cable routing of the RHR system impacts the PRA.	Most of the RHR components credited in the PRA are the same as Appendix R. In limited cases, the PRA credits cold leg recirculation capability, which is not credited in Appendix R. Removing credit in the PRA for cold leg recirculation does not significantly impact the results. Thus the PRA is relatively insensitive to the RHR cable routing in question.
Seismic sequences ranked 10, 39, and others of the seismic analysis lead to core damage because of auxiliary saltwater system failure. Is the model taking credit for the backup firewater cooling to the charging pumps which would prevent core damage?	These sequences contribute only a few percent to core damage and, if the comment is correct, may conservatively overpredict core damage. These sequences will be evaluated in future updates to determine if the model should be revised and this possible source of conservatism removed.
The seismic analysis discussion should be improved to make it clear that both seismic caused failures and non-seismic random failures are included in the sequences leading to core damage.	The discussion in Section 3.1.3 has been revised to clarify this issue as well as the titles of Figures 3-11, 3-12, 3-14 and 3-15.
The seismic model should be reviewed to ensure it considers the small leaks that may result from a seismic event as discussed in Reference 3-23.	The model was revised to assume small leaks from the primary system after seismic initiating events and require operation of the charging pumps. This aspect of the model is discussed in Section 3.1.3.7.

7. PLANT IMPROVEMENTS AND UNIQUE PLANT SAFETY FEATURES

7.1 PLANT IMPROVEMENTS

As part of the Long Term Seismic Program (LTSP), during the development of the DCPRA-1988, several plant modifications were implemented to reduce potential vulnerabilities at Diablo Canyon Power Plant (DCPP) from both internal and external events (Reference 7-1). These modifications are discussed below.

- **Diesel Generator Fuel Oil Transfer System.** The diesel fuel oil transfer system replenishes the day tanks of the emergency diesel generators with fuel oil from the main underground storage tanks. There are two fuel oil transfer trains that supply the diesel generators with fuel oil from the main underground storage tanks. The system is normally in standby unless required to replenish the day tank of an operating diesel generator.

  The original system operated on a demand basis; when the level in a diesel generator day tank reached a low level, the transfer pump would start, refill the day tank to the full level, and then the pump would stop. To increase system reliability, recirculation lines were added to allow the system to operate continuously once a start demand was received. Multiple start demands were eliminated by the addition of recirculation lines.

  In addition, provisions were made to allow for manual operation of the level control valves on the diesel generator day tanks and to allow a portable engine-driven pump to be connected to the system. These features allow recovery from support system or hardware failures. Procedures have been developed for performing these recovery actions.

- **Charging Pump Backup Cooling.** The centrifugal charging pump's lube oil and seal coolers receive cooling from the component cooling water (CCW) system. For scenarios involving a complete loss of CCW, provisions have been made and procedures are in place to allow the use of fire water to cool the centrifugal charging pumps (only one pump can be connected at a time); this is accomplished through the use of dedicated hoses to connect between the fire water header and the charging pump coolers. This design feature allows reactor coolant pump (RCP) seal injection and, consequently, RCP seal cooling to be maintained for scenarios involving a complete loss of CCW.

- **Substation Spare Parts.** For seismic events that result in a loss of offsite power due to switchyard equipment failures, spare parts are stored onsite to allow expeditious recovery. The spare parts include items such as conductors, connectors, insulators, and transformer bushings. Onsite storage ensures that the parts will be available in a timely manner for use by recovery personnel.

One potential plant improvement has been identified as part of the IPEEE process and is described below:

- **Control Room Evacuation Procedure.** As part of the IPEEE, one procedure modification is being evaluated. The control room evacuation procedure (Reference 7-2) modification would require the reactor coolant pumps to be tripped in the event the control room fire is located in cabinets that could result in loss of CCW or auxiliary saltwater (ASW) systems.

There are also other recently completed or ongoing plant improvements, most of which were not PRA-related in origin, but will have a beneficial impact on the PRA analysis, nonetheless. These include the changes listed below.

- **Dedicated Sixth Emergency Diesel Generator.** The most important of these plant modifications is the addition of the sixth emergency diesel generator that was completed in 1993. Prior to installation of the sixth diesel generator, diesel generator 1-3 acted as a swing diesel between the vital AC F buses of Unit 1 and Unit 2. As such, diesel generator 1-3 could only support the vital AC F bus of one unit if needed during a plant transient. The addition of the sixth diesel allows each vital AC bus to be supported and will increase the availability of backup power for vital AC bus F. Installation of the sixth diesel is calculated to have reduced the contribution of loss of offsite power events to the overall core damage frequency. It also is calculated to have reduced the likelihood of ASW or CCW system failures leading to a loss of RCP seal cooling.

- **480 V Switchgear Ventilation.** A design change eliminated the possibility that a single failure of the motor-operated discharge damper could have failed the 480 V switchgear ventilation system.

- **Component Cooling Water System Procedures.** Due to Generic Letter 91-13, as well as the results of the DCPRA-1991, Operating Procedure OP AP-11, "Malfunction of Component Cooling Water System," was revised to better ensure the RCP seal cooling is maintained to prevent RCP seal LOCAs.

- **Eagle 21 Process Protection System and RTD Bypass.** These modifications were installed in Unit 1 and are scheduled for installation in Unit 2 during the fall 1994 refueling outage. The Eagle 21 upgrade improves the reliability and availability of the plant process protection system. The resistance temperature detector (RTD) bypass elimination will reduce plant downtime and radiation exposures to plant personnel.

- **Instrument Inverter Replacement.** The Eagle 21 Process Protection System necessitates replacing the instrument inverters with inverters of increased capacity. The design provides increased reliability by including automatic backup switching (static switch) in the event the instrument inverter fails. These modifications have been installed in Unit 1 and are scheduled for installation in Unit 2 in the fall of 1994.

7.2 UNIQUE SAFETY FEATURES

This section discusses other safety features at DCPP that are significant in reducing in the risk to the plant. Many of these features may also be found at similar vintage pressurized water reactors.

- **Auxiliary Feedwater System.** The auxiliary feedwater system supplies water to the steam generators to provide backup secondary-side cooling. The auxiliary feedwater system contains one full-capacity, turbine-driven pump and two half-capacity, motor-driven pumps (MDP). The system can succeed in removing the decay heat from the core if sufficient flow from any one pump is delivered to any one steam generator. Each pump is supported by a different electrical power train. Flow paths from two steam generators provide steam to the turbine-driven pump. The pumps are located in two separate rooms with one room containing both motor-driven pumps.

- **Auxiliary Saltwater System.** Each of the two ASW system pumps is capable of supplying the minimum flow requirements after a plant trip. The ASW system was designed to withstand flooding of the intake structure; each pump is located in its own watertight vault. The unit-to-unit crosstie capability allows each unit to supply the opposite unit's ASW needs. A train-to-train crosstie allows each ASW train to supply to supply either CCW heat exchanger. The only load on the ASW system is the CCW heat exchanger.

- **High Pressure ECCS.** The high pressure emergency core cooling system contains four injection pumps, two centrifugal charging pumps, and two safety injection pumps. During transients and small LOCAs, any one of the four pumps is capable of supplying the minimum high pressure flow requirements. The charging pumps and the safety injection pumps are located in two separate rooms. As mentioned in the previous section, backup cooling (using the firewater system) is available for the charging pumps in the event CCW is lost.

- **Unit-to-Unit Electrical Power Train Crosstie.** As stated earlier, each unit can provide the opposite unit with ASW. DCPP was also designed and procedures are available to allow for unit-to-unit crosstying of vital electrical power in the event of a failure of an emergency diesel generator.

- **Diesel Generators.** To reduce the probability of diesel generator failure, several measures were taken to isolate the diesel generators from each other and other equipment in the plant. Most importantly, the diesels are located in separate rooms containing fire walls and fire doors. Furthermore, each diesel generator has its own jacket cooling water system, which maintains the operating temperature of the diesel engine by removing engine heat through a radiator. An automatic CO_2 fire protection system is installed for the protection of the equipment in each diesel generator room.

- **Vital Buses.** Three vital buses (F, G, and H) power the safety-related systems at DCPP. The safety-related systems are two-train systems. The effect of losing one or two vital buses is not as significant as some plants with only two vital buses.

- **Seismic Trip sensors.** The plant has seismic trip sensors located in various plant locations that are set to trip the plant for seismic events of a sufficient magnitude.

7.3 USE OF THE DCPRA AT DCPP

The DCPRA models have been and continue to be used to address external event PRA issues in support of DCPP operations as described below.

- **Seismic Trip Monitor AOT and STI** To support a change in the surveillance test interval of the seismic trip monitor, a risk analysis was performed. The analysis results were used in License Amendment Request (LAR) 88-07 submitted to the NRC in PG&E Letter DCL-88-267 on November 10, 1988. The amendment requested an exemption from the required surveillance test of the seismic trip monitor; the exemption would last until the 1989 Unit 1 refueling outage. The exemption was granted.

  Additionally, a permanent change to the surveillance test interval for the seismic trip monitor was requested in LAR 89-03 submitted to the NRC in PG&E Letter DCL-89-071 on March 22, 1989. The LAR requested a change in the surveillance test interval from six to eighteen months and allow the seismic trip system to be removed from service for 72 hours for maintenance or component repair while at power. The amendment was granted with the provision that only one of the three channels can be bypassed for up to 72 hours while at power.

- **Anchor Darling Check Valve JCO.** An analysis was done to evaluate the increase in risk due to stud failures in Anchor Darling 8-inch and 10-inch check valves in the emergency core cooling system. The affected valves were 8740A,B (RCS hot legs), 8948A-D (RCS cold legs), and 8956A-D (accumulators). The results of the analysis were referenced in Justification for Continued Operation (JCO) 88-09 for Unit 1.

- **Block Wall Prioritization.** Insights from the DCPRA-1988 were used to prioritize which safety-related block walls should be modified first; the walls were prioritized in three categories: high, medium, and low priority. The results and method of prioritization were presented to the NRC during a meeting on January 31, 1991. Ultimately, PG&E committed to complete all modifications to high priority walls within one cycle for each unit and all medium and low priority walls within two cycles for each unit.

- **230 kV Switch Yard Fragility Revision.** After the Loma Prieta earthquake, the NRC requested that PG&E, as part of the LTSP, reevaluate the fragility of the 230

kV switchyard based on the Loma Prieta earthquake experience. The reevaluation resulted in a change in the fragility of the 230 kV switchyard. The effect of this change on the seismic risk was evaluated; and the results were submitted to the NRC in the report titled "Reassessment of the HCLPF Level of the Diablo Canyon 230 kV Switchyard Following the 1989 Loma Prieta Earthquake." The revised fragility is used in the IPEEE.

Aside from hardware improvements and PRA analyses, another method to increase safety at DCPP is to make plant personnel more aware of the insights gained from the DCPRA-1988 and the potential applications of PRA. To this end, numerous PRA training classes have been presented to PG&E personnel, primarily focusing on PRA insights. PRA training classes have included a two-day PRA course that was conducted for DCPP engineers and operators at the plant by PLG personnel and members of the PRA Group. As a follow-on to the course, Training Department conducted a PRA class that was presented to operators at DCPP. The Training Department also has incorporated some PRA insights in their system review classes for licensed operator training. Finally, a PRA training class has been provided for the PG&E design engineers.

To assist in future PRA-related applications, PG&E will maintain the DCPRA as a "living PRA" through routine PRA updates to model the current operating condition of DCPP.

7.4 REFERENCES

7-1. Pacific Gas and Electric Company, "Addendum to the Long Term Seismic Program Final Report," PG&E Letter No. DCL-91-027, February 1991.

7-2. Pacific Gas and Electric Company, Control Room Abnormal Operating Procedure, "Control Room Inaccessibility - Establishing Hot Standby," OP AP-8A, Revision 5, August 19, 1993.

8. SUMMARY AND CONCLUSIONS

This report summarizes the plant-specific probabilistic risk assessment conducted to determine vulnerabilities to severe accidents at Diablo Canyon Power Plant (DCPP) that are postulated to occur as a result of external initiating events. The analyses and results presented in this report represent the update and enhancement of the original Diablo Canyon Probabilistic Risk Assessment (DCPRA-1988) that was performed as part of the Long Term Seismic Program. The external events update of the PRA is referred to as DCPRA-1993 and includes the analysis of probable reactor damage and resulting plant damage states as a result of external initiating events. While the study was performed for Unit 1, the results are equally applicable to Unit 2 because of the substantial similarities between the two units.

The results of the study indicate that the core damage frequency due to seismic events is 4.0 X 10^{-5} per year. The core damage frequency due to fire events is 2.7 X 10^{-5} per year. "Other" external events were evaluated and it was determined that each of these external initiating events contributes less than 1 X 10^{-6} per year to the core damage. These were screened out as a result. Core damage vulnerabilities were determined to be any component, system, operator action, or accident sequence that contributes more than 50 percent to the core damage frequency or has a frequency that exceeds 1 X 10^{-4} per year. Based on this criterion, no vulnerabilities with respect to core damage were identified in this study.

Containment performance vulnerabilities were considered to be any containment bypass, or large early release sequence with a frequency exceeding 1 X 10^{-5} per year. No containment vulnerabilities were identified in this study.

As part of the IPEEE process, PG&E reviewed Unresolved Safety Issue USI A-45 and Generic Issue GI-131 for external event implications. PG&E has concluded that these issues can be considered resolved for the Diablo Canyon Power Plant.