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Vol. 5

Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1

Analysis of Core Damage Frequency from Seismic Events for Plant Operational State 5 During a Refueling Outage

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

Traditionally, probabilistic risk assessments (PRAs) of severe accidents at nuclear power plants have explored accidents initiated during full-power operation. However, in 1989 the U.S. Nuclear Regulatory Commission (NRC) initiated an extensive program to examine carefully the potential risks during low-power and shutdown operations. The program included two parallel projects, one at Sandia National Laboratories studying a boiling water reactor (Grand Gulf), and the other at Brookhaven National Laboratory studying a pressurized water reactor (Surry Unit 1). Both the Sandia and Brookhaven projects have examined only accidents initiated by internal plant faults --- so-called "internal initiators". This project, which has explored the likelihood of seismic-initiated core damage accidents during refueling outage conditions, is complementary to the internal-initiator analyses at Brookhaven and Sandia. This report covers the seismic analysis at Grand Gulf, while a companion report documents the Surry seismic analysis.

All of the many systems modeling assumptions, component non-seismic failure rates, and human error rates that were used in the internal-initiator study at Grand Gulf have been adopted here, so that the results of the study can be as comparable as possible. Both the Sandia study and this study examine only one shutdown plant operating state (POS) at Grand Gulf, namely POS 5 representing cold shutdown during a refueling outage. This analysis has been limited to work analogous to a level-1 seismic PRA, in which estimates have been developed for the core-damage frequency from seismic events during POS 5. The methodology is almost identical to that used for full-power seismic PRAs, as widely practiced in the nuclear industry.

The results of the analysis are that the core-damage frequency for earthquake-initiated accidents during refueling outages in POS 5 is found to be quite low in absolute terms, less than 10^{-7}/year. The core-damage frequency is also low relative to the frequency during POS 5 for internal initiators, as analyzed in the companion study by Sandia.

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Traditionally, probabilistic risk assessments (PRAs) of severe accidents at nuclear power plants have explored accidents initiated during full-power operation. However, in 1989 the U.S. Nuclear Regulatory Commission (NRC) initiated an extensive program to examine carefully the potential risks during low-power and shutdown operations. The program included two parallel projects, one at Sandia National Laboratories studying a boiling water reactor (Grand Gulf), and the other at Brookhaven National Laboratory studying a pressurized water reactor (Surry Unit 1). Both the Sandia and Brookhaven projects have examined only accidents initiated by postulated internal plant faults --- so-called "internal initiators".

This project, which has explored the likelihood of seismic-initiated core damage accidents during refueling-outage conditions, is complementary to the internal-initiator analyses at Sandia and Brookhaven. This report covers the seismic analysis at Grand Gulf, while a companion report documents the Surry seismic analysis. The project reported here is the second phase of a two-phase effort, the initial phase of which was scoping in character, having been performed primarily to establish if the issue was judged to be important enough to justify the more extensive and more quantitative evaluation reported here.

All of the many systems modeling assumptions, component non-seismic failure rates, and human error rates that were used in the internal-initiator study at Grand Gulf have been adopted here, so that the results of the two studies can be as comparable as possible. Both the Sandia study and this study examine only one shutdown plant operating state (POS) at Grand Gulf, namely POS 5 representing cold shutdown during a refueling outage.

This analysis has been limited to work analogous to a level-1 seismic PRA, in which estimates have been developed for the core-damage frequency from seismic events during POS 5. The methodology is almost identical to that used for full-power seismic PRAs, as widely practiced in the nuclear industry. However, seismic-induced relay chatter is beyond the scope of this analysis. Seismic hazard curves from both the Electric Power Research Institute (EPRI) and the Lawrence Livermore National Laboratory (LLNL) have been used. Assessments of the likelihood of various post-core-damage plant-damage states (level-2 PRA) and of significant radioactive releases (level-3 PRA) are beyond the scope of this evaluation.

The results of the analysis are that the core-damage frequency for earthquake-initiated accidents during refueling outages in POS 5 is found to be quite low in absolute terms, less than 10^{-7}/year. This is true using both the EPRI and the LLNL seismic hazard curves. The reasons for this are (i) Grand Gulf's seismic capacity in responding to earthquakes during shutdown is excellent, well above its design basis; (ii) the Grand Gulf site enjoys one of the least seismically active locations in the United States; and (iii) the Grand Gulf plant is only in POS 5 during refueling outages for an average (mean) of 3.1% of the time. The core-damage frequency is also low relative to the frequency during POS 5 for internal initiators, as analyzed in the companion study by Sandia.

FOREWORD

(NUREG/CR-6143 and 6144)
Low Power and Shutdown Probabilistic Risk Assessment Program

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events potentially occurring only during full power operation. Some previous screening analyses that were performed for other modes of operation suggested that risks during those modes were small relative to full power operation. However, more recent studies and operational experience have implied that accidents during low power and shutdown could be significant contributors to risk.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL), with the seismic analysis performed by Future Resources Associates. Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied.

The objectives of the program are to assess the risks of severe accidents due to internal events, internal fires, internal floods, and seismic events initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program includes that of a level-3 PRA.

The results of the program are documented in two reports, NUREG/CR-6143 and 6144. The reports are organized as follows:

For Grand Gulf:

NUREG/CR-6143 - Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1

- Volume 1: Summary of Results
- Volume 2: Analysis of Core Damage Frequency from Internal Events for Plant Operational State 5 During a Refueling Outage
  - Part 1: Main Report
    - Part 1A: Sections 1 - 9
    - Part 1B: Section 10
    - Part 1C: Sections 11 - 14
  - Part 2: Internal Events Appendices A to H
  - Part 3: Internal Events Appendices I and J
  - Part 4: Internal Events Appendices K to M
- Volume 3: Analysis of Core Damage Frequency from Internal Fire Events for Plant Operational State 5 During a Refueling Outage
- Volume 4: Analysis of Core Damage Frequency from Internal Flooding Events for Plant Operational State 5 During a Refueling Outage
- Volume 5: Analysis of Core Damage Frequency from Seismic Events for Plant Operational State 5 During a Refueling Outage
- Volume 6: Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage
  - Part 1: Main Report
  - Part 2: Supporting MELCOR Calculations

FOREWORD (continued)

For Surry:

NUREG/CR-6144 - Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit-1

- Volume 1: Summary of Results
- Volume 2: Analysis of Core Damage Frequency from Internal Events During Mid-loop Operations
  - Part 1: Main Report
    - Part 1A: Chapters 1 - 6
    - Part 1B: Chapters 7 - 12
  - Part 2: Internal Events Appendices A to D
  - Part 3: Internal Events Appendix E
    - Part 3A: Sections E.1 - E.8
    - Part 3B: Sections E.9 - E.16
  - Part 4: Internal Events Appendices F to H
  - Part 5: Internal Events Appendix I
- Volume 3: Analysis of Core Damage Frequency from Internal Fires During Mid-loop Operations
  - Part 1: Main Report
  - Part 2: Appendices
- Volume 4: Analysis of Core Damage Frequency from Internal Floods During Mid-loop Operations
- Volume 5: Analysis of Core Damage Frequency from Seismic Events During Mid-loop Operations
- Volume 6: Evaluation of Severe Accident Risks During Mid-loop Operations
  - Part 1: Main Report
  - Part 2: Appendices

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The project could not have been effectively accomplished without the insights gained from the much larger shutdown-risk projects that have recently been completed under NRC support at Sandia National Laboratories (for Grand Gulf) and at Brookhaven National Laboratory (for Surry). We wish to thank all the many participants in those projects for their excellent work, and in particular Donnie W. Whitehead of Sandia and Tsong-Lun Chu of Brookhaven. In addition, the authors wish to thank Heather Schriner and John Forester of Sandia for their work in performing some specific systems analyses especially for this project, which were extremely important in tying down some of the key systems-analysis issues.

Finally, we wish to thank the staff members at Entergy's Grand Gulf power station for their cooperation, both during our walkdowns of the plant and afterward in furnishing technical information.

1. INTRODUCTION

1.1 Objective of the Report

This project has explored the likelihood of seismic-initiated core damage accidents during refueling outage conditions at the Surry (Unit 1) and Grand Gulf nuclear power plants. This report documents the Grand Gulf analysis, while a companion report (Ref. FRA/Surry, 1994) documents the Surry analysis. The project reported here is the second phase of a two-phase effort, the initial phase of which (Ref. FRA, 1991) was scoping in character, having been performed primarily to establish if the issue was judged to be important enough to justify the more extensive and more quantitative evaluation reported here. Throughout this report, we will refer to the earlier study as the "Phase-I study" and, where necessary for clarity, this study as the "Phase-II study".

The Phase-I results were preliminary in character, and served principally both to establish that the issue is important enough to merit further study, and to assure the study team that it is fully feasible to adapt the seismic-PRA methodology for full-power seismic PRA to this problem. The findings on both points were favorable enough to justify continuing with this Phase-II project.

1.2 Importance of the Issue

The issue of whether the risk of a large core-damage accident is significant when nuclear power plants are in shutdown conditions has received increasing attention in recent years. It is beyond our scope here to summarize the several recent studies of this issue: suffice it to state that the problem has been judged important enough that the U.S. Nuclear Regulatory Commission (NRC) has supported a major shutdown-risk research program for the past three years. The program has examined shutdown risk by performing in-depth probabilistic risk assessments (PRAs) at two U.S. nuclear power plants, Grand Gulf and Surry Unit 1, and has resulted in significant insights into various safety issues that may arise during shutdown conditions. The Grand Gulf study was carried out at Sandia National Laboratories, and the Surry study at Brookhaven National Laboratory, both under contract to NRC and both using subcontractors to supplement the in-house expertise at the two laboratories.

The results of this effort have recently been published (Ref. Sandia, 1994; BNL, 1994), and the findings are important: the overall core-damage frequency for potential accidents during the shutdown conditions that were examined is comparable to the core-damage frequency during full power operation.

However, the NRC shutdown-risk projects at Sandia and Brookhaven did not examine risks during shutdown that might arise from earthquake-initiated accidents: they both concentrated on so-called "internal initiators" such as transients, loss-of-coolant accidents, and loss of offsite power. They also both examined scenarios that might arise from internal fires and internal flooding. In the project being reported here for Grand Gulf, and in the companion project for Surry, the earthquake-initiator issue has been examined. Both of these projects are fully coordinated with these two larger studies of internal initiators.

1.3 Scope

The scope of this effort was limited to an examination of two nuclear power plants: Grand Gulf, a General Electric BWR/6 with a Mark-III containment, and Surry Unit 1, a Westinghouse 3-loop PWR with a subatmospheric containment. Grand Gulf, owned by Entergy Corporation, is located on the Mississippi River in east-central Mississippi, and is alone on its site. Surry-1, owned by Virginia Electric Power Company, is located in the tidewater region of southeastern Virginia, and has a nearly identical companion unit. There were two principal reasons for selecting these two plants:

- As mentioned above, both the Grand Gulf and the Surry-1 nuclear stations have been the subject of probabilistic shutdown risk studies (excluding seismic-initiated events) recently completed for the U.S. Nuclear Regulatory Commission. The Grand Gulf study was accomplished by Sandia National Laboratories (Ref. Sandia, 1994), and the Surry study was accomplished by Brookhaven National Laboratory (Ref. BNL, 1994). Appropriate models, data, and results from these studies were available for direct use in this seismic shutdown risk study. This information proved to be invaluable in completing this analysis.

- Both Grand Gulf (Ref. Brown et al., 1990) and Surry (Ref. Breeding et al., 1990) have also been the subjects of an extensive risk assessment for full-power conditions sponsored by the Nuclear Regulatory Commission and performed by Sandia National Laboratories, as part of the very large NUREG-1150 PRA study (Ref. NRC/1150, 1990). For Surry (but not for Grand Gulf), this NUREG-1150 analysis included an investigation of seismic-initiated accidents during full power conditions (Ref. Bohn et al., 1990), in which a limited amount of plant-specific seismic fragility information was developed for Surry's components and structures.

The NUREG-1150 full-power PRA study covered five plants. Seismic-initiated accidents at full power were analyzed for only two of them, the other (besides Surry) being Peach Bottom Unit 2, a BWR. Parts of the Peach Bottom external-events report (Ref. Lambright et al., 1990) also served as an important source of information for the Phase-I scoping project (Ref. FRA, 1991) that preceded this full-scope project.

This analysis has been limited to work analogous to a level-1 seismic PRA.

We have developed an estimate for the core-damage frequency from seismic events during certain conditions. Assessments of the likelihood of various post-core-damage plant-damage states (level-2 PRA) and of significant radioactive releases (level-3 PRA) are beyond the scope of this evaluation.

1.4 Assumptions

The following key assumptions have made for this study of Grand Gulf:

a. As mentioned, the recent shutdown risk study for Grand Gulf (Ref. Sandia, 1994) is the principal basis for the systems-analysis parts of our work here: indeed, we could not have accomplished this work at all without using the other very extensive analysis as our point of departure. The Sandia analysis contains numerous assumptions that are necessary to bound the scope and to simplify the detailed work, and we have adopted all of these assumptions without exception. Most importantly, we have adopted directly the loss-of-offsite-power (LOOP) event trees, including the definitions of the top events and underlying thermalhydraulic and other assumptions that support these event trees.

b. Only refueling outages have been considered in this seismic analysis. Indeed, only one specific Plant Operating State (POS) occurring within the standard refueling outages at Grand Gulf has been considered. This is Plant Operating State 5 ("cold shutdown"), discussed in more detail in Chapter 4. Outages for other reasons frequently occur at nuclear power plants, and they are of two broad types: controlled shutdowns and uncontrolled (rapid) shutdowns. Although these outages for other reasons can produce the same plant operating states but with configurations unique to the reason for the shutdown, resources in this analysis did not permit examining outage configurations other than POS 5 during a refueling outage.

c. We assume that the only seismic events of concern are those that cause loss-of-offsite-power (LOOP) transients. Seismic events of lower acceleration than those causing LOOP are expected to have a negligible probability of causing severe plant accidents for two reasons: (i) Critical plant equipment, including the residual heat removal (RHR) system, can withstand significantly higher accelerations than that which is sufficient to cause LOOP (Ref. Bohn et al., 1990; Lambright et al., 1990). Thus, loss of core-cooling capability will have a very low probability for seismic events too small to cause a LOOP. (ii) With offsite power available, sources of water sufficient to cool the core from alternative pumping sources will generally be available even if the RHR system fails.

d. We also assume that seismic-initiated LOOP is non-recoverable in the time frame of interest in this study (from about 1 to 24 hours). This is a reasonable and only slightly conservative assumption, because the LOOP initiator is most likely to arise from the seismic-caused failure of the ceramic insulators in the plant substation (Ref. Bohn et al., 1990; Lambright et al., 1990). Replacement of these insulators would likely require several hours at a minimum, and probably much longer. Additionally, other damage caused by the earthquake, for example to offsite transmission systems and offsite switchyards, would likely hamper efforts to recover offsite power.

e. The engineering methodology to develop probabilistic seismic fragility curves is described below (Chapter 2). The methodology uses a successive-screening approach, in which structures and equipment that are judged to be very strong under earthquake loading are screened out first, so that the number of items for which actual numerical fragilities must be developed is limited. Also, for a few items the study team was unable to obtain enough information, either because access to some parts of the plants was limited or because engineering information in the appropriate form was not available. For these items, generic fragilities have been used.

f. We have assumed that the seismic failures of identical equipment in similar locations are fully correlated. This means, for example, that when a postulated earthquake causes one service-water pump to fail, we have assumed that the other service-water pumps will also fail. This simplifying assumption is probably conservative in many cases, but perhaps not overly so for truly identical equipment. Sensitivity studies performed in several past seismic PRAs have shown that the bottom-line results are sensitive to this assumption at a level of about a factor of plus-or-minus two.

g. Equipment failure from seismic-induced relay chatter is outside the scope of this analysis. While relay chatter can be important (Ref. Budnitz, Lambert, and Hill, 1987), it is a complicated issue, and the resources to study it were not available within this project. In any event, relay chatter is being studied at every U.S. nuclear plant as part of the current IPEEE program (Ref. Chen et al., 1991), and it is expected that almost all of the relays that are especially sensitive to relay chatter will be identified and, if appropriate, modified in the course of the IPEEE studies.

h. As discussed below (see Chapter 3), for the seismic hazard we have used both the 1993 LLNL analysis (Ref. Sobel, 1993) and the EPRI analysis (Ref. EPRI, 1989).

1.5 Organization of the Report

The report is organized into three parts: first, we discuss the methodology used (Chapter 2 --- risk-assessment methodology and Chapter 3 --- seismic hazard analysis inputs). Next, we discuss the analysis and results (Chapters 4, 5, and 6); and then we summarize in Chapter 7 the principal findings of the analysis.

2. SEISMIC RISK ANALYSIS METHODOLOGY

The objectives of a full-scope seismic PRA of a nuclear power plant are to estimate the frequencies of occurrence of severe core damage, serious radiological releases, and consequences in terms of early fatalities, long term adverse health effects and property damage, and to identify significant contributors to plant risks. In this study, only a level-1 seismic PRA has been performed, leading only to estimates of the frequency of occurrence of severe core damage.

The key elements of a level-1 seismic PRA are:

1. Seismic hazard analysis - estimation of the frequency of various levels of seismic ground motion (acceleration) occurring at the site.

2. Seismic fragility evaluation - estimation of the conditional probabilities of structural or equipment failure for given levels of ground acceleration.

3. Systems/accident sequence analysis - modeling of the various combinations of structural and equipment failures that could initiate and propagate a seismic core damage accident sequence.

4. Evaluation of core damage frequency - assembly of the results of the seismic hazard, fragility, and systems analyses to estimate the frequencies of core damage for various accident sequences and for the plant as a whole.

In the following subsections, the methods used in each of the above stages of a level-1 seismic PRA are outlined.

2.1 Seismic Hazard Analysis

Seismic hazard is usually expressed in terms of the frequency distribution of the peak value of a ground-motion parameter (e.g., peak ground acceleration) during a specified time interval. Somewhat different approaches to implementing the basic methodology are documented in (Bernreuter et al, 1989) and (EPRI, 1989). The different steps of this analysis are as follows:

1. Identification of the sources of earthquakes, such as faults and seismotectonic provinces.

2. Evaluation of the earthquake history of the region to assess the frequencies of occurrence of earthquakes of different magnitudes or epicentral intensities.

3. Development of attenuation relationships to estimate the intensity of earthquake-induced ground motion (e.g., peak ground acceleration) at the site.

4. Integration of the above information to estimate the frequency of exceedance for selected ground motion parameters.

The hazard estimate depends on uncertain estimates of attenuation, upperbound magnitudes, and the geometry and seismic activity of the postulated sources. Such uncertainties are included in the hazard analysis by assigning probabilities or weights to alternative hypotheses

about these parameters. A probability distribution for the frequency of occurrence is thereby developed. The annual frequencies for exceeding specified values of the ground motion parameter are displayed as a family of curves with different weights (Figure 2-1).

A Bayesian estimate of the frequency of exceedance at any peak ground acceleration is obtained as the weighted sum of the frequencies of exceedance at this acceleration given by the different hazard curves; the weighting factor is the probability assigned to each hazard curve.

2.2 Seismic Fragility Evaluation

The methodology for evaluating seismic fragilities of structures and equipment is documented in (Ref. Ravindra and Kennedy, 1983) and (Ref. Kennedy and Ravindra, 1984). Seismic fragility of a structure or equipment item is defined as the conditional probability of its failure at a given value of the seismic input or response parameter (e.g., ground acceleration, stress, moment, or spectral acceleration). Seismic fragilities are needed in a PRA to estimate the conditional probabilities of occurrence of initiating events (e.g., loss of offsite power, large LOCA, small LOCA, RPV rupture) and the conditional failure probabilities of different mitigating systems (e.g., safety injection system, residual heat removal system, containment spray system).

The objective of fragility evaluation is to estimate the ground acceleration capacity of a given component. This capacity is defined as the peak ground motion acceleration value at which the seismic response of a given component located at a specified point in the structure exceeds the component's resistance capacity, resulting in its failure. The ground acceleration capacity of the component is estimated using information on plant design bases, responses calculated at the design analysis stage, as-built dimensions, and material properties. Because there are many variables in the estimation of this ground acceleration capacity, component fragility is described by a family of fragility curves; a probability or weighting value is assigned to each curve to reflect the uncertainty in the fragility estimation (Figure 2-2).

2.3 Analysis of Plant Systems and Accident Sequences

Frequencies of severe core damage and radioactive release to the environment are calculated by combining plant logic with component fragilities and seismic hazard estimates. Event and fault trees are constructed to identify the accident sequences that may lead to severe core damage and radioactive release.

The plant systems and sequence analyses used in seismic PRAs are based on the PRA Procedures Guide (Ref. NRC, 1983) and can generally be summarized for a level-1 PRA as follows:

1. The analyst constructs event and fault trees reflecting (a) failures of key system components or structures that could initiate an accident sequence and (b) failures of key system components or structures that would be called on to stop the accident sequence.

2. The fragility of each such component (initiators and mitigators) is estimated.

3. Fault trees and event trees are used to develop Boolean expressions for severe core damage that lead to each distinct accident sequence.

As an example, the Boolean expression for severe core damage in the Zion Probabilistic Safety Study (Ref. Zion, 1981) is:

$$\text{Boolean} = 4 + 8 + 10 + 14 + 17 + 21 + (12 + 22 + 26) * 9 \quad \text{(Eq. 2-1)}$$

The numbers represent components for which seismic fragilities have been developed, as described earlier. The symbols "+" and "*" indicate "OR" and "AND" operations, respectively. Plant level fragility curves are obtained by aggregating the fragilities of individual components according to Equation 2-1, using either Monte Carlo simulation or numerical integration. The plant level fragility means the conditional probability of severe core damage for a given peak ground acceleration at the site. The uncertainty in plant level fragility is displayed by developing a family of fragility curves; the weight (probability) assigned to each curve is derived from the fragility curves of components appearing in the specific accident sequence.

2.4 Evaluation of Core Damage Frequency

Plant level fragilities are convolved with the seismic hazard curves to obtain a set of doublets for the accident frequency,

$$\{ < p_{ij}, f_{ij} > \} \quad \text{(Eq. 2-2)}$$

where f_{ij} is the seismically-induced accident frequency and p_{ij} is the discrete probability of this frequency,

$$p_{ij} = q_i \, p_j \quad \text{(Eq. 2-3)}$$

$$f_{ij} = f_i(a) \int (dH_j/da) \, da \quad \text{(Eq. 2-4)}$$

Here, H_j represents the j^{th} hazard curve, f_i the i^{th} accident fragility curve; q_i is the probability associated with the i^{th} fragility curve and p_j is the probability associated with the j^{th} hazard curve.

The above equations state that the convolution between the seismic hazard and plant level fragility is carried out by selecting hazard curve j and fragility curve i; the probability assigned to the accident frequency resulting from the convolution is the product of the probabilities p_i and q_j assigned to these two curves. The convolution operation given by Equation 2-4 consists of multiplying the occurrence frequency of an earthquake peak ground acceleration between a and a + da (obtained as the derivative of H_j with respect to a) with the conditional probability of the accident fragility curve, and integrating such products over the entire range of peak ground accelerations zero to infinity. In this manner, a probabilistic distribution on the frequency of a given accident sequence can be obtained.

Severe core damage occurs if any one of the accident sequences occurs. By probabilistically combining the various sequences, the plant level fragility curves for severe core damage are obtained. Integration of the family of fragility curves over the family of seismic hazard curves yields the probability distribution function of the occurrence frequency of severe core damage (Figure 2-3).

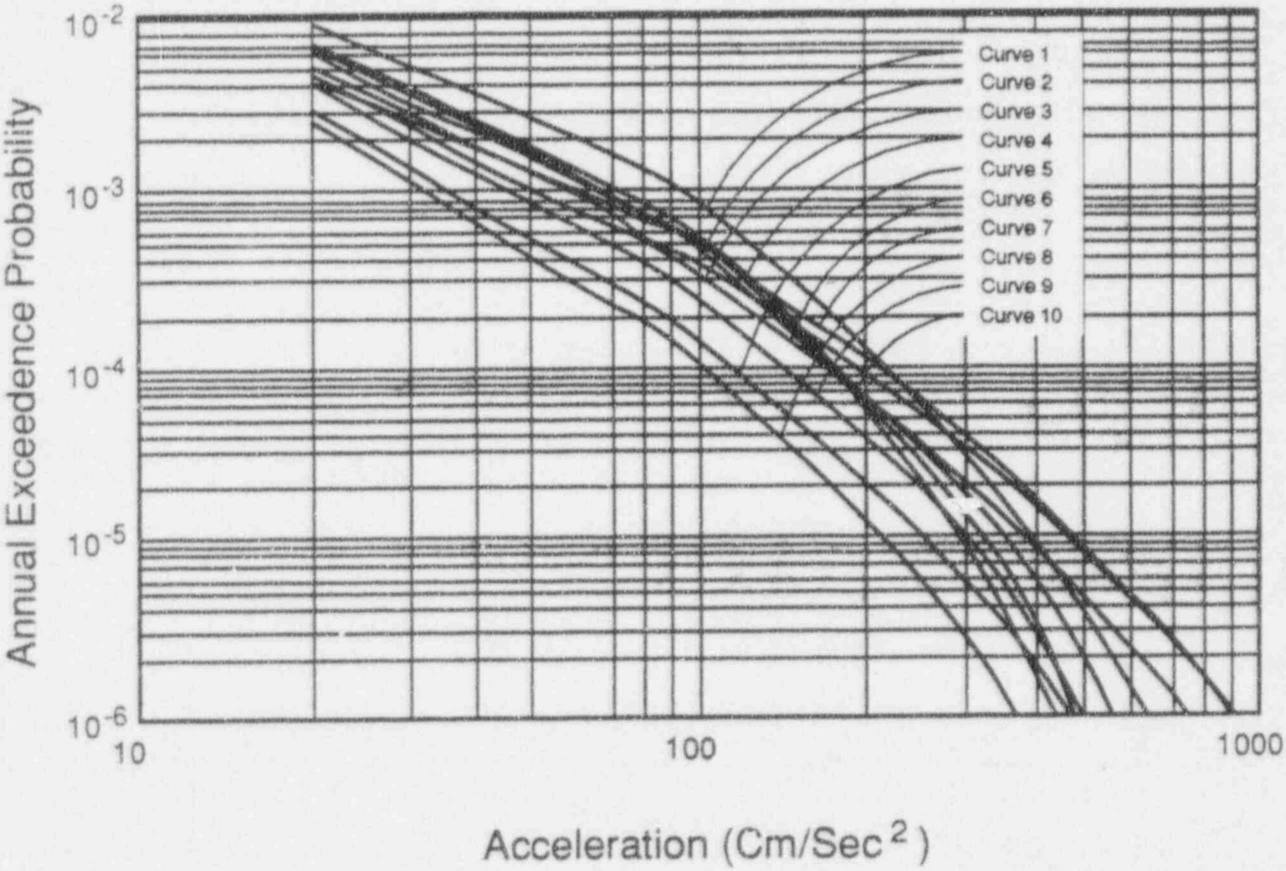


Figure 2-1: Seismic Hazard Curves For A Nuclear Power Plant Site

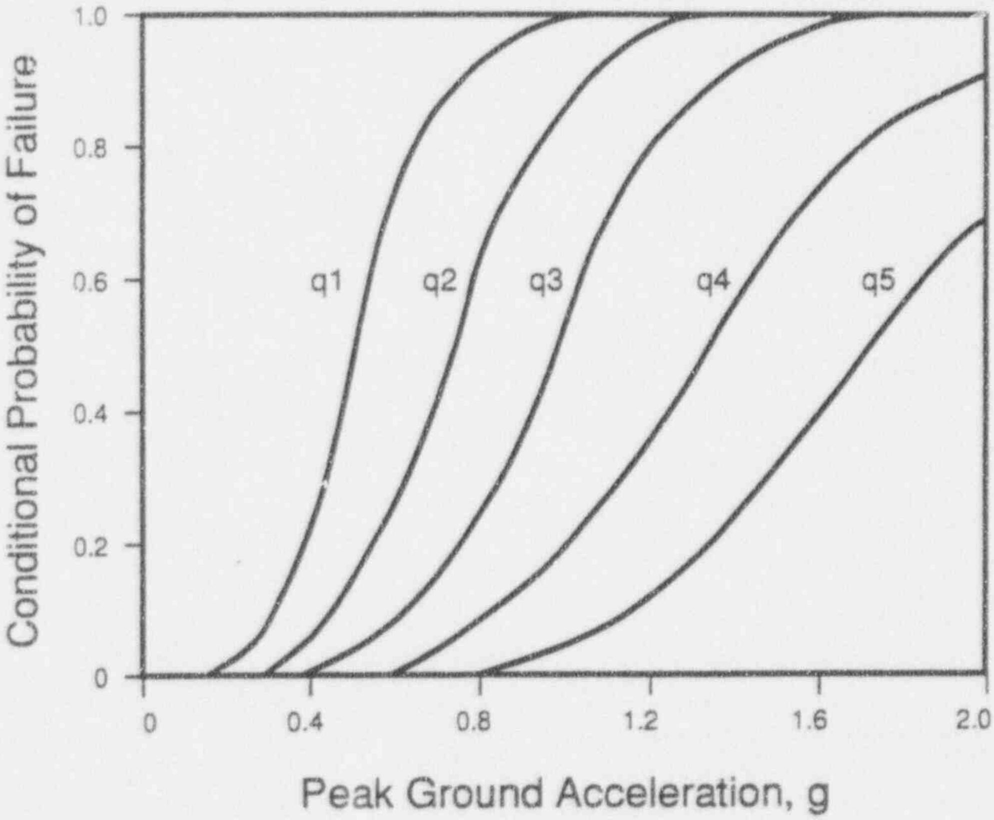


Figure 2-2: Family of Fragility Curves for a Component

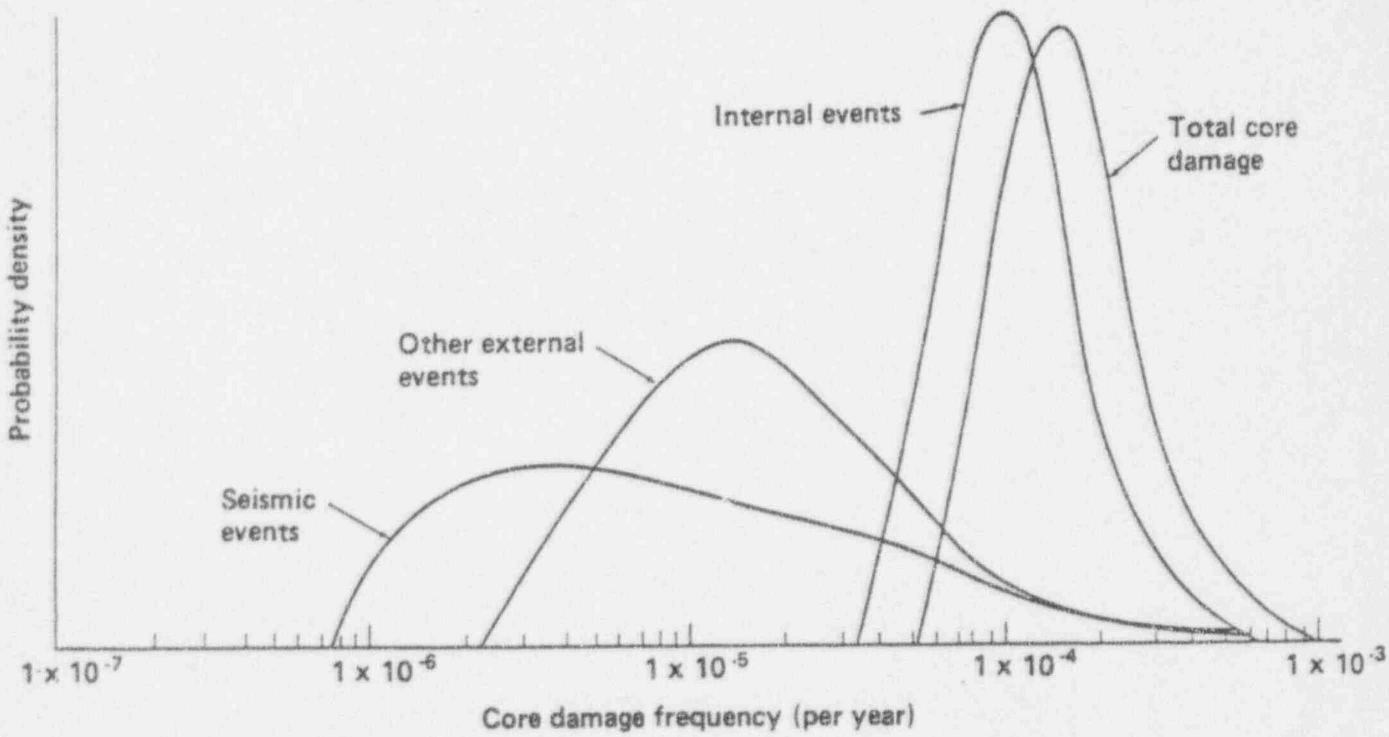


Figure 2-3: Probability Distribution of Seismically-Induced Severe Core Damage Frequency

3. SEISMIC HAZARD ANALYSIS

3.1 EPRI and LLNL Seismic Hazard Studies

Seismic hazard curves are needed to estimate the core-damage frequencies of different plant operating states. Typically, a site-specific seismic hazard analysis is performed to obtain the hazard curves. For the Grand Gulf site, two such studies have already been conducted and the results are available in a usable form.

Figure 3-1 is the set of seismic hazard curves for the Grand Gulf site developed by the Electric Power Research Institute (Ref. EPRI, 1989). Shown are the mean, median, 15th-percentile and 85th-percentile curves. Each curve is the annual probability of exceedance plotted against the peak ground acceleration. Similar curves developed by the Lawrence Livermore National Laboratory (Ref. LLNL, 1993) are shown in Figure 3-2. These two sets of hazard curves are used in the seismic risk quantification reported in Chapter 6. Note that the LLNL hazard curves being used are from the recent 1993 LLNL study (Ref. Sobel, 1993), not the earlier 1989 LLNL study (Ref. Bernreuter et al., 1989) that has now been superseded.

For fragility evaluation, a site-specific ground motion response spectrum is needed. In NUREG-1407 (Ref. Chen et al., 1991), it is recommended that either of the spectral shapes developed in the above EPRI and LLNL studies could be used. The median spectral shape corresponding to a 10,000-year return period along with variability estimates given in the LLNL study (Ref. Sobel, 1993) is suggested for this purpose (Figure 3-3).

Two important considerations arise in the use of these seismic hazard curves. These curves provide estimates of probability of exceedance per year. However, the plant outages only extend for a fraction of the year. This must be taken into account in the estimation of frequencies of different plant operating states during shutdown (See Section 4.4 below). Secondly, the objective of this study is to assess the contribution of the seismic-induced risks in shutdown conditions and to compare it to the risks from other events during refueling outages. It would be interesting to know how the seismic-induced shutdown risk varies from nuclear plant site to site. Towards this end, we have performed sensitivity studies by "moving" the Grand Gulf power station to different sites elsewhere in the eastern United States (with different seismic hazard curves) and estimating the impact. This is discussed in Chapter 6.

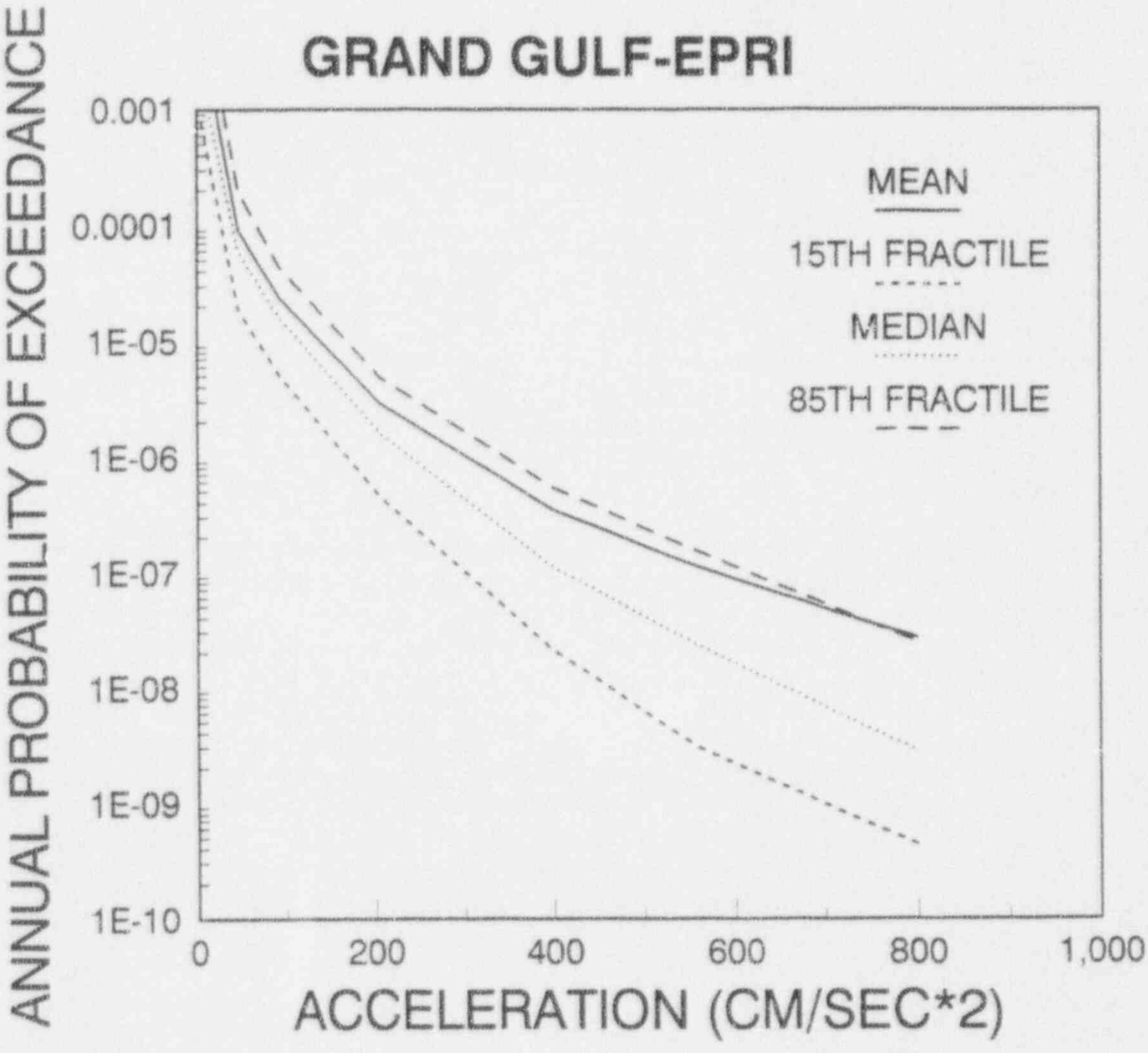


Figure 3-1: EPRI Seismic Hazard Curves for Grand Gulf Site

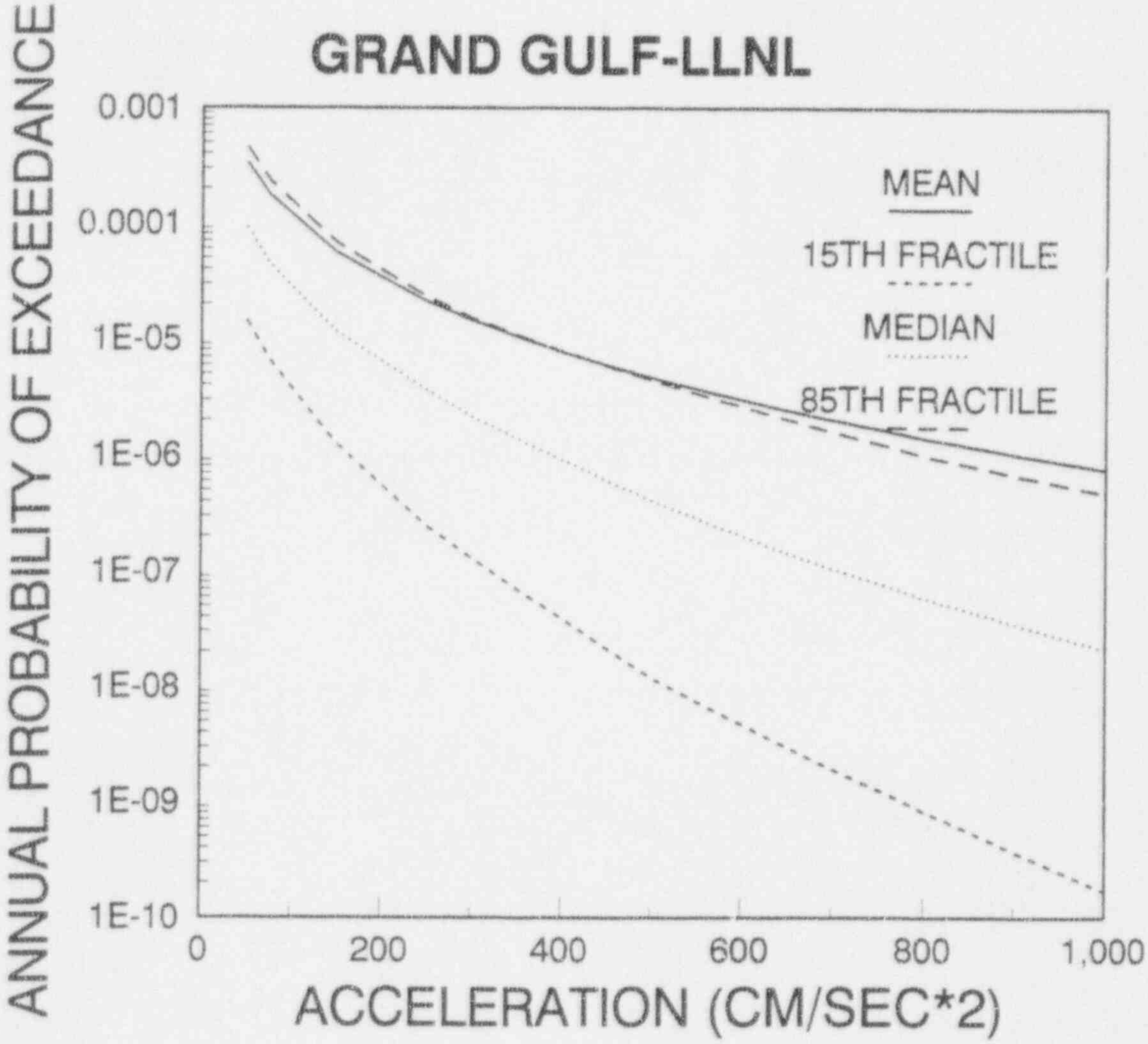


Figure 3-2: LLNL Seismic Hazard Curves for Grand Gulf Site

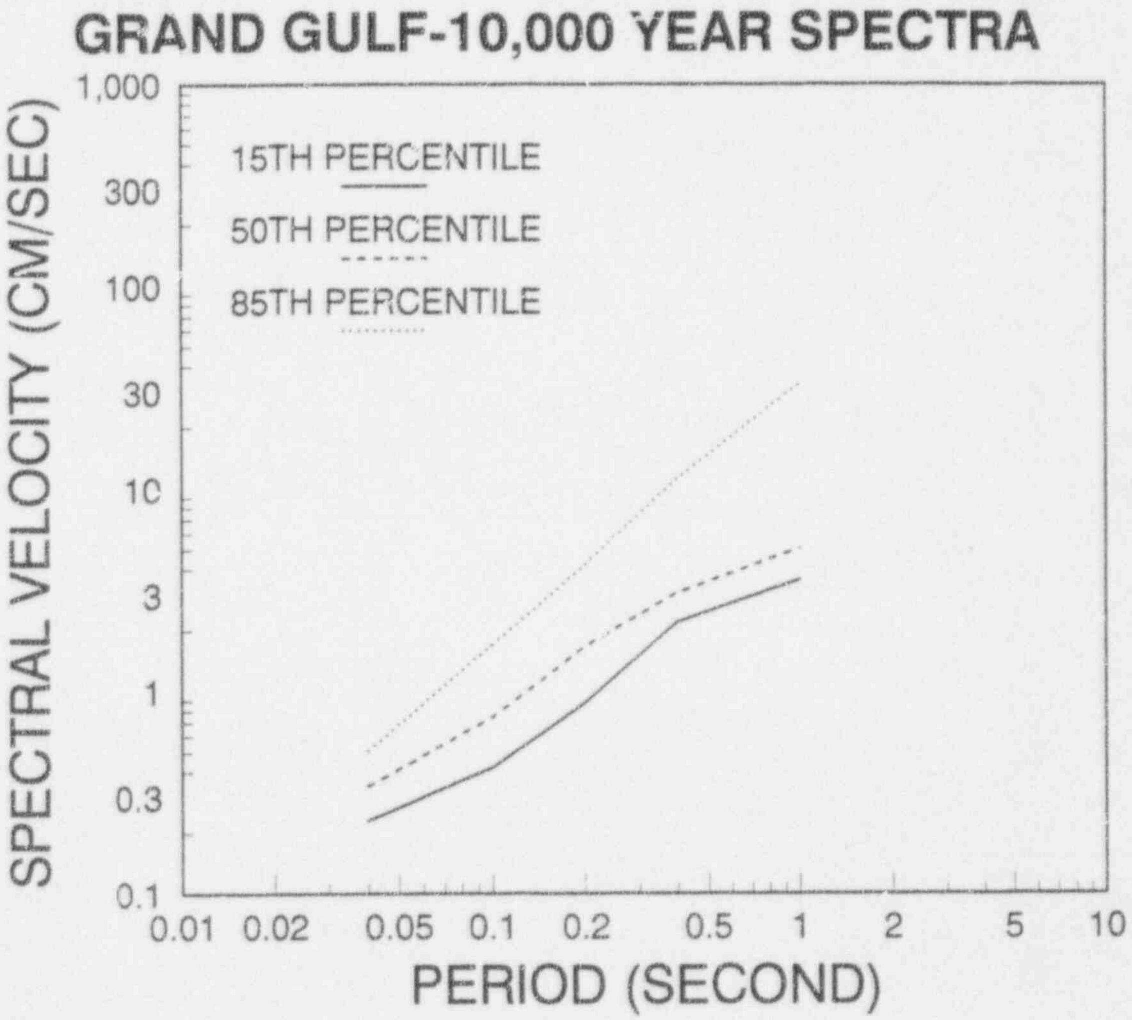


Figure 3-3: 10,000 year Return Period Ground Response Spectrum for the Grand Gulf Site

4. SYSTEMS ANALYSIS FOR GRAND GULF

4.1 Identification of Accident Sequences

This section presents the systems-analysis part of this study, leading up to a very simple expression for the cut-set that, when quantified, will provide an estimate of core damage frequency.

4.1.1 Assumptions

Several assumptions were made at the outset of the study to simplify the analysis and provide meaningful results. These assumptions are as follows:

4.1.1.1 Plant Operating State (POS) 5 Only: As mentioned briefly in the introductory material in Section 1.4 above, the Sandia study of internally-initiated accidents during shutdown (Ref. Sandia, 1994) examined only refueling-outage accidents occurring during Plant Operating State (POS) 5, which is only one of the many possible POS states during shutdown.

As discussed in detail in the Sandia report (Ref. Sandia, 1994), POS 5 is the "cold shutdown" state, consisting of Operating Condition 4 (temperature $\leq$ 200 degrees F) and Operating Condition 5 until the vessel head is off. In POS 5 the vessel head is on, but this is a transition state, either "going down" in power prior to refueling or "coming up" to power after refueling. The meaning of the phrase "going down" is that the plant is in transition from POS 4 ("hot shutdown") to POS 6 ("vessel head off, water level raised to steam lines"), while the phrase "coming up" means that the plant is coming back to power and is in transition from POS 6 to POS 4. In the Sandia report, the two conditions are called POS 5D ("5 down") and POS 5U ("5 up"), respectively.

4.1.1.2 Loss of offsite power: It was determined that the only seismic events which are of concern in this study are those that cause loss of offsite power (LOOP). Seismic events of lower ground-motion than those causing LOOP are found to have a negligible probability of causing severe plant accidents for two reasons. First, as discussed in Chapter 5 (see Table 5.3-1), all other critical plant equipment can withstand significantly higher seismic accelerations than that which is sufficient to cause LOOP. Thus, loss of core-cooling capability or other transients will have a negligible probability for seismic events smaller than those causing a LOOP. Second, based on the internal-events analysis by Sandia (Ref. Sandia, 1994), when offsite power is available numerous alternate sources of water sufficient to cool the core would be available with very high probability. Thus, only LOOP-initiated accidents are considered in this study.

4.1.1.3 LOOP recovery: It is assumed that seismic-initiated LOOP is non-recoverable in the time frame of interest in this study. While this is a slightly conservative assumption, it is considered reasonable because the LOOP initiator is most likely to be failure of the ceramic insulators in the plant substation (see Chapter 5). Replacement of these insulators would very likely require more than a few hours. Furthermore, other damage caused by the earthquake would likely hamper efforts to repair the switchyard. Offsite damage to other parts of the electrical power grid that supplies power to the plant would also be expected, which would likely delay efforts to restore offsite power. Thus, recovery of offsite power is assumed not to occur after an earthquake large enough to cause LOOP.

4.1.2 Event Trees

In order to identify accident sequences for the study, seismic shutdown event trees must be developed. These event trees identify the functions and systems that are important in evaluating the progression of accidents during POS 5 following the occurrence of a seismic event.

As a practical matter, the event trees used in this seismic analysis are the internal-events LOOP event trees taken directly from the Sandia study. These event trees are very complicated, multi-layered trees that can be found in the Sandia report (Ref. Sandia, 1994). They will not be reproduced here.

4.2 Seismic Capacities of Grand Gulf Safety Systems

In Chapter 5 (Table 5.3-1), the seismic fragility values for various items of equipment are presented. Based on this table, the following hierarchy of equipment can be developed, listed in order of increasing strength. Here the weakest equipment is listed first, based on median seismic fragility (in units of PGA, peak ground acceleration):

Component	Median Seismic Fragility (PGA)
Loss of Offsite Power	0.30 g
Condensate Storage Tank (CST)	0.59 g
Standby Service Water (SSW) Pumps	0.82 g
all other equipment	(above 1.50 g)

The key safety systems can be grouped into those that depend on the three items listed above (LOOP, CST, SSW pumps), and those that do not. If an item does not depend on one of these three, it is considered sufficiently rugged that for the purposes of this analysis we can assume that it will not fail --- here "sufficiently rugged" means that the item's median seismic fragility is sufficiently strong that it will not fail for earthquakes with PGA up to about 1.5g, an earthquake almost surely large enough to cause core damage during shutdowns at Grand Gulf, but an earthquake that, if it can happen at all, has an extremely low annual frequency, well below 10^{-8}/year (see Chapter 3).

4.2.1 The "sufficiently rugged" items

The sufficiently rugged systems and components include the following:

- the reactor protection (scram) system;
- all electrical components except offsite power, including inverters, batteries, and diesel generators; this explicitly includes the crucial Emergency DC Power System;
- all motors and associated equipment;
- the turbine and generator;
- all piping;

- o the reactor pressure vessel;
- o all valves, and in particular all relief valves and associated hardware;
- o all large pumps except the SSW pumps;
- o all instruments;
- o air-handling and room-cooling equipment; and
- o all tanks except the CST.

RCIC: One of the key safety systems during power operation is the Reactor Core Isolation Cooling (RCIC) System, which provides high-pressure coolant injection to the reactor vessel during accidents. RCIC's particular feature is its steam-driven pump that is AC-independent. Unfortunately, although RCIC is seismically rugged, in POS 5 the pressure and temperature are reduced sufficiently that there is not enough steam to operate the RCIC system, so it will not be considered further.

Structures: All of Grand Gulf's buildings and other structures are seismically rugged, based on generic seismic-capacity considerations. Besides the various buildings, this specifically includes the suppression pool, the drywell structure, and all containment structures.

4.2.2 Systems and components that are not necessarily "sufficiently rugged"

The principal systems appearing on the LOOP event trees are systems to deliver water to the reactor vessel, systems involved with the RHR heat-removal function, and systems to provide pressure relief. We will next discuss all of these systems and what limits their seismic capacities.

In order to clarify what will follow, we will make one key point here as a preview of the ultimate conclusion:

The seismic capacity of the Standby Service Water (SSW) system (median 0.82g PGA) controls the seismic capacity of almost everything else.

This is because:

- o The seismic capacity of the Emergency AC Power system is controlled by the seismic capacity of the SSW system (median 0.82g PGA).
- o All of the other key systems below depend either on SSW directly or on Emergency AC Power (which itself depends on SSW). Therefore, the seismic capacities of all of these key systems are controlled by the seismic capacity of the SSW system (median 0.82g PGA).

We will begin with the SSW system and the Emergency AC Power System, and then discuss the other systems:

Standby Service Water (SSW) System: The SSW system provides heat removal from plant equipment during emergency shutdown of the plant. It has three trains, each with a motor-

driven pump, motor-operated valves, and heat exchangers. Train C is dedicated to HPCS. All three SSW pumps are vertical centrifugal pumps, taking water from the cooling tower basins. The SSW depends on AC for pumps and valves, and DC for initiating the actuation logic. The controlling seismic fragility is that of the vertical centrifugal service-water pumps at 0.82g.

Emergency AC Power System: The Emergency AC Power System consists of the AC power divisions, driven by the emergency diesel generators, that are used to shut down the plant safely after a transient or LOCA. A key dependency is that the SSW system is needed to supply cooling water to the diesel jacket water coolers. Another key dependency in a LOOP transient is on the emergency DC system, because the AC divisions require DC power for circuit breaker control power, diesel generator field flashing, and the diesel fuel oil booster pump. The controlling fragility is that of the SSW system, whose vertical centrifugal pumps have a median fragility at 0.82g. The diesel generators themselves have a median fragility > 1.5g.

High Pressure Core Spray (HPCS) System: The HPCS system can provide coolant to the reactor vessel, and has its own dedicated electrical support system. It can draw from either the CST (the primary source) or the suppression pool by automatic switchover on low CST level or high SP level. It requires AC power to operate the pumping and DC power for control. Its controlling seismic fragility in LOOP transients is that of the emergency AC system at a median PGA level of 0.82g.

Low Pressure Core Spray (LPCS) System: The single-train LPCS system provides low-pressure coolant to the vessel. Its suction is from the suppression pool. Its motor-driven pump and valving depend on AC power, and it needs DC power for the actuation relay logic and pump breaker. LPCS's controlling seismic fragility in a LOOP transient is that of the emergency AC system at 0.82g.

Low Pressure Coolant Injection (LPCI) System: The three-train LPCI system provides low-pressure coolant to the vessel, taking its suction from the suppression pool. Two trains have RHR pumps that are in common with other RHR modes, the SPC, SDC, and CS modes. These two trains have heat exchangers downstream of the pump, while the third train is injection-dedicated and has no heat exchangers or other RHR function. LPCI's motor-driven pumps, RHR pump cooling, RHR pump room cooling, and RHR valving depend on AC power. LPCI also needs DC power for the actuation relay logic and pump breakers, and depends on the SSW system for pump cooling. The LPCI system's controlling seismic fragility is that of the SSW's vertical centrifugal pumps at 0.82g, and also of the emergency AC system at 0.82g (because emergency AC depends on SSW).

Suppression Pool Cooling (SPC) System: The two-train SPC system removes decay heat from the suppression pool during accidents, taking suction from the pool. It is one of the several modes of the RHR system, and shares several components with other RHR modes: the pumps are common with the LPCI, CS, and SDC modes, and each train's suppression pool suction valve is common to the LPCI and CS modes. Each train has two heat exchangers in series downstream of the pump. The SPC mode must be manually initiated and aligned. The SPC system depends on AC power for pumps and valves, DC power for initiating the actuation logic, and SSW to cool the pumps. The controlling seismic fragility is that of the SSW's vertical centrifugal pumps at 0.82g, and also of the emergency AC system at 0.82g (because emergency AC depends on SSW).

Shutdown Cooling (SDC) System: The two-train SDC system removes decay heat from the vessel during accidents, taking suction from one recirculation pump's suction line in the primary system. Each train has two heat exchangers in series downstream of the pump. It is

one of the several modes of the RHR system, and shares several components with other RHR modes: the pumps are common with the SPC, LPCI, and CS modes, and each train's heat exchangers are common to those of the LPCI, SPC, and CS modes. The SDC mode must be manually initiated and aligned. The SDC system depends on AC power for pumps and valves, DC power for initiating the actuation logic, and SSW to cool the pumps. The controlling seismic fragility is that of the SSW's vertical centrifugal pumps at 0.82g, and also of the emergency AC system at 0.82g (because emergency AC depends on SSW).

Containment Spray (CS) System: The two-loop CS system suppresses pressure in the containment during accidents, taking suction from the suppression pool. Each loop has two heat exchangers in series. It is one of the several modes of the RHR system, and shares several components with other RHR modes: the pumps are common with the SPC, LPCI, and SDC modes, the suppression pool suction valve is common to the SPC and LPCI modes, and each loop's heat exchangers are common to those of the SDC and SPC modes. The CS mode is automatically initiated and controlled, but operators can intervene if autoactuation fails. The CS system depends on AC power for pumps and valves, DC power for initiating the actuation logic, and SSW to cool the pumps. The controlling seismic fragility is that of the SSW's vertical centrifugal pumps at 0.82g, and also of the emergency AC system at 0.82g (because emergency AC depends on SSW).

Standby Service Water (SSW) Cross-Tie System: The SSW cross-tie system can be used as a backup when other sources of emergency vessel injection have failed. It must be manually aligned and actuated. It consists of one train of the SSW system and one train of the LPCI system. Because LPCI's controlling seismic fragility is that of the SSW system at 0.82g, SSW's seismic fragility controls both trains at 0.82g.

Suppression Pool Makeup (SPMU) System: The SPMU system provides water from the upper containment pool to the suppression pool following a LOCA, through two lines that penetrate the side walls in the separator storage area of the upper containment pool. The water flows by gravity but has two normally-closed, motor-operated butterfly valves in series. It requires AC power to operate the valves and DC power for the initiation logic. The SPMU system's controlling seismic fragility is that of the emergency AC system at 0.82g.

Control Rod Drive (CRD) System: The CRD system, a backup source of high-pressure injection, draws from the CST and its controlling seismic fragility is that of the large CST tank, 0.59g. All other dependencies have higher seismic capacity.

Firewater (FW) System: The FW system is a backup low-pressure injection system, with three trains, one using a motor-driven pump and two using diesel-driven pumps. The suction is from two dedicated 300,000-gallon storage tanks. To use the FW system for vessel injection, operators must manually align and actuate the system. The diesel-driven pumps have no outside interfaces or dependencies, with self-contained batteries. The FW system's major importance, besides for fighting fires, is when blackout conditions occur. Its controlling seismic fragility is that of its large storage tanks, which are unanchored and which were found during the walkdown to be too weak to be relied on during any sizeable earthquake. Therefore, the FW system is not included in our analysis.

4.2.3 Summary of System Fragilities

As the above discussion indicates, most of Grand Gulf's safety-system fragilities are controlled either directly by the SSW system's median fragility of 0.82g or indirectly by SSW through the fact that the emergency AC power system's median fragility is controlled by SSW. The CRD

and FW systems have weaker seismic capacities, the CRD system capacity being controlled by the CST capacity (median 0.59g) and the FW system being seismically very weak because its large FW tanks are both unanchored.

The following short table indicates the controlling capacity for each system discussed:

Safety System	Item that Controls Seismic Capacity	PGA at which Median Fragility Occurs
controlling systems:		
FW system	FW tanks	very weak
offsite power	ceramic insulators	0.30g
CRD system	condensate storage tank	0.59g
SSW system	vertical centrifugal pumps	0.82g
systems controlled by other systems:		
AC EPS system	SSW	0.82g
LPCI system	SSW, AC	0.82g
SPC system	SSW, AC	0.82g
SDC system	SSW, AC	0.82g
CS system	SSW, AC	0.82g
SSW cross-tie	SSW	0.82g
HPCS system	AC	0.82g
LPCS system	AC	0.82g
SPMU system	AC	0.82g
DC EPS system	none	very strong

4.3 Systems-Analysis Discussion

4.3.1 Core Damage Due to Seismic Failures Only

Based on the above observations about the seismic ruggedness of the various Grand Gulf safety systems, a few broad conclusions can be simply stated:

- At earthquake levels near 0.30g (PGA), Grand Gulf will experience a LOOP (loss of offsite power) event, but no other seismic failures are expected to occur.
- At larger earthquake levels, near 0.59g (PGA), the Condensate Storage Tank will fail.
- At even larger earthquake levels, Standby Service Water fails (median capacity 0.82g PGA), and this failure causes essentially all other safety systems to fail.

The systems implications of each of these three conditions will be discussed in turn next:

Seismic failure of LOOP alone: The seismic failure of LOOP is predicted to occur at a mean frequency of less than 10^{-5}/year (LLNL seismic hazard curves) or even smaller frequencies (EPRI seismic hazard curves). Even for the higher LLNL seismic-hazard rate, the only way to generate a core-damage accident with seismic-initiated LOOP alone is to experience a series of non-seismic failures, as captured in the Sandia internal-initiators analysis (Ref. Sandia, 1994).

However, based on that analysis, all such seismic-LOOP-only cut sets have frequencies below 10^{-9}/year, taking into account the very small frequency of seismic-caused LOOP. They are therefore unimportant.

Seismic failure of LOOP and CST: This combination of failures will occur at earthquake levels near 0.59g PGA, the median capacity of the CST. This is predicted to occur at a mean frequency of near 10^{-6}/year (LLNL seismic hazard curves) or near 10^{-7}/year (EPRI seismic hazard curves). However, with no other failures besides LOOP, the CST is not needed for any particular safety function. Therefore, as with LOOP alone, this combination can be screened out based on the Sandia internal-initiators analysis (Ref. Sandia, 1994).

Seismic failure of LOOP and SSW: At earthquake levels near 0.82g PGA, the SSW system will fail due to the failure of the vertical-shaft service-water pumps. When this happens, core damage is inevitable. As seen above, this is because in the absence of offsite power essentially all of the key safety systems depend on SSW either directly or indirectly through the dependence of emergency AC power on SSW.

This is confirmed by a special analysis conducted by the Sandia group, from which two key findings emerge:

- o LOOP and CST failures together do not lead to core-damage accidents without other failures that are very unlikely (see the discussion immediately below in subsection 4.3.2).

- o LOOP and SSW failures together inevitably lead to core-damage.

Therefore, the single dominant core-damage accident sequence has the following cut set, where AND is the Boolean "and":

cut set for core damage = (seismic LOOP) AND (seismic SSW failure).

Because "seismic LOOP" occurs at such a low earthquake level compared to "seismic SSW failure" (median of 0.30g vs. 0.82g), the core damage frequency is actually simply the frequency of an earthquake that leads to SSW failure, during the fraction of the year in which Grand Gulf is refueling in POS 5:

core-damage frequency/year =

(fraction of time in POS 5) x (frequency/year of seismic SSW failure).

This is a simple calculation, the results of which are presented in Chapter 6, using the convolution methodology discussed in Chapter 2.

4.3.2 Core Damage Due to a Combination of Seismic Failures, Non-Seismic Failures, and Human Errors

In addition to the core-damage sequences discussed above, in which core damage arises exclusively from seismic-caused failures, there is the possibility of additional core-damage sequences in which non-seismic failures and human errors also contribute. To study whether any such sequences are important at Grand Gulf, the non-seismic systems analysis that was developed by the Sandia team (Ref. Sandia, 1994) was used. Special systems analyses were

performed by the Sandia team, who searched for any sequences with the following properties (here the asterisk [*] means Boolean "AND" and the plus sign [+] means Boolean "OR"):

(Seismic Loss of Offsite-Power)
  * (Seismic Loss of CST)
  * [(All other Non-Seismic Failures) + (Human Errors)]

Given that the firewater (FW) system is very weak under seismic loads (see above) and therefore cannot be relied on after an earthquake, the Sandia analysis identified no such sequences with mean contingent failure probabilities (for the non-seismic failures + human errors) greater than about 10^{-4}, taking into account the duration of time (3.1%) in POS 5, and also the likelihood of being in the appropriate "time window" that Sandia used in their analysis to differentiate various situations within POS 5.

The mean frequency of the combinations of seismic failures contributing to the above sequence type is less than 10^{-5}/year. Therefore, this sequence type contributes negligibly (in the range of 10^{-9}/year or less) to the overall mean core-damage frequency at Grand Gulf. It can thus safely be screened out and ignored, and in this analysis it has been.

4.4. Consideration of Duration in POS 5 During Refueling Outages

This section provides a discussion of the estimated duration, in hours, for the Plant Operating State (POS 5) of interest in this study. This time interval is important in estimating the annual core damage frequency from seismic events because the estimate must account for the fraction of the time during a given year that the plant is in the POS of interest.

Only refueling outages were considered in this seismic analysis. Outages for other reasons frequently occur at nuclear power plants, and they are of two broad types: controlled shutdowns and uncontrolled (rapid) shutdowns. These outages, for reasons other than refueling, can produce the same plant operating states but with unique configurations based on the reason for the shutdown. However, this analysis did not examine outage configurations other than those for refueling. The reason for this limitation is because the companion study from which this analysis derived considerable input information (Ref. Sandia, 1994) also considered only shutdown states during refueling outages.

As part of the Sandia shutdown risk study (Ref. Sandia, 1994), an evaluation was completed to estimate the time that the Grand Gulf plant was in Plant Operating State (POS) 5. Information for all of Grand Gulf's refueling outages except the very first one was collected and analyzed, in order to arrive at the fraction of the time in which the plant was in both POS 5D and 5U. These fractions were found to be as follows:

fraction of time in POS 5D	0.008
fraction of time in POS 5U	0.023
TOTAL FRACTION OF TIME IN POS 5	0.031

This means that, over its operating history, Grand Gulf was in POS 5 during refueling outages for an average of 3.1% of the time, or an average of about 270 hours/year. Because there is a wide distribution of times spent in POS 5 among the several refueling outages in Grand Gulf's operating history, the value is uncertain: the Sandia report estimates the error factor on this value to be plus-or-minus a factor of about 2.

5. SEISMIC FRAGILITY EVALUATION FOR GRAND GULF

5.1 Introduction

In this Chapter, the details concerning the seismic fragility evaluation for Grand Gulf be discussed. Section 5.2 will discuss the seismic walkdown, and Section 5.3 will discuss the seismic fragility evaluation.

5.2. Seismic Walkdown of Grand Gulf

5.2.1 General

The seismic walkdown of Grand Gulf was performea in December, 1991 and included the following personnel:

James Owens	Entergy
Amir Shahkarami	Entergy
Robert J. Budnitz	Future Resources Associates
Peter R. Davis	PRD Consulting
M.K. Ravindra	EQE
Wen H. Tong	EQE.

The purposes of the seismic walkdown were to:

1. Pre-screen all equipment items that have sufficiently high seismic capacities.

2. Clearly define the failure modes of components which are not pre-screened. Review and gather detailed information and measurements on equipment and structures for performing seismic fragility evaluations.

3. Identify spatial system interaction (SI) concerns that are judged to be potentially serious problems (such as heavy, questionably secured space heaters or ceiling fixtures over critical batteries, etc.)

One of the primary objectives of the seismic walkdown was to screen all equipment items that are judged to have High Confidence of Low Probability of Failure (HCLPF) capacities higher than 0.3g peak ground acceleration (pga).

Prior to the walkdown, an initial list of components was made available based on the components modeled in the internal event fault trees that were judged to be critical from a seismic standpoint. Using the plant layout drawings the equipment locations were identified. The following provides a list and brief discussion of the structures and equipment identified as target areas for seismic walkdown review.

Yard Equipment and Structures: Yard equipment and structures reviewed during the walkdown included:

- o Auxiliary Building
- o Containment Building
- o Control Building
- o Diesel Generator Building
- o Standby Service Water Cooling Tower and Storage Basins
- o Condensate Water Storage Tank.

Seismic reviews of these structures were performed using primarily the as-built structural drawings. During the walkdown, seismic separation gaps between the auxiliary building and the diesel generator building and between the containment building and the auxiliary building were verified.

Equipment: The Grand Gulf plant layout drawings were reviewed prior to the walkdown to locate mechanical and electrical equipment identified on the equipment list. Seismic Evaluation Walkdown Sheets (SEWS) were prepared for each component to record detailed information for fragility evaluation.

All of the major equipment components identified by the system analysts were reviewed during the walkdown in accordance with the procedures discussed in the section below. Generically reviewed components included piping, valves, ducting, cable trays, and instrument racks.

5.2.2 Walkdown Procedures

5.2.2.1 Structures: Information necessary for seismic evaluation of civil structures is normally obtained from design drawings rather than walkdowns. Drawings are reviewed to obtain a general understanding of construction and configuration of the structures and to identify any specific data to be obtained during the walkdown. The walkdown of structures is to determine the following:

- o Verify that the structures are in general conformance with the design drawings

- o Identify any gross deficiencies that might result in reduced capacities

- o Confirm that structural separations indicated on the drawings are provided

- o Obtain structural details not available from the drawings.

5.2.2.2 Equipment: Components which were reasonably accessible and located in non-radioactive or moderately radioactive environments were reviewed. To assess components in high-radioactive environments, or within contaminated containment, smaller inspection teams and more hurried inspection were employed. For components which were not accessible, the equipment inspection relied on alternate means such as photographic inspection and seismic reanalysis.

In the event that the walkdown team had a reasonable basis for assuming that a group of components is similar and is similarly anchored, then only a single component of this group was inspected. The "similarity-basis" was developed during the walkdown. The one component of each type which was selected was thoroughly inspected. The other components were then reviewed during the walkdown to ensure similarity with the selected unit. Outliers, lack of similarity, anchorage which was different from that snown on drawings or prescribed in the criteria for that component, potential SI problems, situations that are at odds with the team

members' past experience, and any other areas of concern were looked for during the walkdown. When such concerns surfaced, the limited sample size of one component of each type for thorough inspection was increased. The increase in sample size which should be inspected depended upon the number of outliers and different anchorages, etc., which were observed. The following provides specific procedures used for the review of different classes of equipment inspected during the walkdown.

Tanks. Design drawings for the tanks and their foundations and or supports were reviewed to obtain a general understanding of the tank configurations and anchorage details. Walkdown procedures for the tanks included the following:

- Verification that the overall tank configuration and anchorage details conform with the design drawings.
- Review of piping flexibility and other attachments to identify any potential sources of damage due to seismic anchor movement.
- Inspection of any unique features, which are not common to tanks, but are identified during a review of the drawings.
- Identification and inspection of potential sources for seismic interaction.

Pumps. Historical performance during past earthquakes of horizontal and vertical pumps has shown high seismic capacities. The walkdown procedures concentrated on:

- Verifying pump and motor anchorage including type of anchorage, foundation configuration and integrity.
- Reviewing potential nozzle loads and piping flexibility.
- Identifying interaction potential from attached or adjacent components.

Heat Exchangers. Walkdown procedures for heat exchangers concentrated on:

- Reviewing the supports including support saddles and anchorage details between the saddles and the concrete piers.
- Reviewing nozzle loads and piping flexibility.
- Identifying interaction potential from attached or adjacent components.

SEWS sheets were used to record configuration and dimensional data from the walkdown for heat exchanger support, anchorage, and attached or adjacent component interaction potential details that were not available from the plant data reviewed prior to the walkdown.

Diesel Generators. Past performance of diesel generators demonstrates their lower bound capacity levels higher than 0.5g pga. The walkdown procedures concentrated on:

- Reviewing anchorage and support integrity, noting if any vibration isolators were present.
- Reviewing the peripherals such as engine control panel, diesel day tank, fuel oil lines, air intake and exhaust ducting, and starting air receiver for positive anchorage.

SEWS sheets were used during the walkdown to record any problem areas encountered.

Electrical Distribution Equipment. Walkdown procedures for electrical distribution equipment included:

- o Reviewing and collecting anchorage details of the cabinet or enclosures for subsequent analytical review.

- o Verifying that the internal instruments and components are positively attached to the cabinet framing or enclosure walls and that the device mountings are not excessively flexible.

- o Identifying any system spatial interaction problems or flood or spray concerns.

Past performance of electrical distribution equipment during earthquakes suggests lower bound seismic capacities to exceed 0.5g pga, providing the equipment and internals, instruments, breakers, contactors, etc., are properly anchored.

HVAC Equipment. Two procedures for reviewing the HVAC equipment were used:

- o For HVAC equipment found mounted on vibration isolators, a detailed walkdown review was performed.

- o For HVAC equipment found positively anchored to a supporting structure, an engineering judgmental evaluation was performed and documented during the walkdown.

HVAC equipment positively anchored, as well as vibration isolator supported equipment with positive lateral restraints, have performed well during past earthquakes.

The review for HVAC equipment mounted on vibration isolators included recording the dimensional data and support configuration sufficient to perform an analytical evaluation after the walkdown.

The review of components in the second case included air intake and exhaust dampers, and exhaust fans. The walkdown review assessed anchorage and any seismic deficiencies present in order to judge that the component has a high seismic capacity. The predominant form of documentation for these components was the use of photographs to record the walkdown findings.

HVAC Ducting. The walkdown procedures consisted of two approaches:

- o Inspecting the ducting in close proximity to the HVAC equipment components reviewed. This includes:

    - o Vertical and lateral load resisting members of the ducting

    - o Any possible anchor point displacements that could impart significant loads to connected ducting

- o Inspecting samples of the ducting system selected during the walkdown.

Documentation consisted of noting any anomalies and taking several photographs.

Valves. Walkdown procedures consisted of a review of valves identified in the equipment list. Areas of concern reviewed during the walkdown included observing interaction potential between the valve operator and adjacent structure or component, evaluation of oversized or eccentric operators, and reviewing possible anchor point displacements between piping and valve. SEWS were used to document the walkdowns, and similar valves were reviewed by a less detailed walkby to verify similarity and to verify the absence of a seismic-interaction concern.

Piping. Past seismic PRA studies and earthquake experience data have shown that welded steel piping systems have a very high resistance to seismic loads.

Two piping failure modes that were addressed during the walkdown include:

- Impacting failures of valve operators
- Damage to piping caused by the failure of anchorage of attached equipment.

The valve clearance issue and the equipment anchorage issue were addressed in the evaluation of the specific equipment component and not as a part of the piping review.

The procedure for walking down the piping system included following the piping layout drawings to verify support locations, assessing system interaction potential to the piping, and noting detailed configuration information for piping details that were judged to be of potential concern. Particular attention was placed on evaluation of nonseismic piping, such as fire protection piping, and potential impacts on critical components.

Cable Trays. Inspection of the cable trays was performed with a general survey of cable tray systems in the plant. This general survey was performed to obtain an overview of cable tray construction throughout the plant. This included a review of the variety of cable tray system layouts, support configurations, and construction details. The inspection also considered items identified as being of potential concern, including failure of taut cables due to large relative displacement, severing of cables caused by sharp edges at the ends of cable trays, and weld failure.

Instrument Racks. Walkdown procedures of instrument racks consisted of:

- Reviewing and collecting anchorage details of instrument racks supporting instruments.
- Reviewing and verifying positive attachment of the instruments and components to the racks.
- Identifying seismic spatial interaction concerns to instrument tubing or air lines due to seismic failure of adjacent equipment.

5.2.3 Walkdown Documentation

Walkdown documentation for equipment and structures consisted of recording the findings using SEWS forms (Figures 5.2-1 and 5.2-2) and photographs (Figures 5.2-3 through 5.2-24). The SEWS forms were developed for each particular class of component indicating specific information required to confirm the high seismic capacity of the component in place as well as to record details sufficient to perform a seismic fragility evaluation if necessary. The SEWS forms reflect the varying levels of information required between different classes of equipment depending on their seismic ruggedness (e.g. pumps require little review other than to verify anchorage and interaction potential whereas HVAC components supported by vibration isolators require a detailed review, and thus a greater amount of information must be recorded

for a fragility evaluation).

Photographs were also used to record details of the equipment walkdown review. Photographs provide a permanent record of what was reviewed and support any notes or details taken during the walkdown. System interaction concerns were typically documented with photographs. Additionally, photographs were used in the fragility evaluation to confirm details taken in the walkdown or to provide additional clarification.

5.2.4 Walkdown Findings

5.2.4.1 Structures

Safety Related Structures: The following safety related structures were surveyed during the walkdown:

- Auxiliary Building
- Containment Building
- Control Building
- Diesel Generator Building
- Standby Service Water Cooling Tower and Storage Basins

All the Category I civil structures were constructed of reinforced concrete. Information required to develop structural fragilities is obtained primarily from the structural drawings. Thus, seismic walkdown review of building structures was limited to verification of building separations and identification of block walls which may pose a seismic interaction concern to safety related equipment components.

Block Walls: Masonry block walls that are located adjacent to safety related equipment that were reviewed were noted in two places: 1) at the 480V load control center (Figure 5.2-3) and 2) at the control room (Figure 5.2-4). These walls were designed to the plant design SSE level. For seismic events beyond the design SSE level, a fragility analysis is necessary to determine the median seismic capacity of these block walls.

Yard Tanks: Ground mounted storage yard tanks included in the walkdown are the condensate water storage tank (CST), the fire water storage tanks and the underground diesel generator fuel oil tanks. The CST is a large diameter vertical flat bottom storage tank anchored to a common reinforced concrete foundation mat with the refueling water storage tank. Both tanks are surrounded by dike walls. The CST is anchored to the concrete foundation with (14) anchor bolts of 1.5-inch diameter (Figure 5.2-5). Detailed information for the fragility evaluation was obtained from vendor drawings and foundation drawings.

The 300,000-gallon fire water tanks are unanchored vertical flat bottom storage tanks supported at grade. A pipe near the base of the tank was observed restrained by a U bolt (Figure 5.2-6). As such, the pipe may become damaged in a seismic event as it may behave as a restraint to the unanchored tank. The three diesel generator fuel oil storage tanks are buried 11 feet below grade outside of the diesel generator building. The fuel oil transfer pumps are located inside the tanks. The tanks were not reviewed during the walkdown due to inaccessibility.

5.2.4.2 Equipment

The Grand Gulf seismic walkdown consisted of a review of majority of the equipment components included in the study. The following provides brief summaries for various classes of equipment reviewed during the walkdown.

Cable Trays: Cable tray systems in the plant were sampled. Both trapeze and floor mounted supports were observed as shown in Figures 5.2-7 and 5.2-8. Diagonal bracing members were observed in both longitudinal directions of the cable trays. No seismic deficiency was noted.

Mechanical Equipment: Mechanical equipment including piping, valves, pumps, vessels and heat exchangers, were observed to be well anchored and appeared rugged in construction. Based on our experience with similar equipment in past earthquakes and in conducting other PRAs, we do not expect mechanical equipment in general to be dominant contributors to seismic risk.

Diesel Generators: One HPCS diesel generator and two Division 1 diesel generators were reviewed. All three engine generators are mounted on the diesel generator building floor slabs at the grade level. The HPCS engines and the generator are mounted on separate skids. However, individual skids were observed well anchored to the concrete floor slab such that no differential movement between the engine and the generator is expected.

The diesel oil day tanks were observed to be well constructed and are supported on braced frames constructed of heavy steel angle sections as shown in Figures 5.2-9 and 10. Each support leg is bolted down with a 1-1/2" diameter bolt. Vertical air receiver tanks were also observed to be well anchored in the same room where diesel generators and day tanks are located.

SSW Pump B: The Standby Service Water pump is a vertical long shaft pump. The motor is mounted on the top of the pump discharge head (Figure 5.2-11) which in turn is anchored to the SSW building concrete floor slab with four bolts as shown in Figure 5.2-12. Lateral supports of the pump casing and pump shaft were not reviewed during the walkdown due to inaccessibility.

RHR Pump: The RHR pumps as shown in Figure 5.2-13 for pump "C" are located in the auxiliary building at the elevation 93' slab (i.e. basemat). The driver (i.e. motor) was observed rigidly connected to the pump. The areas where the pumps are located were observed to be free of seismic interaction concerns.

RHR Heat Exchanger B: The RHR heat exchanger "B" is a vertical shell and tube type exchanger located in the north east corner of the auxiliary building. The exchanger is vertically supported on four structural steel columns as shown in Figure 5.2-14. Anchorage at the base of the support columns is shown in Figure 5.2-15. Laterally, the exchanger is restrained at two elevations. Due to high radiation in the room, detailed review of the heat exchanger and its supports was not performed.

Motor Control Centers (MCC) and Load Centers (LC): The MCCs for the standby service water pumps are located in the standby service water building at grade. These MCCs appeared to be well constructed and are welded to inverted channels which in turn are welded to steel embeds. MCC 16B21 was attached to steel channels with machine bolts. Anchorage of MCC 16B11 and LC 16B11 could not be verified during the walkdown.

None of the MCCs that were reviewed were opened for inspection during the walkdown because they were energized. Thus, mounting of the internal components and bolting of the adjacent cabinets were not inspected.

Electrical and Control Equipment: The following electrical and control equipment were reviewed during the walkdown

- o Motor Control Centers and Load Centers
- o 4kV Switchgear
- o 480V Load Control Center
- o Diesel Generator Control Panels
- o Control Instrumentation Panels

4kV Switchgear: The 4kV switchgear (9-foot tall by 5-foot deep) is located in the control building at elevation 119'. The switchgear enclosure was not opened because the unit was energized. As a result, anchorage of the switchgear was not verified during the walkdown.

480V Load Control Center: The ITE load control center (LC 16 BB2) consisting of a 480V switchgear and a transformer is located in the auxiliary building at elevation 139' (Figure 5.2-16). The load center was not opened for inspection during the walkdown since it was energized. As such, mounting of the internal components of the switchgear and the transformer, and bolting of the adjacent cabinets were not inspected. Masonry block walls were observed in the room where the load center is located (Figure 5.2-3).

Diesel Generator Control Panels: The control panel of the diesel generators is located in the diesel generator building at the elevation of the plant grade. The control panel is well anchored to the concrete floor slab with eight expansion bolts of 3/8" diameter at the front and back.

Control Instrumentation Panels: Various control panels and bench boards are located in the control room. A raised floor was observed in the control room. However, panels and bench boards were found anchored at the base to structural steel members as shown Figure 5.2-17. These steel members are in turn welded to steel embeds in the concrete floor slab (Figure 5.2-18). A masonry block wall was observed in the control room next to 1H13 panels (Figure 5.2-4). No other concern was noted during the walkdown for the control instrumentation panels.

Batteries: The 125V station battery in the control building (elevation 119') was reviewed for construction of the battery cells, flexibility of the cables, spacers between the cells and between the cells and rack, and construction and anchorage of the battery racks. A typical battery bank reviewed is shown in Figure 5.2-19. All the batteries and battery racks reviewed in the seismic walkdown were found to be well constructed and well anchored (Figure 5.2-20). The battery enclosures were constructed of Maronite type walls with a steel roof deck.

Offsite Power: A walkdown review was performed of the switchyard components. Equipment observed in the switchyard is similar to what has been observed by the team members in other east coast power stations. Two types of dead tank design circuit breakers (Figure 5.2-21) were observed in the switchyard. Because of the high voltage (500kV), longer than usual cantilevered ceramic insulators were observed on the circuit breakers (Figure 5.2-21), disconnect switches (Figure 5.2-22), and bus support structures (Figure 5.2-23). Damage to switchyard components in past earthquakes has been attributed primarily to failure of the ceramic insulators. Furthermore, anchorage of the steel support frames of circuit breakers to the concrete foundation mat was observed to be provided by friction clips as shown in Figure 5.2-24. Heavy equipment anchored by similar friction clips is known to have displaced in past earthquakes due to failure of the anchorage.

Seismic Spatial Interactions: A few potential seismic spatial interaction (SI) concerns were noted during the seismic walkdown. The SI concerns can be categorized into the following groups:

- o Potential impact of control room control panels or electrical cabinets by a masonry block wall.

- o Potential impact of 480V load control center by the adjacent masonry block wall.

5.3 Seismic Fragility Evaluation

5.3.1 Seismic Fragility Methodology

The seismic fragility of a structure or equipment is defined as the conditional probability of its failure at a given value of peak ground acceleration. The methodology for evaluating the seismic fragilities of structures and equipment is documented in (Ref. Ravindra and Kennedy (1983); PRA Procedures Guide (1983), and Kennedy and Ravindra (1984)). It has been developed and applied in over twenty-five seismic PRAs.

The objective of fragility evaluation is to estimate the ground acceleration capacity of a given component. This capacity is defined as the peak ground acceleration value at which the seismic response of a given component located at a specified point in the structure exceeds the component's resistance, resulting in its failure. The ground acceleration capacity of the component is estimated using information on plant design bases, responses calculated at the design-analysis stage, as-built dimensions, and material properties. The ground acceleration capacity is a random variable which can be described completely by its probability distribution. However, there is uncertainty in the estimation of the parameters of this distribution, the exact shape of this distribution, and in the appropriate failure model for the component. For any postulated failure model and set of parameter values and shape of the probability distribution, a fragility curve depicting the conditional probability of failure as a function of ground acceleration can be obtained. Hence, for different models and parameter assumptions, one could obtain different fragility curves. A satisfactory way to consider these uncertainties is to represent the component fragility by means of a family of fragility curves obtained as above; a subjective probability value is assigned to each curve to reflect the analyst's degree of belief in the model that yielded the particular fragility curve.

At any acceleration value, the component fragility (i.e., conditional probability of failure) varies from 0 to 1; this variation is represented by a subjective probability distribution. On this distribution we can find a fragility value (say, 0.01) that corresponds to the cumulative subjective probability of 5%. We have 5% cumulative subjective probability (confidence) that the fragility is less than 0.01. Similarly, we can find a fragility value for which we have a confidence of 95%. Note that these statements can be made without reference to any probability model. Using this procedure, the median (50%), high (95%), and low (5%) confidence fragility curves can be drawn. On the high confidence curve, we can locate the fragility value of 5%; the acceleration corresponding to this fragility on the high confidence curve is the so called HCLPF (High Confidence Low Probability of Failure) capacity of the component. By characterizing the component fragility through a family of fragility curves, the analyst has expressed all his knowledge about the seismic capacity of the component along with the uncertainties. Given the same information, two analysts with similar experience and expertise would produce approximately the same fragility curves. Development of the family of fragility curves using different failure models and parameters for a large number of components in a seismic PRA is impractical if it is done as described above. Hence, a simple

model for the fragility was proposed as described in the above cited references. In the following this fragility model is described.

5.3.1.1 Fragility Model

The entire fragility family for an element corresponding to a particular failure mode can be expressed in terms of the best estimate of the median ground acceleration capacity, A_m, and two random variables. Thus, the ground acceleration capacity, A, is given by

$$A = A_m e_R e_U \qquad \text{(Eq. 5.3-1)}$$

in which e_R and e_U are random variables with unit medians, representing, respectively, the inherent randomness about the median and the uncertainty in the median value. In this model, we assume that both e_R and e_U are lognormally distributed with logarithmic standard deviations, β_R and β_U, respectively. The formulation for fragility given by Eq. (5.3-1) and the assumption of lognormal distribution allow easy development of the family of fragility curves which appropriately represents fragility uncertainty. For the quantification of fault trees in the plant system and accident sequence analyses, the uncertainty in fragility needs to be expressed in a range of conditional failure probabilities for a given ground acceleration. This is achieved as explained below:

With perfect knowledge (i.e., only accounting for the random variability, β_R), the conditional probability of failure, f_o, for a given peak ground acceleration level, a, is given by

$$f_o = \Phi [\ \ln(a/A_m) / \beta_R\ ] \qquad \text{(Eq. 5.3-2)}$$

where $\Phi(\,)$ is the standard Gaussian cumulative distribution function. The relationship between f_o and a is the median fragility curve plotted in Figure 5.3-1 for a component with a median ground acceleration capacity $A_m = 0.87g$ and $\beta_R = 0.25$. For the median conditional probability of failure range of 5% to 95%, the ground acceleration capacity would range from 0.58g to 1.31g.

When the modeling uncertainty β_U is included, the fragility becomes a random variable (uncertain). At each acceleration value, the fragility f can be represented by a subjective probability density function. The subjective probability, Q (also known as "confidence") of not exceeding a fragility f' is related to f' by

$$f' \quad = \quad \Phi\ [(\ln(a/A_m + \beta_U\ \Phi^{-1}\ (Q))/\beta_R] \qquad \text{(Eq. 5.3-3)}$$

where

Q = $P[f < f'|\,a]$ i.e., the subjective probability (confidence) that the conditional probability of failure, f, is less than f' for a peak ground acceleration a

$\Phi^{-1}(.)$ = the inverse of the standard Gaussian cumulative distribution function.

For example, the conditional probability of failure f' at acceleration 0.6g that has a 95% nonexceedance subjective probability (confidence) is obtained from Eq. (5.3-3) as 0.79. The 5% to 95% probability (confidence) interval on the failure at 0.6g is 0 to 0.79 with a median value of 0.068 and mean of 0.20. Subsequent computations are made easier by discretizing the random variable probability of failure f into different intervals and deriving a probability q_i

for each interval (Figure 2-2). Note that the sum of the q_i probabilities associated with all the intervals is unity. The process develops a family of fragility curves, each with an associated probability q_i.

The median ground acceleration capacity A_m and its variability estimates β_R and β_U are evaluated by taking into account the safety margins inherent in capacity predictions, response analysis, and equipment qualification, as explained below.

5.3.1.2 Failure Modes

The first step in generating fragility curves such as those in Figure 5.3-1 is to develop a clear definition of what constitutes failure for each of the critical elements in the plant. This definition of failure must be agreeable to both the structural analyst generating the fragility curves and the systems analyst who must judge the consequences of component failure. Several modes of failure (each with a different consequence) may have to be considered and fragility curves may have to be generated for each of these modes. The following definitions of failure are assumed for structures and equipment.

Structures: For elements of structures which support safety related equipment, failure is assumed to occur when inelastic deformations due to seismic motions are large enough to potentially affect the operability of equipment or when a concrete wall is cracked sufficiently so that equipment attachments fail. This is a conservative definition of failure of a structure, and is at a lower acceleration level than the acceleration level for total collapse of a building. Considerable margin exists for structural collapse compared to the capacities calculated for failure related to equipment (functional and structural failure modes). Also, a structural failure has been generally assumed to result in a common cause failure of multiple safety systems, housed in the same structure. Structures which are susceptible to sliding are considered to have failed when sufficient sliding deformation has occurred to fail buried or interconnecting piping or electrical duct banks.

Equipment: Safety related equipment is assumed to fail when it can no longer perform its function. Failure can be caused by either direct failure (i.e. structural failure) or functional failure due to inertial loads or relative displacement-induced loading, or indirect failure caused by failure of an adjacent structure or component which can fall onto and fail the safety related equipment. Structural failure includes bending, buckling of supports, anchor bolt pullout, etc. Functional failures include binding of valves, excessive deflection, and relay trip or chatter.

It may be possible to identify the failure mode most likely to be caused by the seismic event by reviewing the equipment design and considering only that mode. Otherwise, fragility curves are developed based on the premise that the component could fail in any one of many potential failure modes. Identification of the credible modes of failure is largely based on the analyst's experience and judgment. Review of plant design criteria, calculated stress levels in relation to the allowable limits, qualification test results, seismic fragility evaluation studies done on other plants, and reported failures (in past earthquakes, in licensee event reports and fragility tests) are useful in this task.

Consideration should also be given to the potential for soil failure modes (e.g., liquefaction, toe bearing pressure failure, base slab uplift, and slope failures). For buried equipment (i.e., piping and tanks), failure due to lateral soil pressures may be an important mode. Seismically induced failures of structures or equipment under impact of another structure or equipment (e.g., a crane) may also be a consideration.

5.3.1.3 Estimation of Fragility Parameters

In estimating fragility parameters, it is convenient to work in terms of an intermediate random variable called the factor of safety. The factor of safety, F, on ground acceleration capacity above the safe shutdown earthquake level specified for design, A_{SSE}, is defined as follows:

$$A = F\,A_{SSE}$$

$$F = \frac{\text{Actual seismic capacity of element}}{\text{Actual response due to SSE}}$$

$$= \frac{\text{Actual seismic capacity of element}}{\text{Calculated capacity}} \times \frac{\text{Calculated capacity}}{\text{Design response due to SSE}} \times \frac{\text{Design response due to SSE}}{\text{Actual response due to SSE}}$$

F is further simplified as:

$$F = \frac{\text{Actual seismic capacity of element}}{\text{Design response due to SSE}} \times \frac{\text{Design response due to SSE}}{\text{Actual response due to SSE}}$$

$$F = F_C F_{RS} \qquad \text{(Eq. 5.3-4)}$$

The median factor of safety, F_m, can be directly related to the median ground acceleration capacity, A_m, as:

$$F_m = A_m / A_{SSE} \qquad \text{(Eq. 5.3-5)}$$

The logarithmic standard deviations of F, representing inherent randomness and uncertainty, are identical to those for the ground acceleration capacity A.

5.3.1.4 Structural Fragility

For structures, the factor of safety can be modeled as the product of three random variables:

$$F = F_S\,F_\mu\,F_{RS} \qquad \text{(Eq. 5.3-6)}$$

The strength factor, F_S, represents the ratio of ultimate strength (or strength at loss-of-function) to the stress calculated for A_{SSE}. In calculating the value of F_S, the nonseismic portion of the total load acting on the structure is subtracted from the strength as follows:

$$F_S = (S - P_N) / (P_T - P_N \qquad \text{(Eq. 5.3-7)}$$

where S is the strength of the structural element for the specific failure mode, P_N is the normal operating load (i.e., dead load, operating temperature load, etc.) and P_T is the total load on the structure (i.e., sum of the seismic load for A_{SSE} and the normal operating load). For higher

earthquake levels, other transients (e.g., SRV discharge, and turbine trip) may have a high probability of occurring simultaneously with the earthquake; the definition of P_N in such cases should be extended to include the loads from these transients.

The inelastic energy absorption factor (ductility), F_μ, accounts for the fact that an earthquake represents a limited energy source and many structures or equipment items are capable of absorbing substantial amounts of energy beyond yield without loss-of-function. A suggested method to determine the deamplification effect resulting from inelastic energy dissipation involves the use of ductility modified response spectra (Ref. Newmark, 1977). The deamplification factor is primarily a function of the ductility ratio μ defined as the ratio of maximum displacement to displacement at yield. More recent analyses (Ref. Riddell and Newmark, 1979) have shown the deamplification factor to be a function of system damping. One might estimate a median value of μ for low-rise concrete shear walls (typical of auxiliary building walls) of 4.0. The corresponding median F_μ value would be 2.4. The variabilities in the inelastic energy absorption factor, F_μ, are estimated as $\beta_R = 0.21$ and $\beta_U = 0.21$, taking into account the uncertainty in the predicted relationship between F_μ, μ, and system damping.

The structure response factor, F_{RS}, recognizes that in the design analyses structural response was computed using specific (often conservative) deterministic response parameters for the structure. Because many of these parameters are random (often with wide variability) the actual response may differ substantially from the design-calculated response for a given peak ground acceleration.

The structure response factor, F_{RS}, is modeled as a product of factors influencing the response variability:

$$F_{RS} = F_{SA}\, F_\phi\, F_\delta\, F_M\, F_{MC}\, F_{EC}\, F_{SD}\, F_{SS}, \qquad \text{(Eq. 5.3-8)}$$

where

F_{SA} = spectral shape factor representing variability in ground motion and associated ground response spectra

F_ϕ = direction factor representing the variability in the two earthquake direction response spectral values about the mean value

F_δ = damping factor representing variability in response due to difference between actual damping and design damping

F_M = modeling factor accounting for uncertainty in response due to modeling assumptions

F_{MC} = mode combination factor accounting for variability in response due to the method used in combining dynamic modes of response

F_{EC} = earthquake component combination factor accounting for variability in response due to the method used in combining earthquake components

F_{SD} = factor to reflect the reduction with depth of seismic input

F_{SS} = factor to account for the effect of soil-structure interaction

The median and logarithmic standard deviations of F are expressed as:

$$F_m = F_{Sm} F_\mu F_{SAm} F_{\delta m} F_{Mm} F_{MCm} F_{ECm} F_{SDm} F_{SSm} \quad \text{(Eq. 5.3-9)}$$

and

$$\beta_F^2 = (\beta_S^2 + \beta_U^2 + \beta_{SA}^2 + \cdots + \beta_{SS}^2)^{1/2} \quad \text{(Eq. 5.3-10)}$$

The logarithmic standard deviation β_F is further divided into random variability, β_R, and uncertainty, β_U. To obtain the median ground acceleration capacity A_m the median factor of safety, F_m, is multiplied by the safe shutdown earthquake peak ground acceleration.

5.3.1.5 Equipment Fragility

For equipment and other components, the factor of safety is composed of a capacity factor, F_C, a structure response factor, F_{RS}, and an equipment response (relative to the structure) factor, F_{RE}. Thus,

$$F = F_C F_{RE} F_{RS} \quad \text{(Eq. 5.3-11)}$$

The capacity factor F_C for the equipment is the ratio of the acceleration level at which the equipment ceases to perform its intended function to the seismic design level. This acceleration level could correspond to a breaker tripping in a switchgear, excessive deflection of the control rod drive tubes, or failure of a steam generator support. The capacity factor for the equipment may be calculated as the product of F_S and F_μ. The strength factor, F_S, is calculated using Eq. (5.3-7). The strength, S, of equipment is a function of the failure mode. Equipment failures can be classified into three categories:

1. Elastic functional failures
2. Brittle failures
3. Ductile failures.

Elastic functional failures involve the loss of intended function while the component is stressed below its yield point. Examples of this type of failure include the following:

- Elastic buckling in tank walls and component supports
- Excessive blade deflection in fans
- Shaft seizure in pumps.

The strength of the component is considered to be the load level at which functional failure occurs.

Brittle failure modes are those which have little or no system inelastic energy absorption capability. Examples include the following:

- Anchor bolt failures
- Component support weld failures
- Shear pin failures.

Each of these failure modes has the ability to absorb some inelastic energy on the component level, but the plastic zone is very localized and the system ductility for an anchor bolt or a

support weld is very small. The strength of the component failing in a brittle mode is therefore calculated using the ultimate strength of the material.

Ductile failure modes are those in which the structural system can absorb a significant amount of energy through inelastic deformation. Examples include the following:

- o Pressure boundary failure of piping
- o Structural failure of cable trays and ducting
- o Polar crane failure.

The strength of the component failing in a ductile mode is calculated using the yield strength of the material for tensile loading. For flexural loading, the strength is defined as the limit load or load to develop a plastic hinge.

The inelastic energy absorption factor, F_μ, for a piece of equipment is a function of the ductility ratio, μ. The median value F_μ is considered close to 1.0 for brittle and functional failure modes. For ductile failure modes of equipment that respond in the amplified acceleration region of the design spectrum (i.e., 2 to 8 Hz):

$$F_\mu = e\,(2\mu - 1)^{1/2} \qquad \text{(Eq. 5.3-12)}$$

where e is a random variable reflecting the error in Eq. (5.3-12) and has a median value of 1.0 and a logarithmic standard deviation, β_U, ranging from 0.02 to 0.10 (increasing with the ductility ratio). For rigid equipment, F_μ is given by

$$F_\mu = e\,\mu^{0.13} \qquad \text{(Eq. 5.3-13)}$$

Again, e is a random variable of median equal to 1.0 and logarithmic standard deviation ranging from 0.02 to 0.10.

The median and logarithmic standard deviation of ductility ratios for different equipment are calculated considering recommendations of (Newmark, 1977). This reference gives a range of ductility ratios to be used for design. The upper end of this range might be considered to represent approximately the median value, while the lower end of the range might be estimated at about two logarithmic standard deviations below the median.

The equipment response factor F_{RE}, is the ratio of equipment response calculated in the design to the realistic equipment response; both responses are calculated for design floor spectra. F_{RE} is the factor of safety inherent in the computation of equipment response. It depends upon the response characteristics of the equipment and is influenced by some of the variables listed under Eq. (5.3-8). These variables differ according to the seismic qualification procedure. For equipment qualified by dynamic analysis, the important variables that influence response and variability are as follows:

- o Qualification method (QM)
- o Spectral shape (SA) - including the effects of peak broadening and smoothing, and artificial time history generation
- o Modeling (affects mode shape and frequency results) (M)
- o Damping (δ)

o Combination of modal responses (for response spectrum method) (MC)

o Combination of earthquake components (EC).

For rigid equipment qualified by static analysis, all variables, except the qualification method, are not significant. The equipment response factor is the ratio of the specified static coefficient divided by the zero period acceleration of the floor level where the equipment is mounted. If the equipment is flexible and was designed via the static coefficient method, the dynamic characteristics of the equipment must be considered. This requires estimating the fundamental frequency and damping, if the equipment responds predominantly in one mode. The equipment response factor is the ratio of the static coefficient to the spectral acceleration at the equipment fundamental frequency.

Where testing is conducted for seismic qualification, the response factor must take into account the following:

o Qualification method (QM)

o Spectral shape (SA)

o Boundary conditions in the test versus installation (BC)

o Damping (δ)

o Spectral test method (sine beat, sine sweep, complex waveform, etc.) (STM)

o Multi-directional effects (MDE).

The overall equipment response factor is the product of these factors of safety corresponding to each of the variables identified above. The median and logarithmic standard deviations for randomness and uncertainty are estimated following Eqs. (5.3-9) and (5.3-10).

The structural response factor, F_{RS}, is based on the response characteristics of the structure at the location of component (equipment) support. The variables pertinent to the structural response analyses used to generate floor spectra for equipment design are the only variables of interest to equipment fragility. Time-history analyses using the same structural models used to conduct structural response analysis for structural design are typically used to generate floor spectra. The applicable variables are as follows:

o Spectral shape
o Damping
o Modeling
o Soil-structure interaction.

For equipment with a seismic capacity level that has been reached while the structure is still within the elastic range, the structural response factors should be calculated using damping values corresponding to less than yield conditions (e.g., about 5% median damping for reinforced concrete). The combination of earthquake components is not included in the structural response since the variable is to be addressed for specific equipment orientation in the treatment of equipment response.

Median F_m and variability β_R and β_U estimates are made for each of the parameters affecting

capacity and response factors of safety. These median and variability estimates are then combined using the properties of lognormal distributions in accordance with Eqs. (5.3-6), (5.3-8), and (5.3-11) to obtain the overall median factor of safety F_m and variability β_R and β_U estimates required to define the fragility curves for the structure or equipment. For each variable affecting the factor of safety, the random (β_R) and uncertainty (β_U) variabilities must be separately estimated. The differentiation is somewhat judgmental, but it can be based on general guidelines. Essentially, β_R represents variability due to the randomness of the earthquake characteristics for the same acceleration and to the structural response parameters which relate to these characteristics. The dispersion represented by β_U is due to factors such as the following:

- o Our lack of understanding of structural material properties such as strength, inelastic energy absorption, and damping.
- o Errors in calculated response due to use of approximate modeling of the structure and inaccuracies in mass and stiffness representations.
- o Usage of engineering judgment in lieu of complete plant-specific data on fragility levels of equipment capacities, and responses.

5.3.2 Grand Gulf Seismic Fragilities

The seismic fragilities of equipment included in the Grand Gulf shutdown PRA are presented in Table 5.3-1. Selection of these equipment items was discussed in Section 5.2. Detailed discussions of the fragility evaluation are provided in this section. For structures such as concrete shear walls, prestressed concrete containment, steel frames, masonry walls, field-erected tanks, and buried structures, the fragility parameters are generally estimated using plant-specific information. Fragilities of Grand Gulf Category I civil structures were not calculated in this study. All the Category I civil structures are constructed of reinforced concrete. Based on the cursory walkdown review and a review of design documents, it was judged that Grand Gulf Category I structures generally would not be significant seismic risk contributors.

For major passive equipment (e.g., reactor pressure vessel, steam generator, reactor coolant pump, recirculation pump, major vessels, heat exchangers, and major piping), it is preferable to develop plant-specific fragilities using original design analyses. Because of the large quantities of other types of passive equipment (e.g., piping and supports, cable trays and supports, HVAC ducting and supports, conduit, and miscellaneous vessels and heat exchangers), it is generally necessary to use generic fragilities. For active equipment, use of a combination of generic and plant-specific information is needed to develop fragilities.

5.3.2.1 Screening of Equipment

Grand Gulf equipment that is inherently seismically rugged was screened from detailed fragility evaluation. Such equipment included all the horizontal motor driven pumps, small shell and tube type heat exchangers, and emergency diesel generators (including peripherals). Screening of equipment was performed following the procedures in EPRI NP-6041-M (EPRI, 1988).

5.3.2.2 Generic Equipment Seismic Fragilities

Some equipment items in this study were assigned generic seismic fragilities for the following reasons:

- Some equipment was not reviewed during the walkdown due to inaccessibility

- Some anchorages could not be verified during the walkdown since the equipment was energized and could not be opened. This included some motor control centers, load centers, and the 480V transformer.

- Some items lacked design information such as as-built drawings for performing a fragility evaluation.

For these equipment items, conservatively estimated seismic fragility values which were developed either from limited design information or from a review of UCID-20571 were assigned as indicated in Table 5.3-1.

5.3.2.4 Grand Gulf Specific Equipment Fragilities

For the remaining Grand Gulf shutdown components, seismic fragilities were calculated following the methodology discussed in Section 5.2 using information such as walkdown data, design documents, and seismic qualification packages.

5.3.2.4.1 Basis for Factors of Safety

There was a general lack of detailed information available for this study on the seismic fragility of specific Grand Gulf equipment. This condition occurs because existing codes and standards do not require determination of ultimate seismic capacities, either for structures or equipment qualified by analysis, or for equipment or components qualified by testing. Therefore, most median safety factors, estimates of variability, and conditional probabilities of failure estimated in this study are based on existing analyses, qualified engineering judgment and assumptions. Limited additional analyses were conducted to evaluate the expected failure capacities of the important equipment. The additional analyses were based for the most part on the original design or qualification analyses which were available.

For most of the safety equipment, information on analysis methods was available in the design analyses and in summary form in the updated FSAR. Seismic response information for the selected sample of safety-related equipment evaluated in this study was obtained from available vendor seismic qualification reports or design calculations for specific components. In some cases such as for piping, only the seismic analysis requirements and stress acceptance criteria were known. Safety factors for response and structural or functional capacity were estimated from existing information when available.

In-structure response spectra for all Category I structures were generated during the design process. From these typical floor response spectra and knowledge of estimates of equipment fundamental frequencies, an estimate was made of the median peak equipment response. The median peak equipment response estimate was then compared to the dynamic response or equivalent static coefficient used in design to determine a median safety factor on response.

Capacity factors are derived from several sources of information: plant-specific design reports, test reports, generic earthquake experience data and generic analytical derivations of capacity based on governing codes and standards. Two failure modes were considered in developing capacity factors for piping and equipment: structural and functional. Equipment and piping design reports delineate stress levels for the specified seismic loading plus normal operating conditions. Where the equipment fails in a structural mode (i.e., pressure boundary rupture or loss of support), the median capacity factor and its variability were derived considering strength and energy absorption capability (ductility). In cases where equipment must function, the capacity factor was derived by comparing the equipment functional failure (or fragility)

level to the design level of seismic loading. Some fragility test data are available on generic classes of equipment that have been utilized in hardened military installations. Such equipment was off-the-shelf without special shock-resistant design but is similar, in many cases, to nuclear power plant equipment. Other equipment fragility data used in this study are from Holman (1986), Bandyopadhyay et al., (1986 and 1987) and EPRI (May 1987). These data provide estimates of the fragility levels, and thus, safety factors can be developed for the specified design earthquake. Fragility levels are not normally determinable from equipment qualification reports, but the achieved test levels can be utilized to update generic fragilities derived from the military or experience data.

5.3.2.4.2 Grand Gulf Seismic Design Criteria

The Grand Gulf Nuclear Station was designed in the mid-1970s in accordance with criteria and codes in effect at that time. The Grand Gulf systems and components that are essential to the prevention or mitigation of consequences of accidents which could affect the public health and safety were designed to enable the facility to withstand the effects of natural forces including earthquakes. The design criteria included the effects of simultaneous earthquake and loss-of-coolant-accident (LOCA) conditions. The plant was designed to withstand both a Safe Shutdown Earthquake (SSE) and an Operating Basis Earthquake (OBE). The structural design criteria for the SSE were based on 0.15g and the OBE on 0.075g peak horizontal ground accelerations for all Seismic Category I structures. The maximum vertical SSE ground acceleration for Grand Gulf was specified to be two-thirds of the horizontal acceleration. A comparison of the Grand Gulf design ground response spectrum with the EPRI median 10^{-4} uniform hazard spectrum (UHS) anchored to 0.15g SSE level is presented in Figure 5.3-2.

5.3.2.4.3 Differences Between Criteria Used for Design of Grand Gulf and Parameters Used in the Evaluation of the Seismic Capacity

The seismic design of the Grand Gulf structures and equipment was based for the most part on currently accepted methodology and criteria in conformance with USNRC licensing requirements. These criteria and methods together with the design codes in use at the time of the design form a conservative design basis and ensure that substantial factors of safety are introduced at various stages in the design procedure. The exact magnitude of many of these safety factors is a matter of considerable discussion. Nevertheless, in order to establish a realistic value of the actual seismic capacity of a structure or equipment, the amount of conservatism along with its variability must be established as accurately as possible.

The general approach used in the evaluation of the Grand Gulf seismic capacities is to develop the overall factor of safety associated with each important potential failure mode. Based on the governing design parameters, a median seismic capacity is then obtained in terms of the free field ground acceleration. The overall factor of safety is typically composed of several important contributions such as strength, inelastic energy absorption (ductility), and differences in median response compared to design values resulting from such parameters as earthquake characteristics, damping, and directional load components. The following section provides the fragility evaluation of an equipment item that was qualified by testing.

5.3.3 Example of Equipment Fragility Evaluation

In this section, we describe the seismic fragility derivation of the station battery located at the Elevation 111 feet of the control building. The station battery was seismically qualified by shake table test. The test was performed in accordance with IEEE 344-1975 requirements with multi-frequency, biaxial input. The following presents the derivation of the station battery seismic fragility based on shake table test results.

Equipment Capacity Factor

Strength Factor: For an equipment item qualified by shake table testing, the equipment is subjected to shake table motions such that the response spectrum of the input motion envelopes the required floor response spectrum. The tests are generally not carried to a failure level. Thus, when no structural distress is observed in the test, the actual acceleration capacity is somewhat above the qualification acceleration level.

There was no structural distress noted in the qualification of the station battery. Thus, appropriate increase factors are required to estimate the margin between the median acceleration capacity for structural failure and the qualification acceleration level. For lack of measurements of stresses in the test, the yield level of the critical element is estimated to be a factor of 1.2 above the qualification level. A second factor which accounts for the difference between the median failure level and the yield level is estimated to be 1.5. Thus, the median equipment strength factor is

$$F_{Sm} = (1.2)\,(1.5)\,(S_{aTRS}/S_{aRRS})$$

in which S_{aTRS} is the test spectral acceleration and S_{aRRS} is the required floor response spectral acceleration at the fundamental frequency of the equipment. The seismic qualification report documented that the test response spectrum is about three times that of the required response spectrum over the entire frequency range. Thus, the (S_{aTRS}/S_{aRRS}) ratio of 3.0 is assumed. Therefore, the median equipment strength factor was estimated to be 5.4.

The variabilities in the strength factor are derived from the two increase factors, i.e. 1.2 and 1.5. Based on that, the randomness and uncertainty variabilities in the strength factor were estimated to be:

$$\beta_R = 0.05$$
$$\beta_U = 0.11$$

Since the variability in the yield level is influenced somewhat by the characteristics of the seismic input, a β_R of 0.05 was estimated.

Inelastic Energy Absorption Factor: Because the generic equipment strength factor could correspond to a non-ductile anchorage failure, no additional credit was taken for ductility to increase the acceleration capacity. Thus,

$$F_{\mu m} = 1.0$$
$$\beta_R = 0$$
$$\beta_U = 0$$

Thus, the overall equipment capacity factor is

$$F_{Cm} = 5.4$$
$$\beta_R = 0.05$$
$$\beta_U = 0.11$$

Equipment Response Factors

Qualification Method Factor: Because the station battery was qualified by random, multi-frequency shake table testing, the qualification method is taken to be median centered. Thus,

F_{QMm} = 1.0

β_R = 0

β_U = 0

Damping Factor: The qualification was based on test levels in which the test damping values are median centered by definition. Moreover, in evaluating the equipment strength factor, the test spectral acceleration was compared to the required floor response spectrum at the same damping. Thus, the median damping factor is unity and the randomness and uncertainty variabilities are both zero.

$F_{\delta m}$ = 1.0

β_R = 0

β_U = 0

Boundary Condition Factor: The variabilities associated with the boundary condition factor reflect the variation in the equipment response that is attributed to differences in the mounting or anchorage conditions between the test fixture and the as-built plant installation. Changes in the boundary conditions can include changes in mounting stiffness and mounting strength. Variability can be estimated by estimating the equipment overtest factors (i.e. ratio of test spectral acceleration to the required spectral acceleration) at the median and plus or minus $1.65^*\beta_f$ frequencies. Because the test response spectrum is three times the required response spectrum over the entire frequency range, the variability in the overtest factor due to frequency shift will not be significant. Thus, the median boundary condition factor and its associated variability (entirely uncertainty) were estimated to be:

F_M = 1.0

β_R = 0

β_U = 0.10

Spectral Test Method Factor: This factor characterizes the ability of the test method to excite all important modes of the equipment sufficiently. The random, multi-frequency, biaxial tests were utilized for qualifying the station battery. Therefore, the test method was judged to be capable of exciting the important modes of the station battery. Thus, the median spectral test method factor is unity and the associated variabilities are zero.

Earthquake Component Combination Factor: The earthquake component combination factor attempts to quantify the disparity between the test conditions and the actual earthquake response. Biaxial tests tend to be unconservative because they do not represent three directional input of an actual earthquake. However, given the configurations of the station battery (i.e rigid in the side to side direction), the front to back direction response governs the failure mode. The biaxial test was then judged to be median centered and the median factor of safety is unity. The associated variability is entirely randomness and was estimated to be 0.10.

Thus, the overall equipment response factor is:

F_{ER} = 1.0

β_R = 0.10

β_U = 0

Structural Response Factor: Variabilities of the following structural response factors were estimated by using the median and 84th-percentile floor response spectra:

- o Spectral shape factor
- o Structural damping factor
- o Structural modeling factor
- o Soil-structure interaction

Spectral Shape Factor: At the fundamental frequency of the control building (i.e. about 4 Hz), the 5% damped design spectral acceleration and the median spectral acceleration, from the design spectrum and the Uniform Hazard Spectrum for recurrence of 0.0001/year, are 0.40g and 0.15g, respectively. Thus,

F_{SS} = 0.40/0.15 = 2.7

β_R = 0.20

The β_R variability is estimated to account for randomness of the earthquake ground motion, or variability in the shape of the Uniform Hazard Spectrum. There is no uncertainty variability associated with this factor.

Structural Damping Factor: Given the high soil radiation damping used in the Grand Gulf design soil-structure interaction analysis, the effect of structural material damping on the response is not significant such that the median damping factor is unity, and the associated uncertainty variability was estimated to be 0.05.

Modeling Factor: Assuming that the median fundamental frequency of the control building is 4 Hz and a β_f of 0.3, the lower bound and upper bound frequencies ($\pm 1 * \beta_f$ from the median) are 3 Hz and 5.4 Hz, respectively. The corresponding spectral accelerations from the UHS are 0.15g and 0.15g. Thus, the modeling uncertainty associated with estimates of structural frequencies was estimated to be zero. The total modeling uncertainty including that associated with the mode shape is 0.15. Since the station battery is located at Elevation 111 feet which is close to the basemat, a smaller modeling uncertainty of 0.10 was estimated.

Soil-Structure Interaction: Two types of soil-structure interaction effects are usually considered in the analysis of nuclear power stations at deep soil sites. The first effect is the variation of peak acceleration and frequency content with the depth. The second effect involves the variation in frequency and response of the structure due to flexibility of the soil media and dissipation of energy into this soil by radiation. The design basis soil-structure interaction analysis for Grand Gulf structures considered only the second effect. Due to lack of detailed soil-structure interaction (SSI) analysis for this shutdown-PRA study, the conservatism in the design SSI analysis from not considering the first effect was neglected. Thus, the factor of safety of SSI was assumed to be unity. The associated variabilities were estimated to be 0.05 and 0.15, respectively, for randomness and uncertainty. A relatively low uncertainty variability was assumed since the design analysis conservatively set an upper

bound limit of 10% damping for the composite modal damping.

Thus, the overall structural response factor and associated variabilities are:

$$F_{SR} = 1.0$$

$$\beta_R = 0.21$$

$$\beta_U = 0.19$$

Thus, the median peak ground acceleration capacity of the station battery was determined to be:

$$A_m = (5.4)\,(2.7)\,(1.0)\,(0.15g)$$

$$= 2.2g$$

The variabilities associated with this median PGA capacity were determined to be 0.23 and 0.24, respectively for β_R and β_U, by combining variabilities of equipment capacity factor, equipment response factor and structural response factors (using Equation 5.3-10). The HCLPF capacity of the station battery is 1.0g.

TABLE 5.3-1

GRAND GULF LOW POWER PRA
COMPONENT SEISMIC FRAGILITIES

COMPONENT DESCRIPTION	A_m (g)	β_R	β_U	HCLPF (g)	COMMENTS
SSW Pump	0.82	0.25	0.30	0.33	Grand Gulf specific fragility.
SSW MCC	1.60	0.25	0.30	0.65	Grand Gulf specific fragility.
RHR Pump	--	--	--	0.30	Screened per EPRI NP-6041.
RHR Heat Exchanger	1.65	0.25	0.30	0.67	Grand Gulf specific fragility.
Standby Diesel Generators	--	--	--	0.30	Screened per EPRI NP-6041.
HPCS Diesel Generator	--	--	--	0.30	Screened per EPRI NP-6041.
Fuel Oil Day Tanks	--	--	--	0.30	Screened per EPRI NP-6041.
Diesel Fuel Oil Tanks	--	--	--	0.30	Screened per EPRI NP-6041.
D/G Control Panels	1.60	0.23	0.24	0.75	Grand Gulf specific fragility.
Station Batteries	2.20	0.23	0.24	1.00	Grand Gulf specific fragility.
D/G Air Receiver Tanks	>2g	--	--	--	High median capacity based on equipment capacity factor.

TABLE 5.3-1 (CONTINUED)

GRAND GULF LOW POWER PRA
COMPONENT SEISMIC FRAGILITIES

COMPONENT DESCRIPTION	A_m (g)	β_R	β_U	HCLPF (g)	COMMENTS
480V Load Control Center	1.87	0.25	0.30	0.75	Grand Gulf specific fragility; adjacent masonry wall has vertical rebars as well as horizontal Dur-O-Wall reinforcing steel such that it does not pose a collapse hazard to the load center.
4kV Switchgear	2.43	0.25	0.30	1.00	Grand Gulf specific fragility.
Condensate Storage Tanks	0.59	0.21	0.18	0.31	Grand Gulf specific fragility.
Offsite power	0.30	0.25	0.35	0.11	Generic values.

Revision 2, Corrected, 6/28/91
Status Y N U

SCREENING EVALUATION WORK SHEET (SEWS) Sheet 1 of 3

Equip. ID No. MCC 16B21 Equip. Class 1 - Motor Control Centers

Equipment Description (1B B:21)

Location: Bldg. Aux Floor El. 139' Room, Row/Col 1A 313

Manufacturer, Model, Etc. (optional)

SEISMIC CAPACITY VS DEMAND

1	Elevation where equipment receives seismic input		139'
2.	Elevation of seismic input below about 40' from grade		Y N U
3.	Equipment has fundamental frequency above about 8 Hz		Y N U N/A
4.	Capacity based on:	Existing Documentation	DOC
		Bounding Spectrum	BS
		GERS	GERS
5.	Demand based on:	Ground Response Spectrum	GRS
		1.5 x Bounding Spectrum	ABS
		Conserv. Des. In-Str. Resp. Spec.	CRS
		Realistic M-Ctr. In-Str. Resp. Spec.	RRS

Does capacity exceed demand? ~~Y N U~~ N/A

CAVEATS - BOUNDING SPECTRUM (Identify with an asterisk (*) those caveats which are met by intent without meeting the specific wording of the caveat rule and explain the reason for this conclusion in the COMMENTS section below)

1.	Equipment is included in earthquake experience equipment class	Y N U N/A
2.	600 V rating or less	Y N U N/A
3.	Adjacent cabinets which are close enough to impact, or sections of multi-bay cabinets, are bolted together if they contain essential relays	Y N U N/A
4.	Attached weight (except conduit) less than about 100 lbs per cabinet assembly	Y N U N/A
5.	Externally attached items rigidly anchored	Y N U N/A
6.	General configuration similar to NEMA Standards	Y N U N/A
7.	Cutouts in lower half less than 6 in. wide and 12 in. high	Y N U N/A
8.	All doors secured by latch or fastener	Y N U N/A
9.	Natural frequency relative to 8 Hz limit considered	Y N U N/A
10.	Anchorage adequate (See checklist below for details)	Y N U N/A
11.	Relays mounted on equipment evaluated	Y N U N/A
12.	Have you looked for and found no other adverse concerns?	Y N U N/A

Is the intent of all the caveats met for Bounding Spectrum? Y N U N/A

Figure 5.2-1: SEWS form of MCC 16B21

SCREENING EVALUATION WORK SHEET (SEWS)

Equip. ID No. MCC 16B21 Equip. Class 1 - Motor Control Centers

Equipment Description

CAVEATS - GERS (Identify with an asterisk (*) those caveats which are met by intent without meeting the specific wording of the caveat rule and explain the reason for this conclusion in the COMMENTS section below)

1.	Equipment is included in generic seismic testing equipment class	Y N U N/A
2.	Meets all Bounding Spectrum caveats	Y N U N/A
3.	Floor mounted cabinet	Y N U N/A
4.	Average weight per section less than 800 pounds	Y N U N/A
5.	..e anchorage utilizing MCC base channels	Y N U N/A
6.	..ate strength and stiffness in load transfer from anchorage to base frame (only for "function after" GERS)	Y N U N/A
7.	Essential relays have GERS > 4.5g (only for "function during" GERS)	Y N U N/A
8.	Able to reset starters (only for "function after" GERS)	Y N U N/A

Is the intent of all the caveats met for GERS? Y N U N/A

ANCHORAGE (will use conservative assumptions for calcs)

1	Appropriate equipment characteristics determined (mass, CG, natural freq., damping, center of rotation)	Y N U N/A
2.	Type of anchorage covered by GIP	Y N U N/A
3.	Sizes and locations of anchors determined	Y N U N/A
4.	Adequacy of anchorage installation evaluated (weld quality and length, nuts and washers, expansion anchor tightness, etc.)	Y N U N/A
5.	Factors affecting anchorage capacity or margin of safety considered: embedment length, anchor spacing, free-edge distance, concrete strength/condition, and concrete cracking	Y N U N/A
6.	For bolted anchorages, gap under base less than 1/4-inch	Y N U N/A
7.	Factors affecting essential relays considered: gap under base, capacity reduction for expansion anchors	Y N U N/A
8.	Base has adequate stiffness and effect of prying action on anchors considered	Y N U N/A
9.	Strength of equipment base and load path to CG adequate	Y N U N/A
10.	Embedded steel, grout pad or large concrete pad adequacy evaluated	Y N U N/A

Are anchorage requirements met? Y N U

SCREENING EVALUATION WORK SHEET (SEWS)

Equip. ID No. MCC 16B21 Equip. Class 1 - Motor Control Centers

Equipment Description ______

INTERACTION EFFECTS

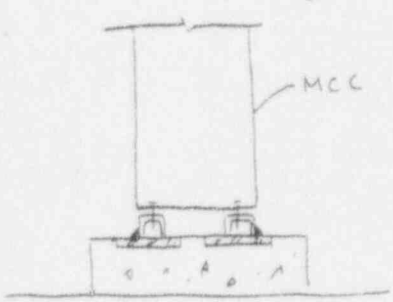
1. Soft targets free from impact by nearby equipment or structures — Y N U N/A
2. If equipment contains sensitive relays, equipment free from all impact by nearby equipment or structures — Y N U N/A
3. Attached lines have adequate flexibility — Y N U N/A
4. Overhead equipment or distribution systems are not likely to collapse — Y N U N/A
5. Have you looked for and found no other adverse concerns? — Y N U N/A

Is equipment free of interaction effects? Y N U

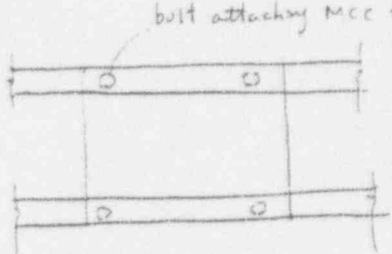
IS EQUIPMENT SEISMICALLY ADEQUATE? Y N U

COMMENTS

Each panel is bolted to inverted channel w/ 4 bolts. The channels are welded to steel embeds w/ continuous welds.



Side View



Evaluated by: W.H. Tong [signature] Date: 12/91

Revision 2, Corrected, 6/28/91
Status Y N U

SCREENING EVALUATION WORK SHEET (SEWS) Sheet 1 of 2

Equip. ID No. HPCS D/G Equip. Class 13 - Motor-Generators

Equipment Description Diesel Generator

Location: Bldg. D/G Floor El. Grade Room, Row/Col

Manufacturer, Model, Etc. (optional) GM (motor)

SEISMIC CAPACITY VS DEMAND

1. Elevation where equipment receives seismic input Grade level
2. Elevation of seismic input below about 40' from grade Y N U
3. Equipment has fundamental frequency above about 8 Hz Y N U N/A
4. Capacity based on: Existing Documentation DOC
   Bounding Spectrum BS
5. Demand based on: Ground Response Spectrum GRS
   1.5 x Bounding Spectrum ABS
   Conserv. Des. In-Str. Resp. Spec. CRS
   Realistic M-Ctr. In-Str. Resp. Spec. RRS

Does capacity exceed demand? Y N U N/A

CAVEATS - BOUNDING SPECTRUM (Identify with an asterisk (*) those caveats which are met by intent without meeting the specific wording of the caveat rule and explain the reason for this conclusion in the COMMENTS section below)

1. Equipment is included in earthquake experience equipment class Y N U N/A
2. Main driver and driven equipment connected by a rigid support or skid * But each is well anchored to a common foundation mat Y N* U N/A
3. Base vibration isolators adequate for seismic loads Y N U N/A
4. Attached lines have adequate flexibility Y N U N/A
5. Anchorage adequate (See checklist below for details) Y N U N/A
6. Relays mounted on equipment evaluated Y N U N/A
7. Have you looked for and found no other adverse concerns? Y N U N/A

Is the intent of all the caveats met for Bounding Spectrum? Y N U N/A

ANCHORAGE

1. Appropriate equipment characteristics determined (mass, CG, natural freq., damping, center of rotation) Y N U N/A
2. Type of anchorage covered by GIP Y N U N/A
3. Sizes and locations of anchors determined Y N U N/A
4. Adequacy of anchorage installation evaluated (weld quality and length, nuts and washers, expansion anchor tightness, etc.) Y N U N/A
5. Factors affecting anchorage capacity or margin of safety considered: embedment length, anchor spacing, free-edge distance, concrete strength/condition, and concrete cracking Y N U N/A

Figure 5.2-2: SEWS form of HPCS Diesel Generator

Revision 2, Corrected, 6/28/91
Sheet 2 of 2

SCREENING EVALUATION WORK SHEET (SEWS)

Equip. ID No. HPCS D/G Equip. Class 13 - Motor-Generators

Equipment Description

ANCHORAGE (Cont'd)

6. For bolted anchorages, gap under base less than 1/4-inch — Y N U N/A
7. Factors affecting essential relays considered: gap under base, capacity reduction for expansion anchors — Y N U N/A
8. Base has adequate stiffness and effect of prying action on anchors considered — Y N U N/A
9. Strength of equipment base and load path to CG adequate — Y N U N/A
10. Embedded steel, grout pad or large concrete pad adequacy evaluated — Y N U N/A

Are anchorage requirements met? Y N U

INTERACTION EFFECTS

1. Soft targets free from impact by nearby equipment or structures — Y N U N/A
2. If equipment contains sensitive relays, equipment free from all impact by nearby equipment or structures — Y N U N/A
3. Attached lines have adequate flexibility — Y N U N/A
4. Overhead equipment or distribution systems are not likely to collapse — Y N U N/A
5. Have you looked for and found no other adverse concerns? — Y N U N/A

Is equipment free of interaction effects? Y N U

IS EQUIPMENT SEISMICALLY ADEQUATE? Y N U

COMMENTS

Evaluated by: W. H. Tong Date: 12/91

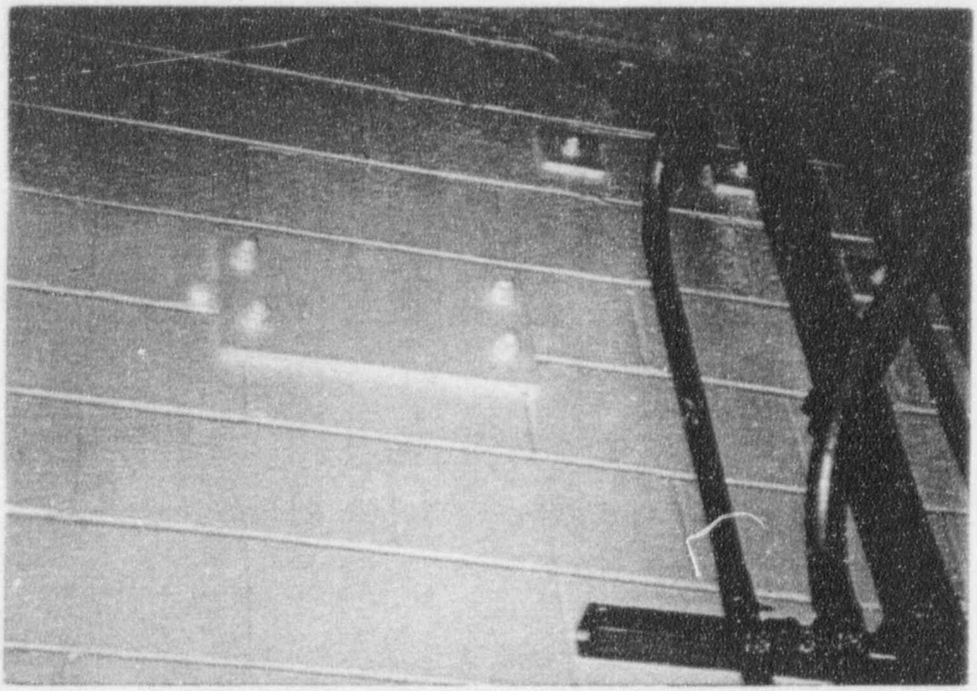


Figure 5.2-3: Masonry block wall near the 480V load center

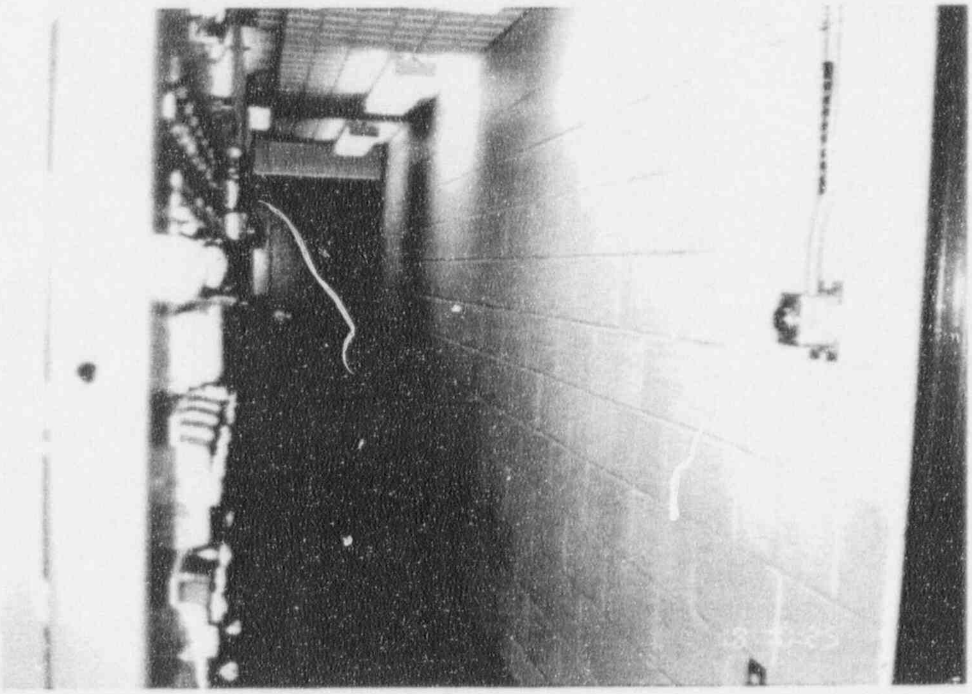


Figure 5.2-4: Masonry block wall at the control room

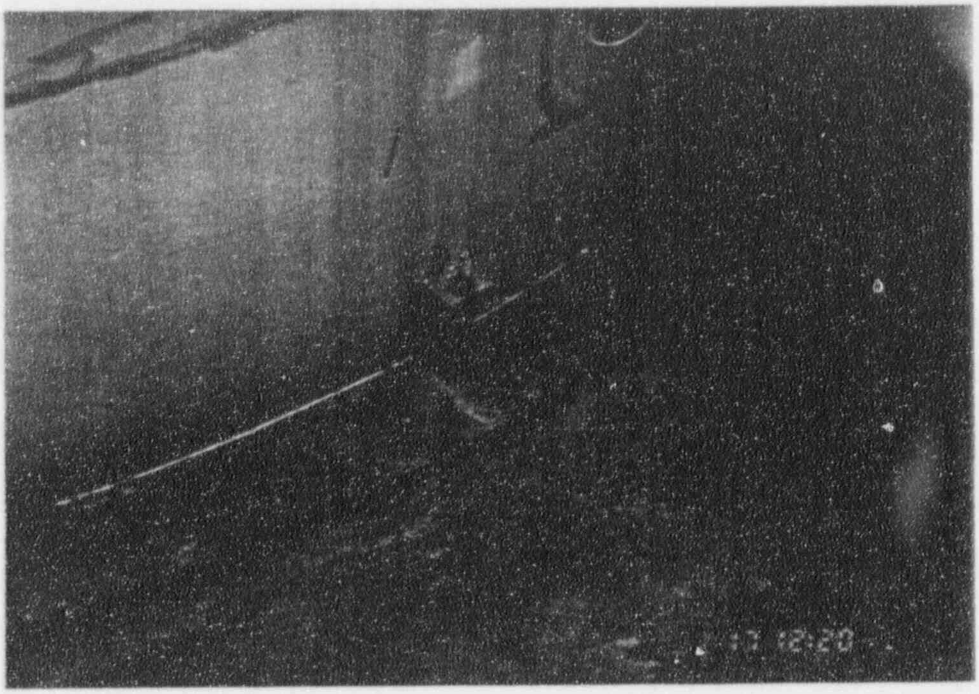


Figure 5.2-5: Base anchorage detail of the condensate water storage tank

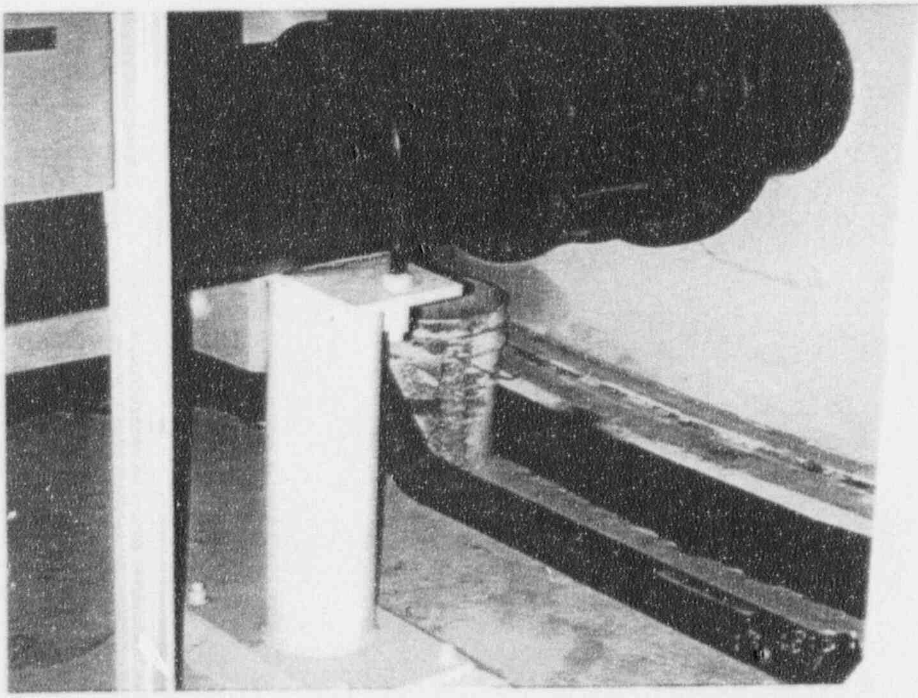
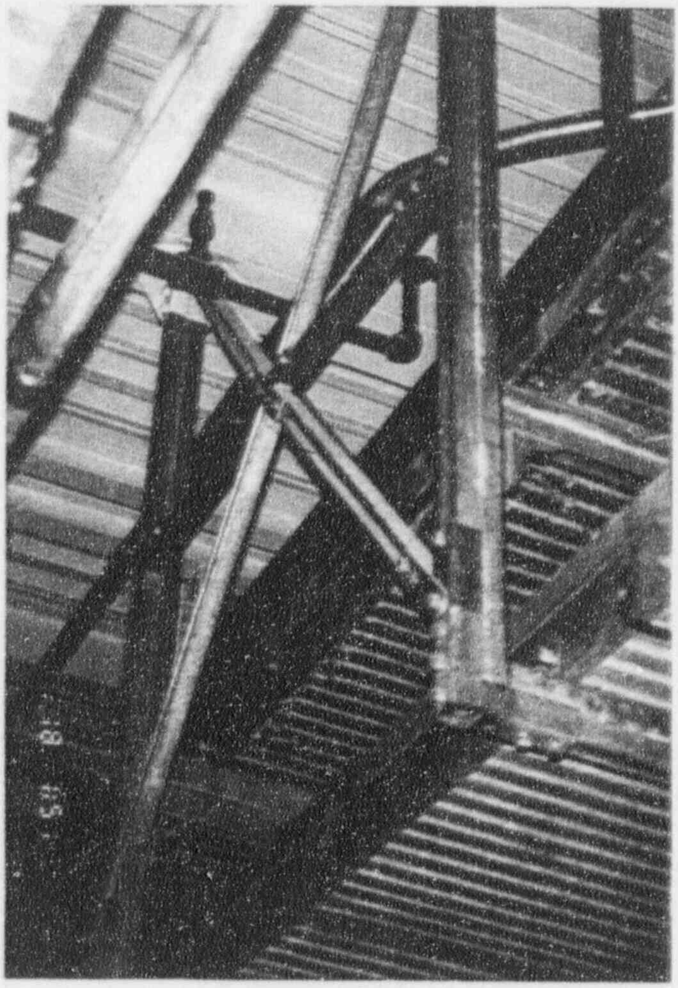


Figure 5.2-6: Pipe at the base of the 300,000 gallon fire water tank (unanchored)

Figure 5.2-7:
Trapeze cable tray supported
off the floor above



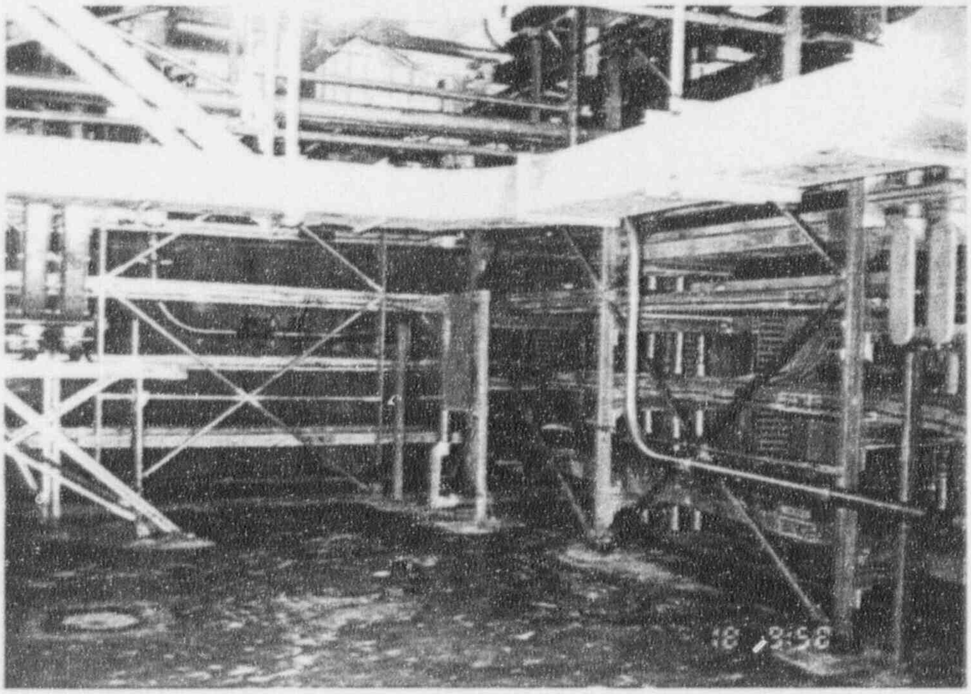
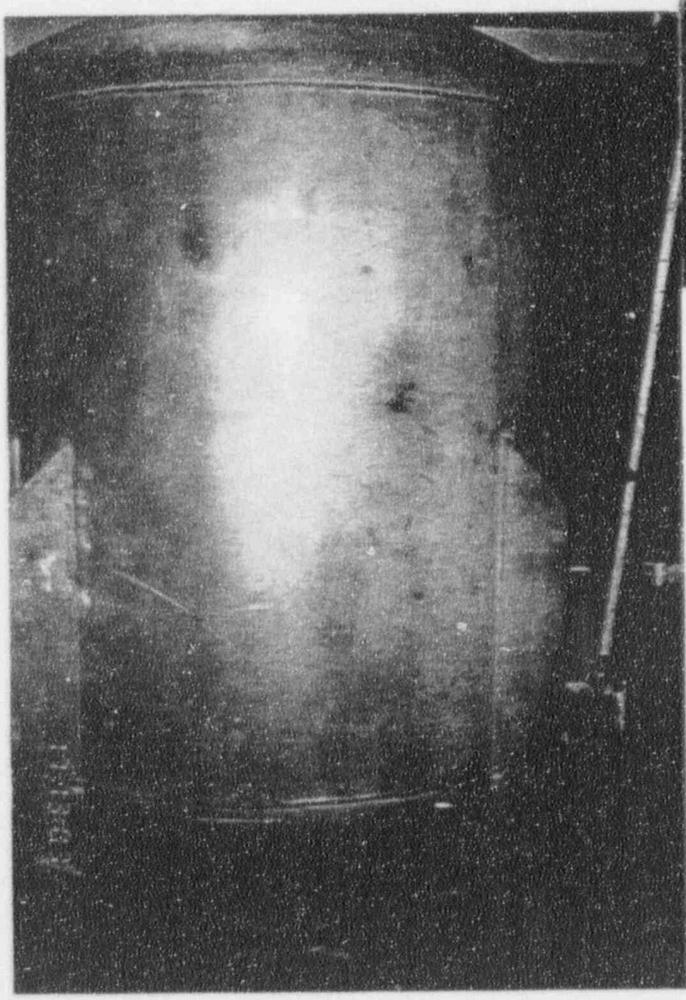


Figure 5.2-8: Cable trays supported on the floor

Figure 5.2-9:
Day tank of the emergency diesel generator



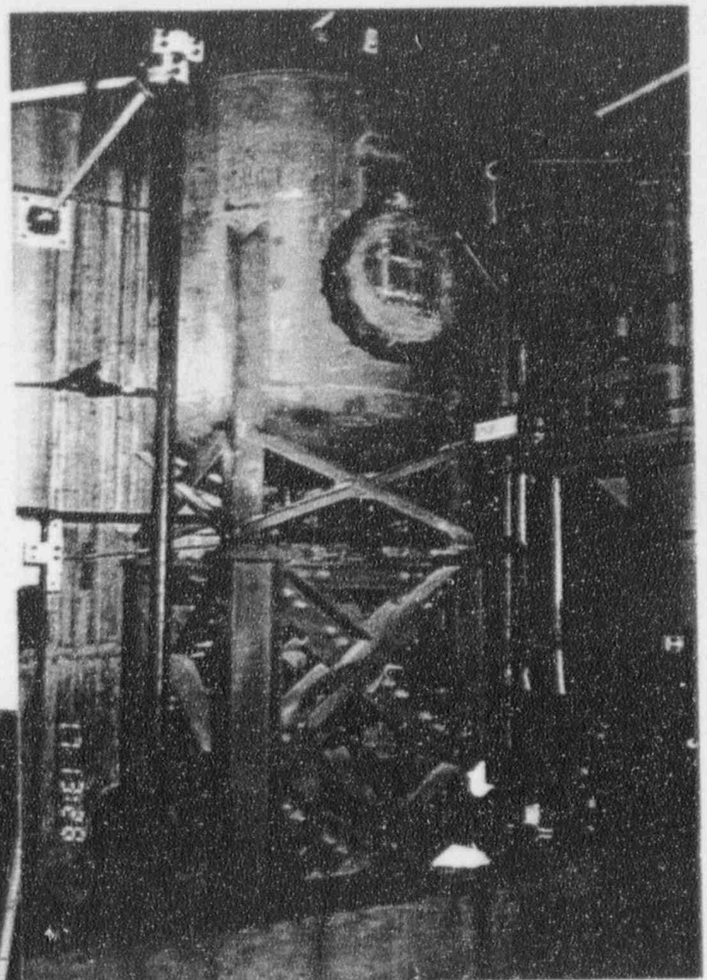


Figure 5.2-10:
Day tank of the HPCS diesel generator

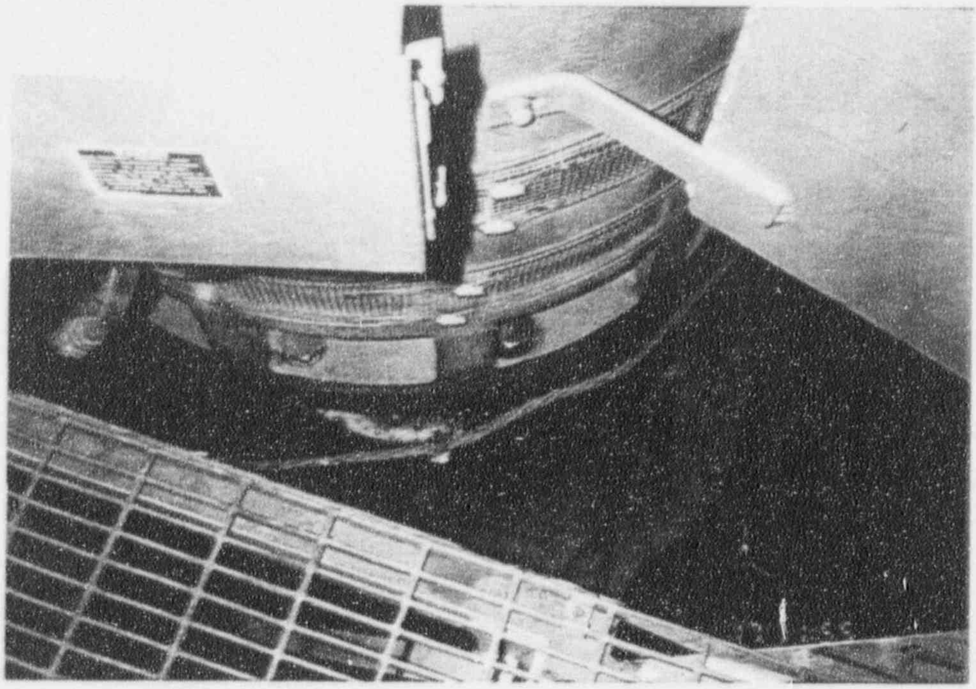


Figure 5.2-11: SSW pump motor mounting

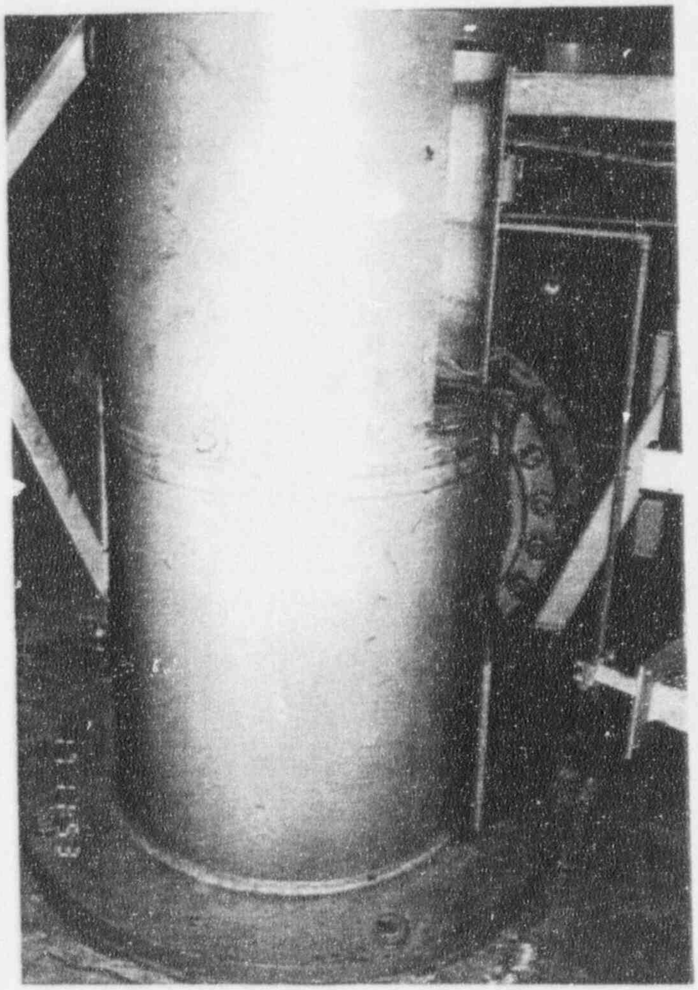
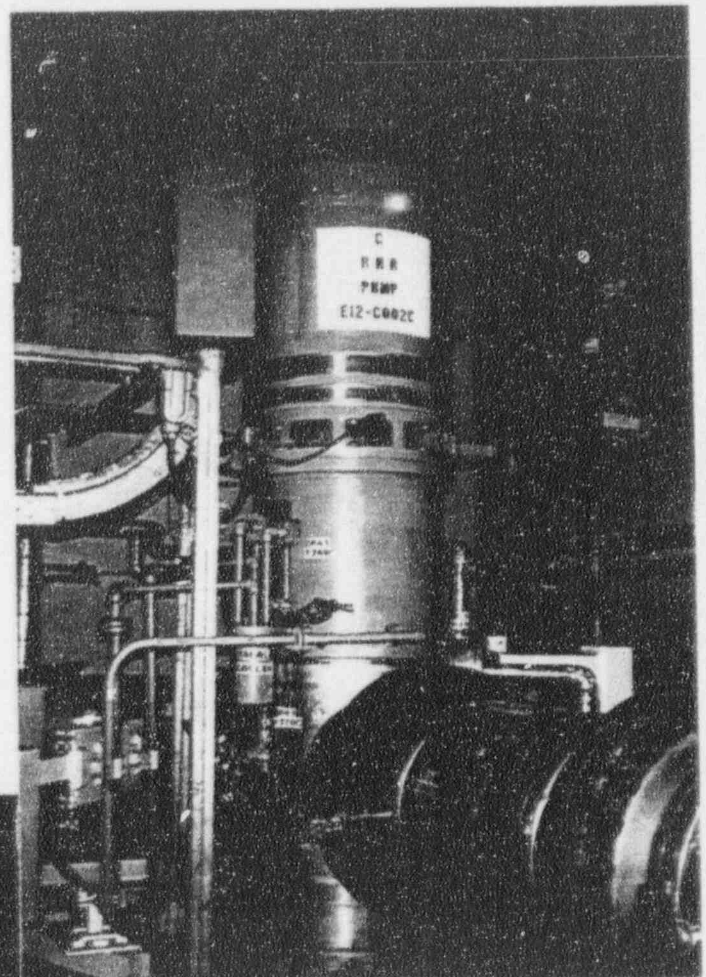


Figure 5.2-12:
Base anchorage of the SSW pump

Figure 5.2-13:
RHR Pump



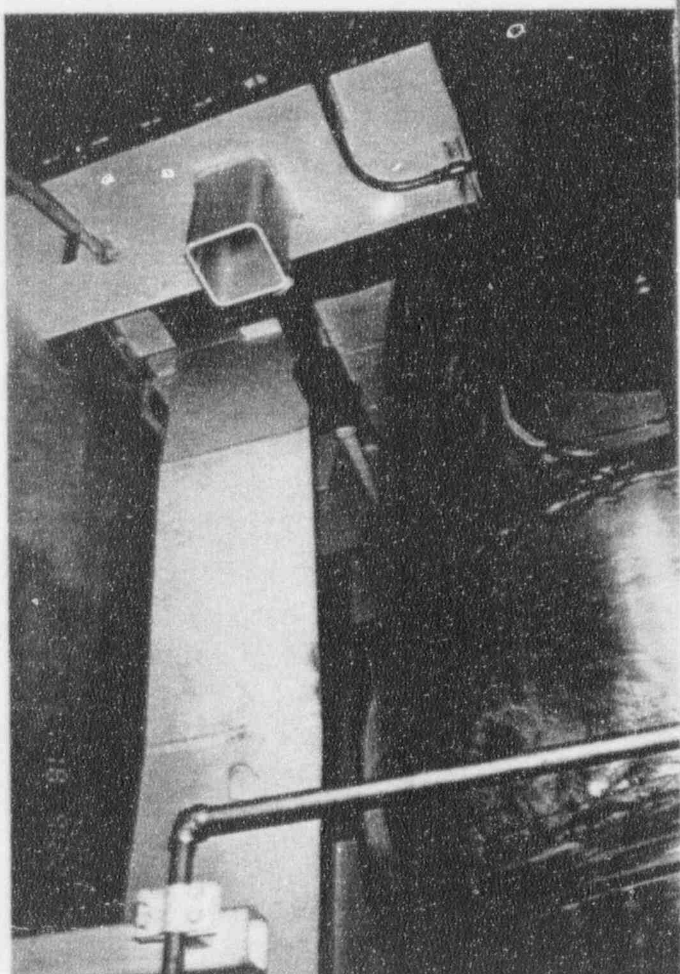


Figure 5.2-14:
RHR heat exchanger support column

Figure 5.2-15: RHR heat exchanger support column anchorage

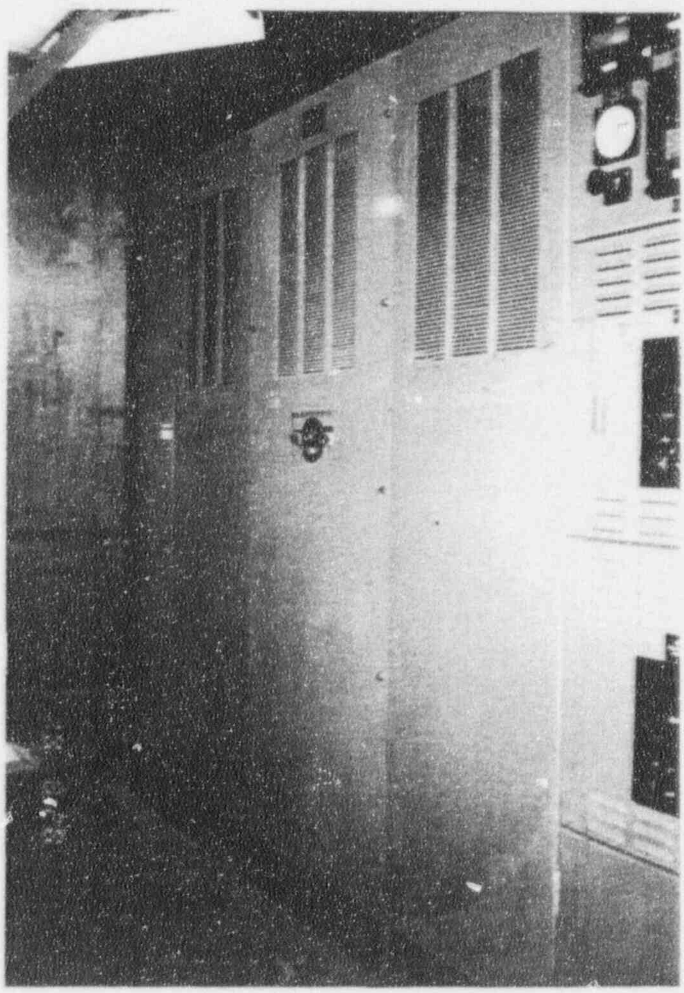


Figure 5.2-16:
480V load center

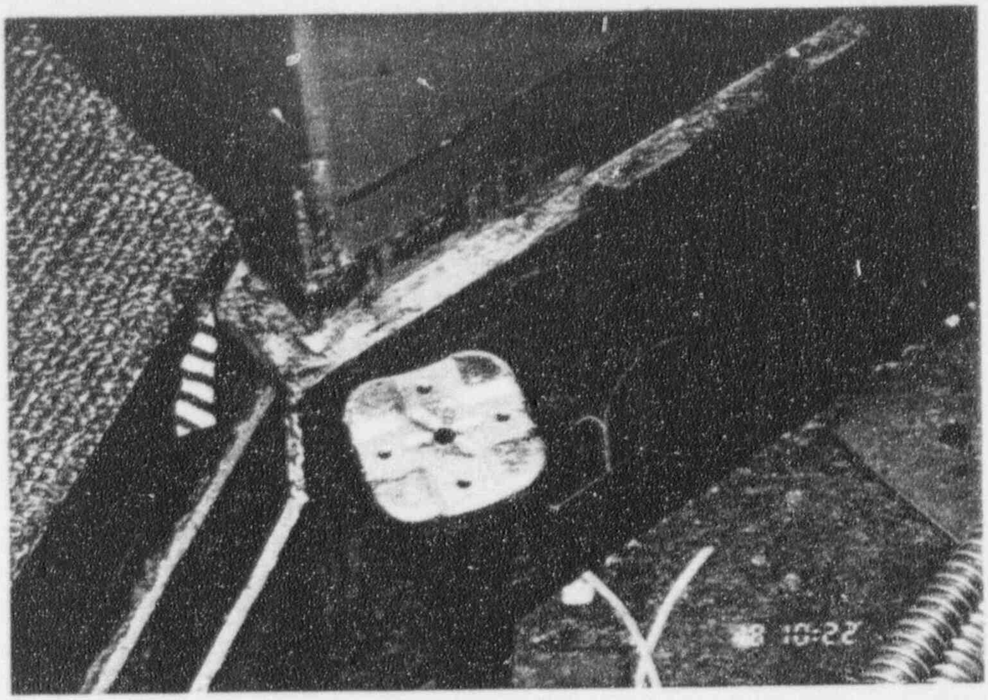


Figure 5.2-17: Anchorage detail of control room panels

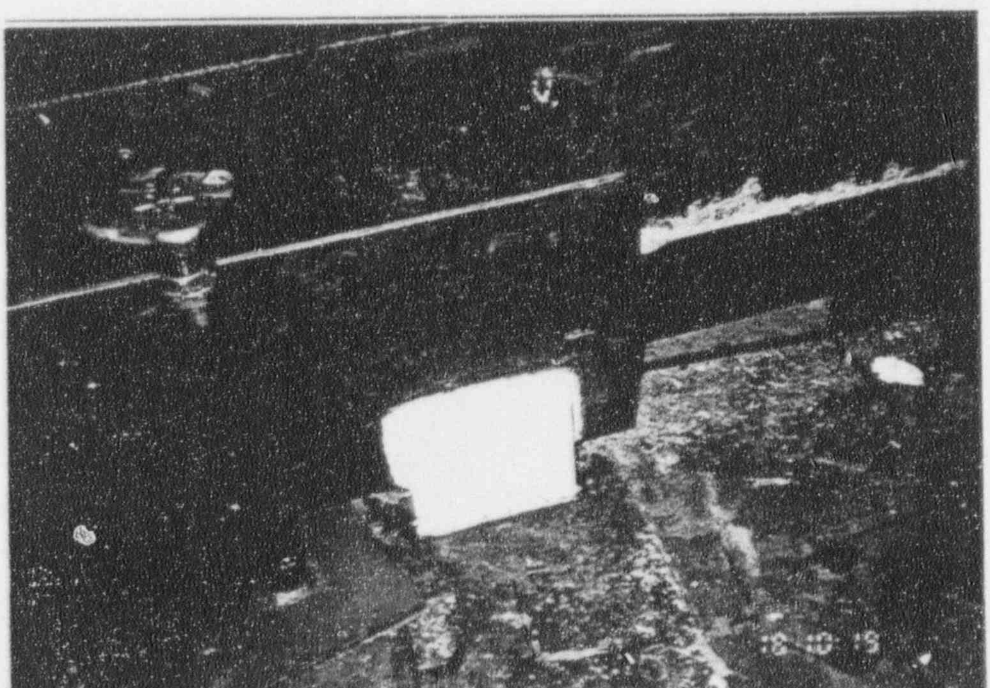


Figure 5.2-18: Details of control room raised floor

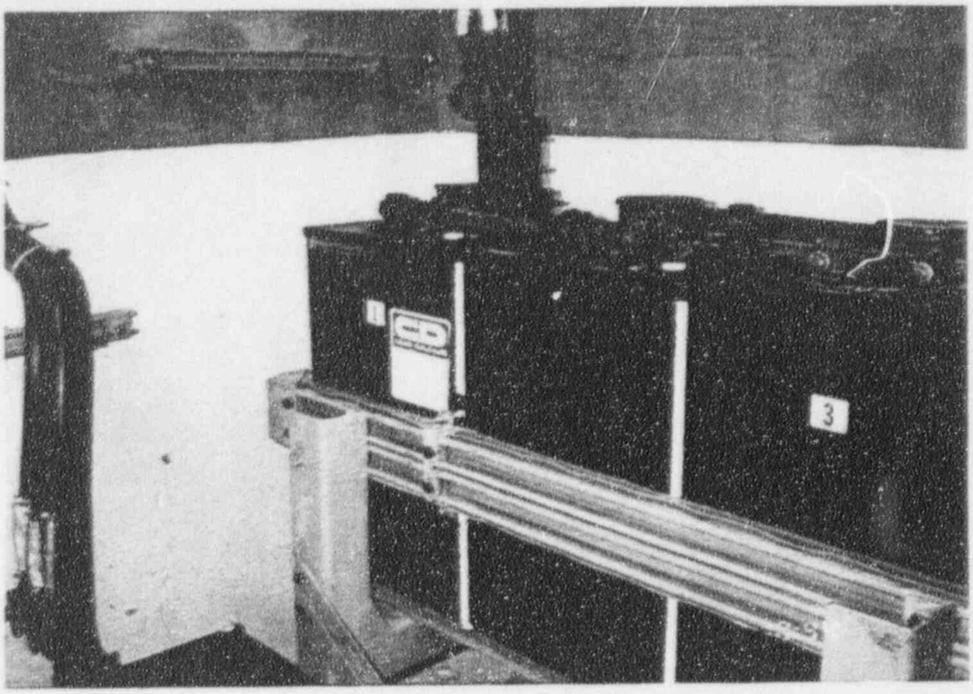


Figure 5.2-19: Station battery details

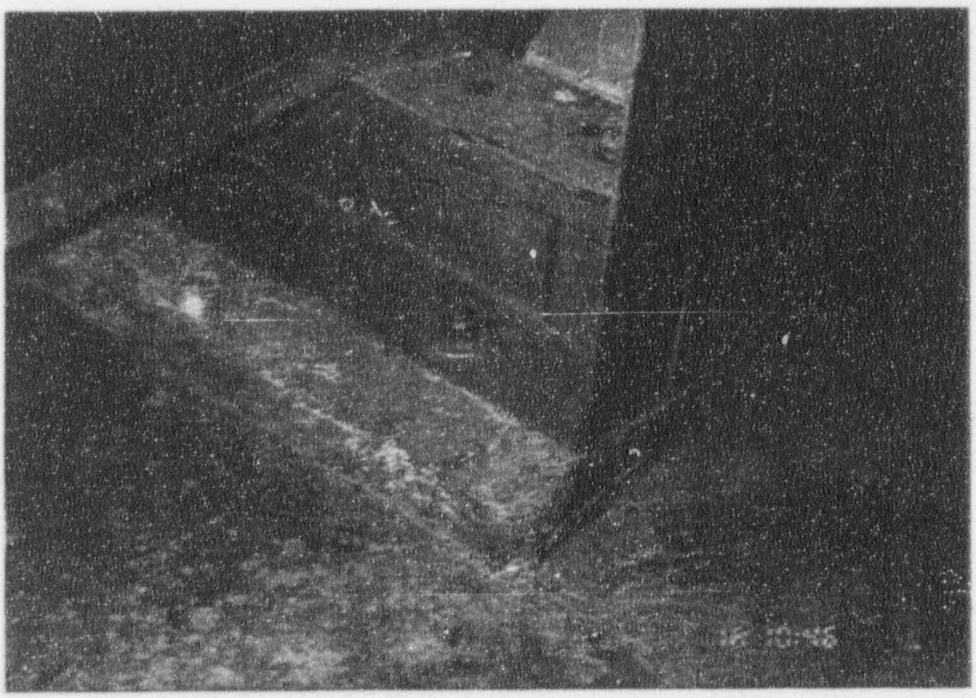


Figure 5.2-20: Station battery rack anchorage details

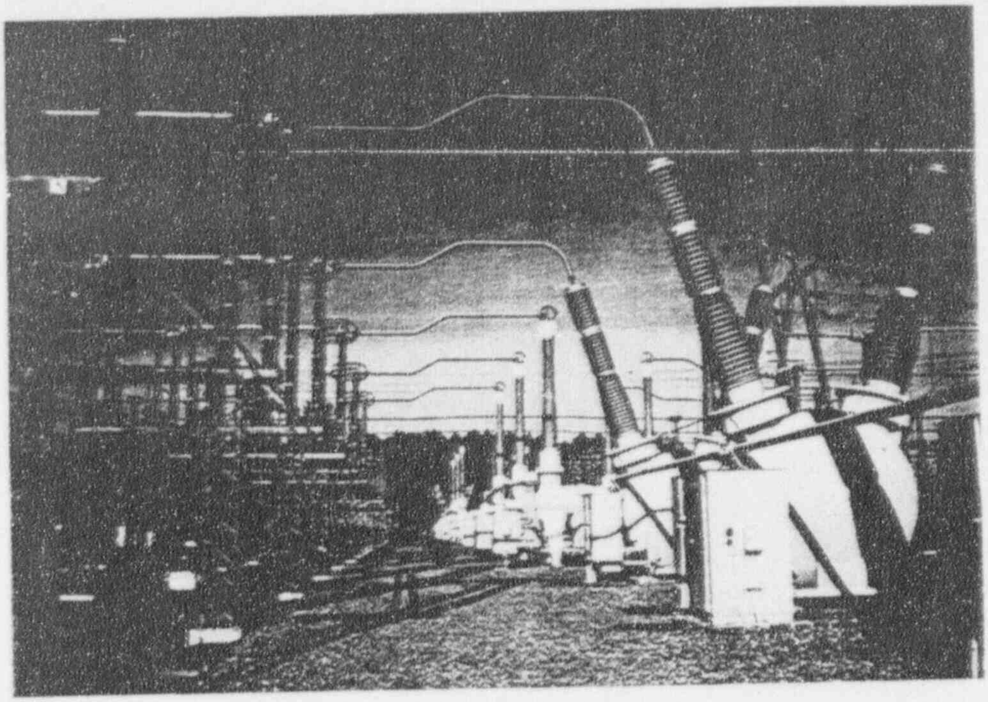


Figure 5.2-21: Two types of circuit breakers at the Grand Gulf 500kV switchyard

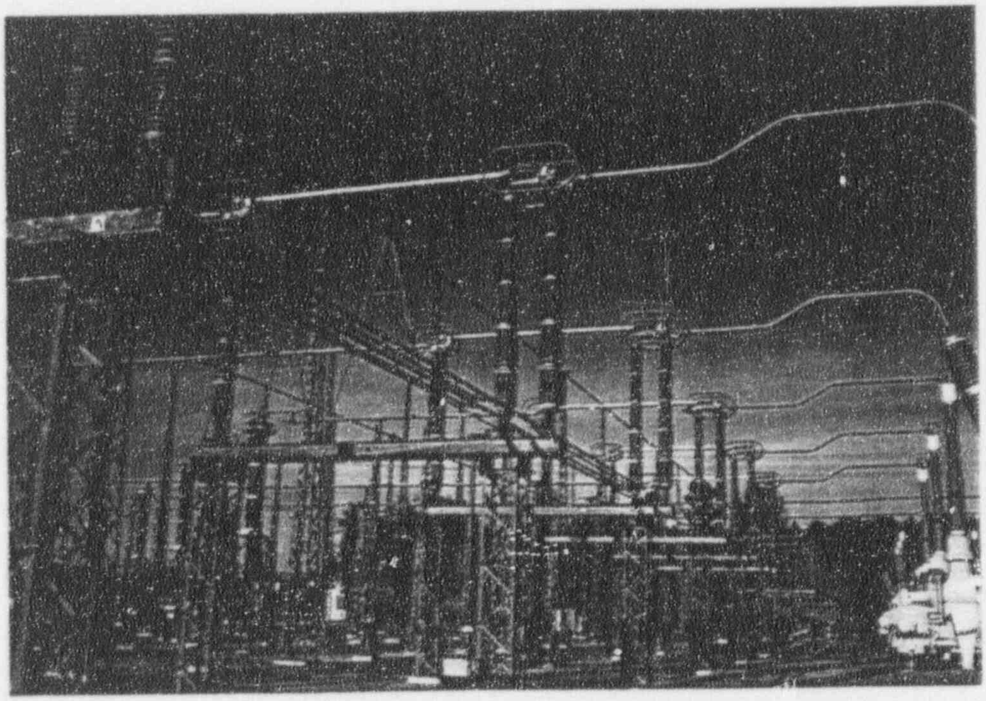


Figure 5.2-22: Disconnect switches at the 500kV switchyard

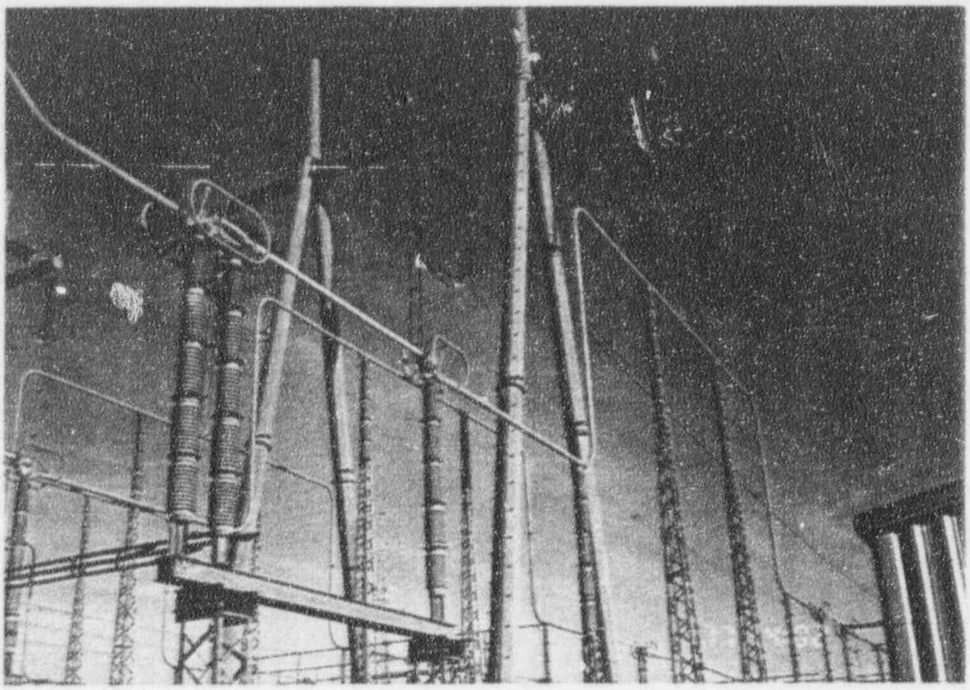


Figure 5.2-23: 500kV switchyard bus support structure and ceramic insulators

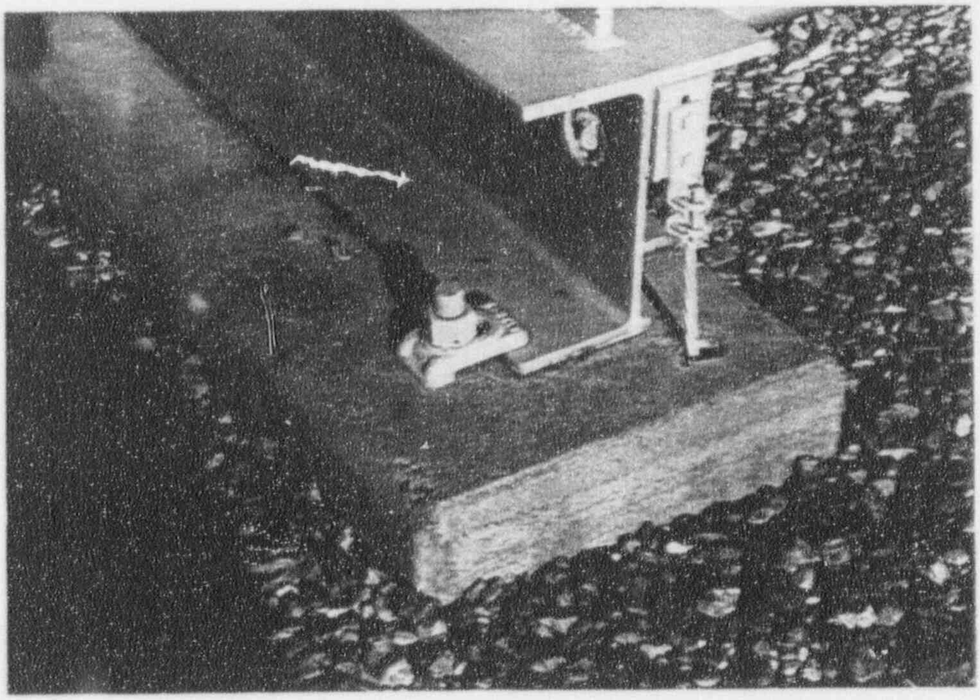
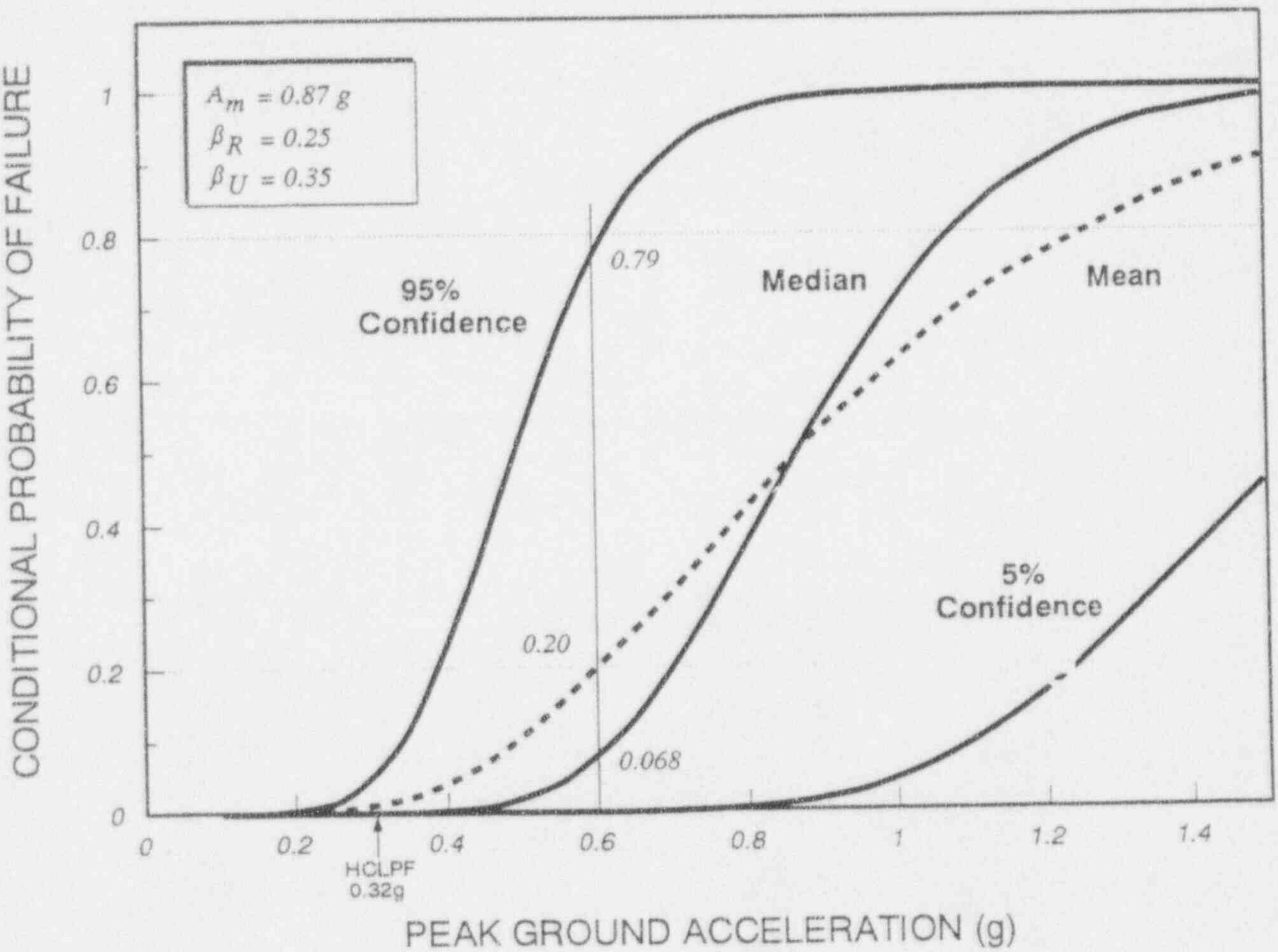


Figure 5.2-24: Clip at the base of circuit breaker support skid



2HD 072nb/CHART-1

Figure 5.3-1: Median, 5% Non-Exceedance, and 95% Non-Exceedance Fragility Curves For a Component

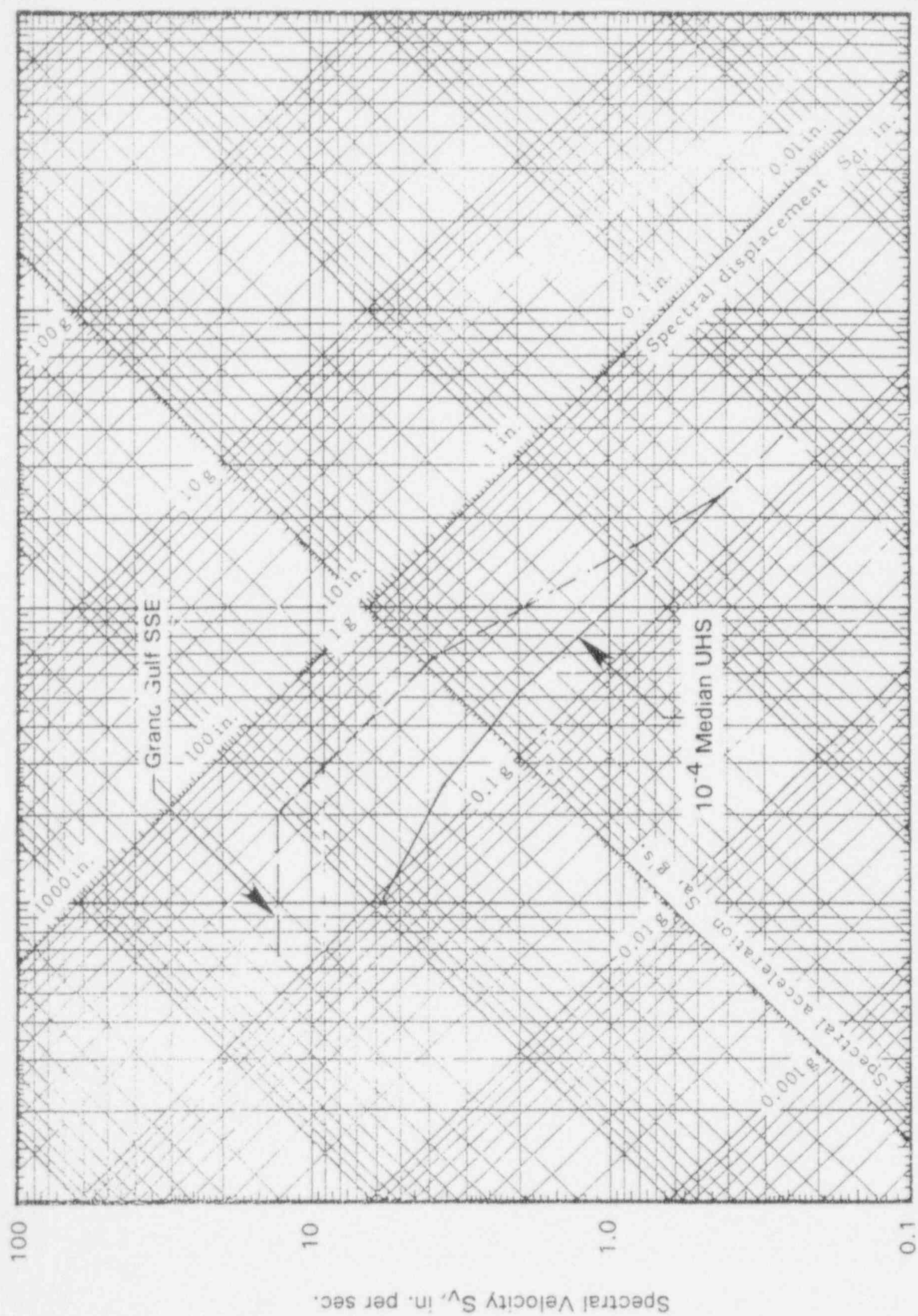


Figure 5.3-2: Comparison of Grand Gulf SSE Design Spectrum with 10^{-4} UHS

6. SEISMIC RISK QUANTIFICATION FOR GRAND GULF

6.1 Base Case

The key elements of seismic risk quantification are seismic hazard analysis (Chapter 3), systems analysis (Chapter 4) and seismic fragility evaluation (Chapter 5). In this Chapter, these are assembled together to obtain estimates of the frequencies of different plant operating states.

In Chapter 4, Plant Operating State (POS) 5 during refueling outages was studied, because it was selected by the Sandia team (Ref. Sandia, 1994) for analysis for internal initiators during shutdown. Based on the systems analysis described in Chapter 4, we have developed a simple Boolean equation for POS 5, given in Chapter 4.

Note that the basic event in this Boolean equation is a seismic-induced failure (service-water system failure). The seismic fragilities of components have been estimated as described in Chapter 5.

The seismic quantification, using the methodology described in Chapter 5, was done using the software package EQESRA (proprietary to EQE International Inc.) which takes as input the family of seismic hazard curves and the family of seismic fragility curves for all components appearing in the Boolean equations. The component failures are treated as statistically independent in the computations.

Table 6-1 shows the base case core-damage-frequency results. The base case consists of the Grand Gulf plant with both EPRI and LLNL seismic hazard curves. Note that the LLNL hazard curves being used are from the 1993 LLNL study (Ref. Sobel, 1993), not from the earlier 1989 LLNL study (Ref. Bernreuter et al., 1989) that has now been superseded. In Table 6-1, the mean, median, 5 percentile and 95 percentile core-damage frequencies for POS 5 are shown. It is seen from the table that the mean annual core-damage frequency for POS 5 is below 10^{-7} per year using either the LLNL or the EPRI seismic hazard curves. Therefore, we conclude that the seismic contribution to core damage frequency in POS 5 during refueling outages is very small at Grand Gulf.

6.2 Sensitivity Studies

Because one objective of this study is to derive generic conclusions on the significance of seismic events to shutdown risks, it is of interest to know how the seismic core damage frequency would vary with the site location. For this purpose, we assumed that the Grand Gulf plant could be at a site with a seismic hazard typical of the ensemble of sites in the eastern United States. The Zion nuclear power plant site in Illinois was chosen. To study the effect for the plant with one of the highest seismic hazards in the eastern U.S., we chose the Pilgrim site in Massachusetts. Table 6-2 compares the core-damage frequency of plant operating state POS 5 for the two sites with that at the Grand Gulf site.

It can be seen from Table 6-2 that the core-damage frequency would be a factor of about 30 higher if the Grand Gulf plant were located at Zion, and slightly less than a factor of 200 higher if it were located at the Pilgrim site.

6.3 Comparison with CDF from Internal Initiators During Cold Shutdown and with CDF at Full Power

It is instructive to compare the results for annual core-damage frequency (CDF) from this study with the CDF during shutdown arising from so-called "internal initiators", which has been analyzed by Sandia National Laboratories (Ref. Sandia, 1994). Also instructive is a comparison with the NUREG-1150 findings for CDF at Grand Gulf for full-power operation.

The comparison of core-damage-frequency results is shown in Table 6.3. From examining the Table, several important observations emerge:

- During shutdown conditions (in POS 5) the total annual mean CDF arising from earthquakes is small compared to the CDF arising from internal initiators from (Ref. Sandia, 1994): a factor of about 30 smaller for the LLNL seismic hazard curves and a factor of about 800 smaller using the EPRI hazard curves.

- The CDF per year at full power, from the NUREG-1150 analysis (Ref. NRC/1150, 1990) is a factor of 2 larger than the CDF arising from internal initiators during POS 5, so the results of this study are even smaller when compared with those from NUREG-1150.

- The Error Factor (EF) in this seismic study is significantly greater than the EF in Sandia's analysis of CDF from internal initiators during refueling outages in POS 5. This is primarily due to the large uncertainty in the seismic hazard curves but another contribution arises from the uncertainty in the seismic fragilities.

TABLE 6.1

CORE-DAMAGE-FREQUENCY ESTIMATES
(all values are core-damage frequency per year)

POS State	Mean	Median	5% Confidence	95% Confidence
LLNL (1993) Hazard Curves				
POS 5	7.1 E-8	2.4 E-9	2.1 E-11	2.2 E-7
EPRI Hazard Curves				
POS 5	2.5 E-9	2.0 E-10	2.5 E-12	1.1 E-8

TABLE 6.2

SENSITIVITY OF POS 5 CDF TO SEISMIC SITE HAZARD

(all values are core-damage frequency per year)
(using EPRI hazard curves)

Site	Mean	Median	95% Confidence
Grand Gulf Site	2.5 E-9	2.0 E-10	1.1 E-8
Pilgrim Site	4.7 E-7	3.7 E-8	2.1 E-6
Zion Site	7.8 E-8	5.3 E-9	3.7 E-7

NOTE: This sensitivity study represents "moving" the Grand Gulf reactor to the other two sites shown, with all other features of Grand Gulf remaining the same.

TABLE 6.3

GRAND GULF: COMPARISONS OF CDF FOR SHUTDOWN vs. FULL-POWER CONDITIONS AND FOR SEISMIC INITIATORS vs. INTERNAL-INITIATORS

Analysis Condition	Reference	Mean CDF/year	Error Factor[a]
Shutdown (POS 5), internal initiators[b]	Sandia (Ref. Sandia,1994)	2.0 E-6	4.2
Shutdown (POS 5) seismic initiator	This study	7.1 E-8 (LLNL)[c] 2.5 E-9 (EPRI)[c]	90 55
Full power internal initiators[b]	NUREG-1150 (Ref. NRC/1150,1990)	4.0 E-6	10

Footnotes for Table 6.3:

a) The Error Factor is the ratio of the 95%-percentile value to the median value of CDF (core-damage frequency/year).

b) The notation "internal initiators" includes all initiators that start with internal plant faults or loss of offsite power, but excludes internal fires and internal flooding.

c) The notation (LLNL) and (EPRI) indicates use of the LLNL or the EPRI seismic hazard curves for the Grand Gulf site.

7. CONCLUSIONS AND SUMMARY

A number of important insights emerge from this Grand Gulf analysis, including:

Core-damage frequency: The core-damage frequency for earthquake-initiated accidents during refueling outages in Plant Operating State (POS) 5 (cold shutdown) is found to be low in absolute terms, below 10^{-7}/year. The reasons for this are (i) Grand Gulf's seismic capacity in responding to earthquakes during shutdown is excellent, well above its design basis; (ii) the Grand Gulf site enjoys one of the least seismically active locations in the United States; and (iii) the Grand Gulf plant is only in POS 5 for an average (mean) of 3.1% of the time.

The core-damage frequency is also low relative to the core-damage frequency during POS 5 for internal initiators, as analyzed in the companion study by Sandia (Ref. Sandia, 1994). This can be seen in Table 6.3.

The results are plant-specific: We believe that the results for Grand Gulf are highly plant-specific, in the sense that the seismic capacities, the specific cut set that is found to be most important, and the seismicity of the site are all difficult to generalize to other reactors elsewhere.

Shutdown seismic sequences are similar to full-power seismic sequences: Nevertheless, it is important to observe that all of the sequence types, components, and human errors that emerge in the key sequences in this analysis are similar or identical to sequences, components, and human errors that appear in typical full-power seismic PRAs. That is, nothing that has arisen as important in this study appears to be unique to earthquakes occurring during shutdown conditions. Whether this observation is generalizable to other reactors at other sites is unknown to us.

Sensitivities: Sensitivity studies reveal that if the Grand Gulf reactor were moved to the Zion site in Illinois (a typical midwestern site) or the Pilgrim site in Massachusetts (one of the most seismically active sites among all of the reactor sites in the eastern U.S.), the mean annual CDF from this study would increase by a factor of about 30 or 200, respectively.

Uncertainties: While there are significant uncertainties in the numerical values of core-damage frequencies found in this study (see Tables 6.1 through 6.3), the above conclusions are relatively robust --- they do not depend on the detailed numerical values found.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects being performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL). Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied. The objectives of the program are to assess the risks of severe accidents initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The objective of this report is to document the approach utilized in the Grand Gulf plant and discuss the results obtained. A parallel report for the Surry plant is prepared by SNL.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Grand Gulf Reactor, reactor shutdown, risk assessment, failure mode analysis, reactor accidents, reactor core disruption, reactor start-up, RHR systems, systems analysis, thermodynamics, seismic events

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