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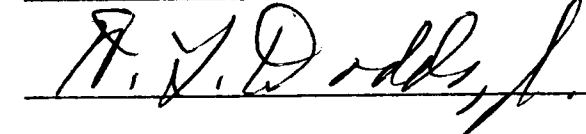
I am submitting herewith a thesis written by Martin A. Stutzke entitled "Seismically Induced Core-Damage Accidents in the Clinch River Breeder Reactor Plant." I have examined the final copy of this thesis for form and content and recommend that it be accepted in partial fulfillment of the requirements for the degree of Master of Science, with a major in Nuclear Engineering.



P. F. Pasqua, Major Professor

We have read this thesis and
recommend its acceptance:





Accepted for the Council:



The Graduate School

SEISMICALLY INDUCED CORE-DAMAGE ACCIDENTS
IN THE CLINCH RIVER BREEDER
REACTOR PLANT

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Presented for the
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Degree
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ABSTRACT

An estimate of the occurrence frequency of seismically induced core-damage accidents in the Clinch River Breeder Reactor Plant (CRBRP) was made using readily available data and techniques. Site seismicity was determined through statistical analysis of a historical listing of earthquakes surrounding the site. Component fragilities were based on analyses conducted by CRBRP vendors and information presented in light water reactor risk assessments. Seismically induced accident sequences leading to core damage were developed by event tree and fault tree analysis.

The total seismically induced core-damage frequency was determined to be 3.2×10^{-5} per year, which comprises 90% of the total core-damage frequency of the CRBRP. Earthquakes that cause the peak ground acceleration at the CRBRP site to exceed 0.25 g (the design-base earthquake) were found to be major contributors to the total frequency.

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CHAPTER I

INTRODUCTION

This thesis provides an estimate of the likelihood of seismically induced core-damage accidents in the Clinch River Breeder Reactor Plant. The following sections discuss the motivation for performing the study, identify the objectives of the study, provide an overview of the approach used to obtain the study's objective, and summarize the contents of the thesis.

I.1 MOTIVATION

Nuclear power plant designers have long realized that earthquakes may cause the failure of essential plant systems due to vibration or building collapse. Such seismically induced system failures may lead to plant shutdown, resulting in a loss of revenue to the utility. In addition, earthquakes may also fail redundant safety systems; such failures may preclude removal of decay heat from the reactor core, resulting in core damage and possible release of radionuclides to the environment.

In order to reduce the likelihood of seismically induced core-damage accidents, nuclear power plants are designed to satisfy a set of deterministic criteria. Briefly, the seismic design process consists of:

1. Establishment of a maximum credible earthquake which may affect the plant based on historical data,

2. Determination of the plant equipment and systems required to remain intact following an earthquake in order to ensure that core damage does not occur,
3. Calculation of the forces and moments imposed on the collection of required equipment due to occurrence of the maximum credible earthquake, and
4. Comparison of the calculated forces and moments to the forces and moments required to cause equipment failure.

However, the seismic design process cannot eliminate the possibility of seismically induced core-damage accidents. Earthquakes with magnitudes greater than the design-base earthquake may occur. Errors in equipment design, fabrication, and installation may cause some equipment to fail when subjected to loads below those imposed by occurrence of the design-base earthquake.

In order to assess the likelihood of seismically induced core-damage accidents, a number of probabilistic techniques have been developed in recent years. These techniques, which are collected in the PRA Procedures Guide¹, take into account the random nature of earthquake occurrence and equipment operability. Applied to a number of nuclear power plants, they show that the contribution of seismically induced core-damage accidents to the overall plant core-damage frequency may be significant. Hence, probabilistic assessments of seismically induced core-damage accidents provide insights into the adequacy of nuclear power plant design criteria.

I.2 OBJECTIVE

The objective of this thesis is the estimation of the seismically induced core-damage frequency of the Clinch River Breeder Reactor Plant (CRBRP) using readily available techniques and data. Such an estimate provides both a measure of potential accident risk in the CRBRP due to earthquakes and insights into the adequacy of the CRBRP design. In addition, the usability of existing seismic risk analysis techniques is tested through application to an actual nuclear plant design.

I.3 APPROACH

The seismically induced core-damage frequency of the Clinch River Breeder Reactor Plant (CRBRP) was evaluated by completion of the following four steps:

1. The CRBRP site was evaluated to obtain the seismic hazard in terms of the likelihood of occurrence of ground motions of various magnitudes.
2. The capacities of important plant structures and equipment to withstand earthquakes were evaluated to determine conditional probabilities of failure as a function of ground acceleration.
3. Event-tree and fault-tree logic models were developed to reflect plant response to seismic events.
4. The modified logic models were solved to obtain Boolean expressions for the seismic event sequences of interest.

These expressions were then quantified by convolving the probabilistic site seismicity and the fragilities for the plant structures and equipment obtained in steps 1 and 2.

Each of these steps was approached from the viewpoint of an analyst with a grasp of engineering principles, probabilistic methods, and the design and operation of nuclear power plants. In general, each step was completed through application of the methods expounded in the PRA Procedures Guide.¹

I.4 ORGANIZATION OF THE TEXT

The estimation of CRBRP site seismic hazard likelihood (site seismicity) is presented in Chapter II. Chapter III discusses the methods and data used to estimate plant structure and equipment fragilities. Seismically induced accident sequences are developed in Chapter IV. Chapter V describes the evaluation of seismically induced accident sequences using the information developed in Chapters II through IV. Results of the study, along with a discussion of the factors affecting the uncertainty of the final results, are presented in Chapter VI. Conclusions resulting from the analysis are given in Chapter VII.

CHAPTER II

SEISMICITY ANALYSIS

Site seismicity is defined as the annual frequency at which ground motion parameters at the CRBRP site exceed specified values due to the occurrence of earthquakes. An earthquake is a complex occurrence involving the geologic characteristics of the site, the buildup of crustal strains, slippage along fault planes, rupture surface, focus, and other similar variables. The transmission of earthquake disturbance from the source to the site depends on the magnitude, distance, depth of focus, geologic characteristics of the region, type of fault movement, characteristics of shock waves (body and surface waves), and other similar effects. Ground motion at the site may be characterized by parameters such as peak acceleration, velocity, displacement, the duration of motion, and a set of response-spectra amplitudes corresponding to the modal frequencies and damping of the structure. Other possible descriptors are the Modified Mercalli (MM) intensity or energy-related ground-motion-intensity quantities, such as the root mean square of the ground acceleration or velocity.

Obviously, a complete description of the seismic hazard at the site should include all of the above variables for earthquake occurrence, wave transmission, and ground-motion input to the buildings. In this analysis, the modeling was simplified by considering only a few key variables. The specific information considered was:

1. A historical listing of earthquakes surrounding the CRBRP site. Each event was characterized by two parameters:
 - a. Its Modified Mercalli intensity, and
 - b. The distance from its epicenter to the CRBRP site.
2. The peak ground acceleration at the CRBRP site due to the occurrence of the above earthquakes.

The following sections discuss the methods used to transform the historical earthquake data into the CRBRP site seismicity, summarize the data used in the analysis, and present the final CRBRP site seismicity.

II.1 METHOD

The seismic hazard at the CRBRP site results from the potential occurrence of earthquakes whose effects can be transmitted to the site. Traditional methods summarized in the PRA Procedures Guide¹ require the identification of source areas for these earthquakes, postulation of the frequencies and magnitudes of possible future earthquakes from historical data, and calculation of the degree to which their intensities would be attenuated in being transmitted to the site. However, the causal mechanisms for earthquakes occurring in the eastern United States are not well understood. As a result, it is not generally possible to identify the specific geologic features that are responsible for eastern earthquakes. Hence, it was

decided to partition historical earthquake data according to the distance between the earthquake's epicenter and the site.

The annual frequency of earthquakes in which the CRBRP site's peak ground acceleration A is smaller than a specified acceleration a was obtained by assuming that earthquakes are Poisson events. (There is some doubt that this assumption is valid;^{2,3,4} however, the Poisson assumption is commonly used.¹⁾ Accordingly, the annual frequency of earthquakes in which the CRBRP site peak ground acceleration A is greater than or equal to a specified acceleration a is given by:

$$1 - \exp [-v(a)] \quad (\text{II-1})$$

where:

$v(a)$ = annual mean number of earthquakes in which $A \geq a$ from
all sources

The term " $1 - \exp [-v(a)]$ " is called the site seismicity.

In order to calculate $v(a)$, a historical listing of N earthquakes that occurred during T years was partitioned into 50 groups. Each group contained approximately an equal number of earthquakes; group boundaries were defined by the distance to the CRBRP site from the epicenters of the group's earthquakes. That is, each group consisted of about $N/50$ earthquakes occurring in T years such that the distance from the site to the epicenter of each earthquake in the group fell within a specified inner and outer radius.

The distribution of earthquakes in a group according to their magnitudes was obtained by use of the truncated Gutenberg-Richter correlation:

$$\log_{10} N(m) = \begin{cases} a & m < m_0 \\ a - bm & m_0 \leq m \leq m_x \\ 0 & m > m_x \end{cases} \quad (\text{II-2})$$

where:

$N(m)$ = number of earthquakes with magnitude greater than or equal to m over a given time interval T

m_0 = lower magnitude bound

m_x = upper magnitude bound

(Historical data suggests that earthquake magnitudes saturate;⁵ other correlations have been postulated to model this phenomenon.^{6,7,8,9} The truncated Gutenberg-Richter correlation represents a simplified approach to the actual phenomenon.)

Maximum likelihood techniques with grouped magnitude data were used to estimate the parameter b for each group. All earthquakes belonging to a certain group were placed into one of eleven magnitude intervals. The lowest magnitude considered, m_0 , was Modified Mercalli intensity 4.0 (Richter magnitude 3.5); the highest magnitude considered, m_x , was Modified Mercalli intensity 12.0 (Richter magnitude 9.0). Thus, the Richter width of each magnitude interval was 0.5. The following relation¹⁰ was solved numerically to determine the parameter b for each group:

$$\frac{1}{\beta} \left\{ \frac{\beta \delta}{\tanh(\beta \delta)} - \frac{\beta \frac{m_x - m_0}{2}}{\tanh\left(\beta \frac{m_x - m_0}{2}\right)} \right\} = \frac{\sum n_j m_j}{N} - \frac{m_x + m_0}{2} \quad (\text{II-3})$$

where:

$$\beta = b \ln(10) = 2.303b$$

δ = magnitude group half width (i.e., the magnitude range m_0 to m_x was divided into $(m_x - m_0)/2\delta$ intervals)

n_j = number of earthquakes in the group with magnitudes between $m_j - \delta$ and $m_j + \delta$

Given that an earthquake has occurred in a specified group j , the probability that its magnitude M lies between $m - \delta$ and $m + \delta$ is given by:

$$\Pr [m - \delta < m < m + \delta \mid E_j] = \frac{2e^{-\beta_j(m-m_0)} \sinh(\beta_j \delta)}{1 - e^{-\beta_j(m_x - m_0)}} \quad (\text{II-4})$$

Thus, the annual mean number of earthquakes with magnitudes between $m - \delta$ and $m + \delta$ for a particular group j is:

$$f_j(m - \delta < m < m + \delta) = \frac{N}{501} \Pr [m - \delta < m < m + \delta \mid E_j] \quad (\text{II-5})$$

Once each group was characterized by a frequency-magnitude relation (Equation II-5), the peak ground acceleration at the site, A , due to earthquakes in each group was determined. The following attenuation relation¹ was used to translate group magnitudes into site peak ground accelerations:

$$A_j = 0.584 \exp [-0.427 \exp (-0.444 m_b) + 1.098 m_b]$$

if

$$\bar{R}_j < 10 \text{ km}$$

$$A_j = 3.98 \bar{R}_j^{-5/6} \exp [-0.0427 \bar{R}_j \exp(-0.444 m_b) + 1.098 m_b]$$

if

$$\bar{R}_j \geq 10 \text{ km} \quad (\text{II-6})$$

where:

A_j = site peak ground acceleration in cm/s^2 due to an earthquake in the j th group with body wave magnitude m_b

m_b = body wave magnitude of the earthquake in the j th group
 = (Gutenberg-Richter magnitude - 1.71)/1.34

$\bar{R}_j$ = average distance from the site to the earthquake epicenter in km

Thus, the annual mean number of earthquakes from group j that cause site peak ground accelerations between a_l and a_u is:

$$v_j (a_l < A < a_u) = f_j (m - \delta < m < m + \delta) \quad (\text{II-7})$$

where:

a_l is found by Equation II-6 using $m_b = (m - \delta - 1.71)/1.34$

a_u is found by Equation II-6 using $m_b = (m + \delta - 1.71)/1.34$

$f_j (m - \delta < m < m + \delta)$ is found by Equation II-5.

The annual mean number of earthquakes from all groups that cause site peak ground acceleration A to exceed a specified acceleration a is:

$$v(a) = \sum_j \sum_{a_l \geq a} v_j (a_l < A < a_u) \quad (\text{II-8})$$

II.2 DATA

The historical listing of earthquake data used to estimate the CRBRP site seismicity was obtained from Table 2.5-2 of the CRBRP Preliminary Safety Analysis Report (PSAR).¹¹ Only earthquakes that had a reported Modified Mercalli intensity and whose epicenters were less than 750 nautical miles (nm) from the CRBRP site were considered. (The PSAR table lists a total of 535 earthquakes; of these, 104 events either had epicenters beyond 750 nm, or their intensities were not reported.) Figure II-1 presents a polar plot of the earthquakes used in the analysis; the CRBRP site is at the origin of this plot.

The CRBRP site lies in a relatively active region of the United States. As shown in Figure II-1, the predominant contribution to site seismic hazard is the Appalachian Mountains. In addition, the site is located about 300 nm from the New Madrid, Missouri earthquake region (this area bears 280° true from the CRBRP site) and about 300 nm from the Charleston, South Carolina earthquake region (this area bears 128° true from the site). These two regions are the source of the largest known earthquakes (Modified Mercalli intensities up to 12.0) in the eastern United States. All reported earthquakes from these two regions were included in the analysis. [In reality, their effect on CRBRP site seismicity is minimal. Using Equation II-6, one

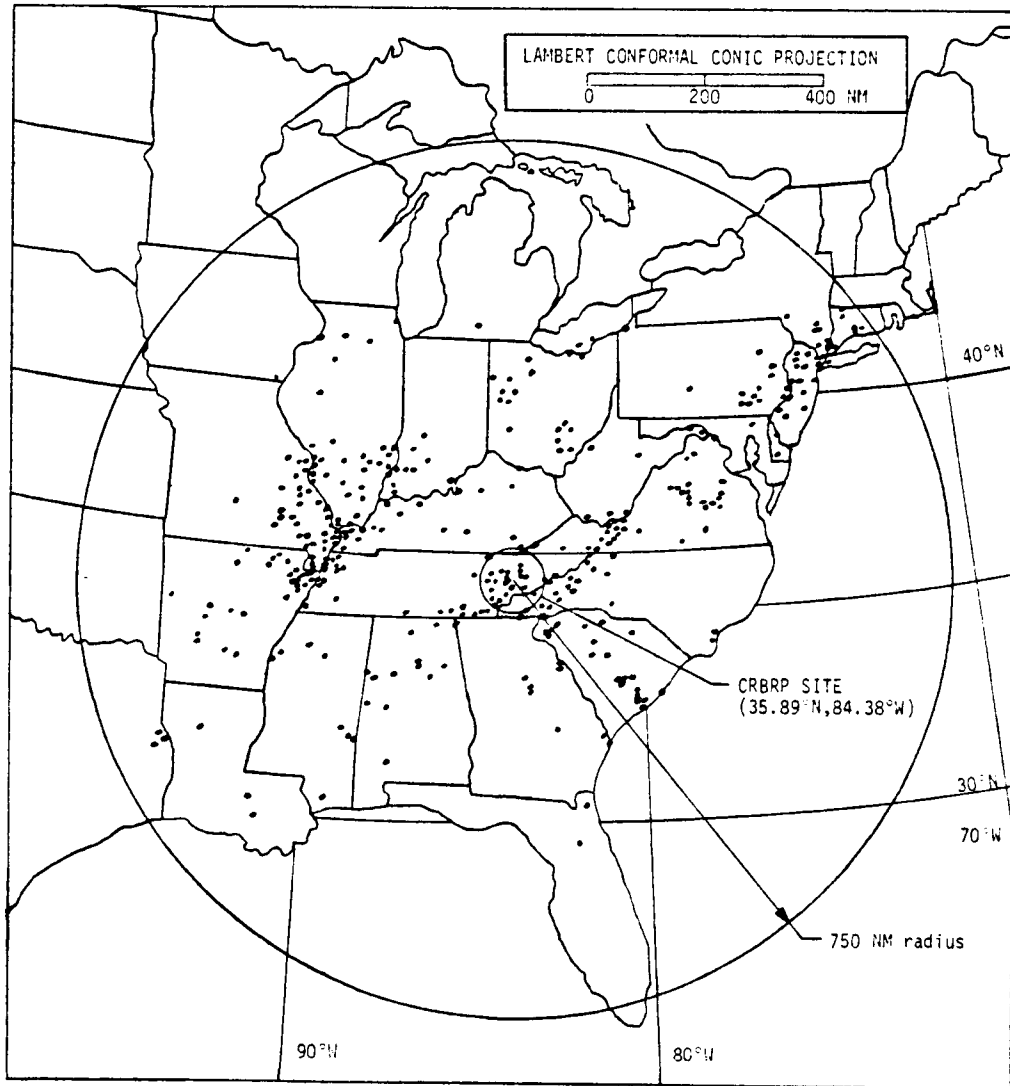


Figure II-1. Earthquake Occurrence Data.

finds that an earthquake with Modified Mercalli intensity 12.0 (Richter magnitude 9.0, body wave magnitude 7.75) located 300 nm (560 km) from the site results in a peak ground acceleration of 47.7 cm/s^2 (0.047 g), which is well below the CRBRP safe shutdown earthquake (SSE) of 0.25 g and the operating basis earthquake (OBE) of 0.125 g.]

II.3 RESULTS

A plot of site seismicity as a function of peak ground acceleration is given in Figure II-2. The annual frequency of earthquakes with which the CRBRP site acceleration exceeds the OBE of 0.125 g is $2.82 \times 10^{-4}/\text{yr}$; the exceedance frequency corresponding to the SSE of 0.25 g is $2.85 \times 10^{-5}/\text{yr}$.

In order to verify the accuracy of the analysis, the results of this study were compared with the CRBRP Safety Study.¹² Table II-1 shows the results of this comparison. In general, the two studies favorably compare, particularly for accelerations that are between the OBE (0.125 g) and the SSE (0.25 g).

Table II-1
COMPARISON OF CRBRP SITE SEISMICITY

Range of Acceleration	This Study	CRBRP Safety Study ¹²
0.05 g to 0.15 g	2.3×10^{-3}	1.4×10^{-3}
0.15 g to 0.35 g	1.6×10^{-4}	1.5×10^{-4}
$\geq 0.35 \text{ g}$	1.4×10^{-5}	3.5×10^{-5}

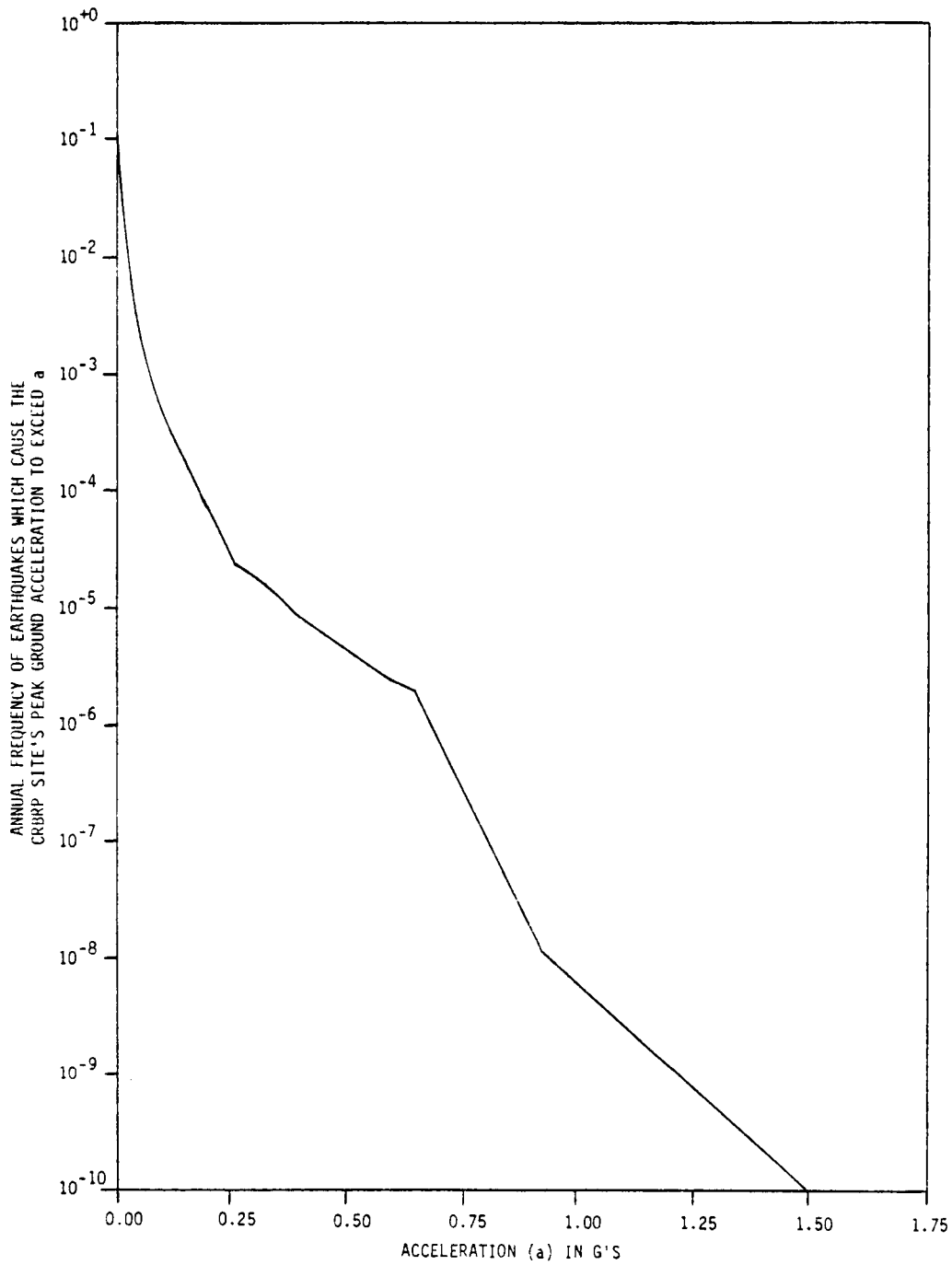


Figure II-2. CRBRP Site Seismicity.

CHAPTER III

COMPONENT FRAGILITIES

The fragility of a component is defined as the probability that the component fails to perform its intended function given the occurrence of an earthquake that causes a specified ground acceleration at the CRBRP site. That is, component fragility is the component's failure probability as a function of site acceleration. The following sections discuss the methods used to obtain component fragilities and summarize the results obtained by application of these methods to the CRBRP components.

III.1 METHOD

Component fragility depends on the component's design, method of installation (support, shock absorbers, etc.), location within the nuclear island, and the response of the nuclear island structures to an earthquake. In general, these dependencies may be summarized by a component's reserve seismic margin (the factor of safety that exists for only seismically induced loads). The reserve seismic margin is the ratio of the ground acceleration that is calculated to cause component failure to the SSE acceleration (which is 0.25 g for the CRBRP). (The SSE is used during component design to maintain acceptable stress limits in the component's materials.) For example, if a component's reserve seismic margin is 2.0, then any earthquake that causes the site ground acceleration to reach or exceed 0.5 g will

fail the component. The failure acceleration of the component (0.5 g in the above example) is termed the "acceleration capacity" of the component.

However, the reserve seismic margin (and hence the acceleration capacity) is subject to uncertainty. A step function fragility (suggested by the above discussion), where the component's failure probability "steps" from zero to unity at the acceleration capacity, cannot adequately model the component's seismically induced failure behavior. Consistent with the PRA Procedures Guide,¹ it is assumed that component fragility curves are lognormal in shape. That is:

$$P(a) = \phi \left[\frac{\ln (a/\overset{v}{A})}{\beta_C} \right] \quad (\text{III-1})$$

where:

$P(a)$ = component fragility as a function of acceleration

a = site acceleration due to an earthquake

$\overset{v}{A}$ = median acceleration capacity [i.e., when $a = \overset{v}{A}$, then

$$P(a) = 0.5]$$

β_C = combined logarithmic standard deviation (which represents the inherent randomness of the capacity about the median and uncertainty in the median capacity)

$$\phi [X] = \frac{1}{\sqrt{2\pi}} \int_{-\infty}^x \exp [-t^2/2] dt$$

The assumption that the fragility curves have a lognormal shape is reasonable since the variation of many material properties and seismic response variables can be represented by this distribution. Also, this assumption is consistent with the Central Limit Theorem since the fragilities result from the products of the distributions of a number of variables. It should be noted that:

$$1. \lim_{a \rightarrow 0} P(a) = 0$$

That is, a component is not likely to fail if the ground acceleration is small.

$$2. \lim_{a \rightarrow \infty} P(a) = 1$$

That is, a component is very likely to fail if the ground acceleration is large.

$$3. \text{ As } \beta_c \rightarrow 0, P(a) \text{ approaches a step function centered about } \bar{A}^v.$$

In summary, fragilities for CRBRP components were specified by determining a median acceleration capacity ($\bar{A}^v$) and a corresponding logarithmic standard deviation (β_c). The median acceleration capacity ($\bar{A}^v$) was taken to be the product of the reserve seismic margin times the SSE acceleration of 0.25 g.

III.2 DATA

Component fragility parameters ($\bar{A}^v$ and β_c) for most components were based on previous studies and analyses. All of the components under consideration fall into one of three categories:

1. CRBRP-specific components whose reserve seismic margins had been previously determined by either the CRBRP Project Office¹⁴ or Westinghouse Electric Corporation.¹⁵
2. Components that are similar in function and design to light-water reactor (LWR) components.
3. CRBRP-specific components that have no LWR counterparts and whose reserve seismic margins had not been determined.

The origin of fragility parameters for each category is discussed below.

III.2.1 Category 1: CRBRP-Specific Components

The CRBRP has two diverse decay heat removal paths:

1. Decay heat is transported from the reactor core to the steam generating system using the primary and intermediate heat transport systems. From the steam generating system, decay heat is transported to either the Clinch River using the turbine bypass and heat rejection systems or to the atmosphere using the steam generator auxiliary heat removal system. This decay heat removal path is hereafter referred to as "SGS cooling."
2. Decay heat is transported from the reactor core to the atmosphere using the direct heat removal service. This decay heat removal path is hereafter referred to as "DHRS cooling."

A leak in the primary heat transport system piping may fail both, either, or none of these decay heat removal paths (depending on

the leak's location). The CRBRP risk assessment¹³ related specific sections of the primary heat transport system piping to the SGS and DHRS cooling paths (i.e., the occurrence of a leak in a section of primary heat transport system piping related to the DHRS cooling path would preclude use of the DHRS cooling path). Accordingly, several reserve seismic margins were determined for the primary heat transport system piping by finding the lowest reserve seismic margin of all piping sections that comprised a potential leak source by consulting the CRBRP Heat Transport System Incontainment Piping Reserve Seismic Margins¹⁴ report. Appendix A (Table A-1) summarizes the origins of the heat transport system piping reserve seismic margins.

The reserve seismic margin for the reactor vessel guard vessel was calculated from information supplied by O'Donnell and Associates.¹⁶ As discussed in Section III.1, reserve seismic margin (RSM) is defined as the ratio of the median acceleration capacity (A^v) to the SSE acceleration (0.25g). It is related to the seismic-only margin (SOM), conservatism margin (CM), ASME code stress criteria (S), seismic-only stress (σ_{CS}), and nonseismic stress (σ_{CN}) as follows:

$$\begin{aligned} \text{RSM} &= \text{CM} \times \text{SOM} \\ &= \text{CM} \times \frac{S - \sigma_{CN}}{\sigma_{CS}} \end{aligned} \quad (\text{III-2})$$

where:

CM = conservatism margin (1.45 for CRBRP components¹⁵),

S = ASME code stress criteria

σ_{cn} = calculated stress at the SSE due to all loads except seismic loads, and

σ_{cs} = calculated stress due to SSE loads.

If k is defined as the ratio of seismic-only stress to nonseismic stress ($k = \sigma_{cs}/\sigma_{cn}$), then

$$RSM = CM \frac{SM}{k} + 1$$

where:

SM = safety margin at the SSE

$$= S/(\sigma_{cn} + \sigma_{cs}) - 1$$

Appendix A (Table A-2) lists the reserve seismic margin and median acceleration capacity for various reactor vessel guard vessel parts as determined by the above relation. These margins include consideration of the fluid-structure interaction when the annulus between the reactor vessel and its guard vessel is filled with sodium.

A review of Table A-2 shows that the most fragile part of the guard vessel which threatens integrity is the inlet nozzle and elbow. (It is assumed that failure of the support skirt bolts would not threaten guard vessel integrity.) Thus, the median capacity for the reactor vessel guard vessel (A) was taken to be 0.40 g. Stress reports concerning the loop guard vessels (PHTS pump and intermediate heat exchanger guard vessels) were not available; hence, it was assumed that the median capacities of all loop guard vessels were 0.40 g.

The reserve seismic margins for most other CRBRP-specific components were obtained from the Clinch River Breeder Reactor Plant Reserve Seismic Margins study.¹⁵ Following discussions with the CRBRP Project Office¹⁷, the reserve seismic margin for all plant buildings was estimated to be 4.0, which corresponds to a median acceleration capacity of 1.0 g. (Reference 14 gives a minimum reserve seismic margin of 2.49 for buildings and structures.) The reserve seismic margin given in the Clinch River Breeder Reactor Reserve Seismic Margins study¹⁵ for the Sodium-Water Reaction Product Relief System (SWRPRS) rupture disk assemblies only addressed buckling of the upstream disk. Since both the upstream and downstream disks must rupture in order to fail a specified SGS loop, the fragility of both disks was assumed to be the square of the fragility of the upstream disk for accelerations less than the median acceleration capacity of the upstream disk. For accelerations greater than or equal to the median acceleration capacity of the upstream disk, the fragility of both disks was assumed to be equal to the fragility of the upstream disk. (This model of SWRPRS disk fragility is based on the assumption that seismically induced SWRPRS disk failures are most likely due to the impact of an IHTS sodium slug into the disks rather than differential motion between the disks and knife edges. At low accelerations, a majority of an IHTS sodium slug's momentum would be dissipated by deformation and rupture of the upstream disk; at high accelerations, the IHTS sodium slug would have sufficient momentum to rupture both disks.)

Neither of the documents consulted provided any quantitative assessment of the uncertainty in the reported reserve seismic margins. Discussions with the author of the Clinch River Breeder Reactor Reserve Seismic Margins study¹⁵ indicated that the reported margins were "best estimates"; hence, the median acceleration capacity (A) for each component was calculated by multiplying the reported reserve seismic margin by the SSE acceleration (0.25 g). The standard deviation (β_C) for each component was estimated using data contained in LWR risk assessments supplemented with engineering judgment.

During the assessment of component fragilities, it was fully acknowledged that CRBRP components (e.g., sodium piping) are unique designs; however, the methods used to assess their seismic responses (response spectrum analysis and time history analysis) are the same methods used to assess LWR component seismic response. In addition, the uncertainties in material properties (such as yield strength) for CRBRP components are approximately the same as for LWR components since the same methods are used to determine both CRBRP and LWR component material properties. Thus, the combined uncertainties (β_C) for CRBRP components are reasonably estimated by analysis of LWR data.

Four LWR probabilistic risk assessments (Zion,¹⁸ Indian Point,¹⁹ Oconee,²⁰ and McGuire²¹) were reviewed to determine logarithmic standard deviation (β_C) values for liquid metal piping, liquid metal heat exchangers, control rods, SGS modules, liquid metal

expansion tanks, and buildings. Logarithmic standard deviations for the above CRBRP components were taken to be the averages of the LWR values reported for similar equipment. Appendix A (Tables A-3 through A-8) lists the LWR data, average value of the logarithmic standard deviation, minimum reported value, and maximum reported value. The logarithmic standard deviations for the SGS module tubes, loop and reactor vessel guard vessels, and the SWRPRS disks were taken to be 0.50. (No relevant LWR data was found for these components; the value of 0.50 is representative of the logarithmic standard deviation for most components).

III.2.2 Category 2: LWR Analogous Components

The fragility parameters for CRBRP components that are similar in function and design to LWR components were based on fragility studies performed in four previous PRAs. The four PRAs consulted were:

1. Zion¹⁸ (SSE = 0.17 g),
2. Indian Point¹⁹ (SSE = 0.15 g),
3. Oconee²⁰ (SSE = 0.1 g), and
4. McGuire²¹ (SSE = 0.15 g).

Appendix A (Table A-9) lists the fragility parameters ($\bar{A}^v$ and β_C) reported in these PRAs for components which were similar to CRBRP components.

The four median acceleration capacities ($\bar{A}^v$) were divided by the SSE accelerations (given above) to determine four reserve seismic

margins for each component. Next, the median reserve seismic margins for the CRBRP components were taken to be the average of these four derived reserve seismic margins. Finally, the median acceleration capacities (A^V) for the CRBRP components were obtained by multiplying their median reserve seismic margins by the SSE acceleration for the CRBRP (0.25 g). The logarithmic standard deviations (β_C) for the CRBRP components were taken to be the average of the four reported values.

Occasionally, no LWR data could be found for analogous CRBRP components, e.g., valves, piping, etc. A review of the four PRAs showed that these components are not fragile (i.e., they have median acceleration capacities above 2.0 g) and hence were not significant contributors to seismic risk. These strong components were removed from the CRBRP seismic model.

III.2.3 Category 3: CRBRP Components That Have No LWR Counterparts and Have Not Been Analyzed

No direct data pertaining to seismic fragility was found for the following systems:

1. Recirculating Gas Cooling System (RGCS), and
2. Steam Generator Auxiliary Heat Removal System (Protected Air-Cooled Condensers).

It may be surmised that these systems are not fragile; otherwise, they would have been analyzed in detail. The CRBRP Reserve Seismic Margin report¹⁵ only analyzed two shutdown heat removal systems: normal (which was limited by the SWRPRS rupture disks) and

backup (DHRS components). This suggests that the PACCs were examined and found to be stronger than the SWRPRS rupture disks. Hence, seismically induced PACC failures were assumed to be insignificant contributors to the overall CRBRP seismic risk.

The recirculating gas cooling system (RGCS) provides cooling to the electromagnetic pumps in the DHRS. Major components of the RGCS are its fans and coolers. These components are similar in design and location to LWR reactor building ventilation components. The Ocone PRA²⁰ provided fragility parameters for coolers; it concluded that the fans are not fragile. Hence, it was assumed that the RGCS fragility parameters were about the same as the values reported for the Ocone reactor building ventilation system coolers.

III.3 RESULTS

Component fragility parameters (A^v and β_c) were determined for each component specified in the core-damage event tree top logic and the system level fault trees. Table III-1 summarizes the values obtained.

The following list tabulates the assumptions made while determining CRBRP component fragility parameters:

1. All parameters are based on previous studies and analyses.
2. Reserve seismic margins obtained from the CRBRP Heat Transport System Incontainment Piping Reserve Seismic Margins¹⁴ and the CRBRP Reserve Seismic Margins¹⁵ reports

Table III-1

COMPONENT FRAGILITY PARAMETERS

System/Component	V_A	B_C
<u>Auxiliary Feedwater</u>		
Motor-driven pumps	12.07	0.49
Protected water storage tank	2.02	0.59
Turbine-driven pumps	11.96	0.47
<u>Buildings</u>		
All buildings	1.00	0.43
<u>Direct Heat Removal Service</u>		
Air blast heat exchangers	0.72	0.54
Electromagnetic pumps	2.30	0.63
Ex-vessel storage tank coolers	0.93	0.54
NaK expansion tank	0.98	0.53
Overflow heat exchanger	2.11	0.54
Piping	0.62	0.63
<u>Electric Power</u>		
Batteries and racks	2.12	0.60
Cable trays	5.32	0.65
Ceramic insulators	0.20	0.32
DC buswork	1.43	0.59
Diesel generators	2.33	0.56
Diesel generator clay tanks	3.64	0.45
Distribution panels	1.43	0.59
Inverters	5.34	0.48
Switchgear (480 V)	1.79	0.59
Switchgear (4160 V and above)	1.81	0.84
Transformers	1.10	0.64

Table III-1 (Continued)

System/Component	V A	B _C
<u>Emergency Chilled Water</u>		
ECW chillers	8.41	0.42
ECW pumps	2.19	0.51
<u>Emergency Plant Service Water</u>		
EPSW pumps	2.19	0.51
<u>Heat Rejection</u>		
Circulating water pumps	2.19	0.51
<u>Heat Transport</u>		
Guard vessel leak	0.40	0.50
IHTS leak	1.93	0.63
IHTS pumps	3.91	0.52
PHTS leak inside a loop guard vessel	0.69	0.63
PHTS leak inside the reactor vessel		
guard vessel	1.43	0.63
PHTS leak that disables SGS and DHRS		
heat removal	0.98	0.63
PHTS leak that disables SGS heat removal		
but allows DHRS operation	1.57	0.63
PHTS leak that disables DHRS heat		
removal but allows SGS heat removal	0.64	0.63
PHTS leak that allows SGS and DHRS		
heat removal	1.93	0.63
PHTS pumps	3.91	0.52
<u>Main Feedwater and Condensate</u>		
Main condensate pumps	6.61	0.57
Main feedwater pumps	6.61	0.57

Table III-1 (Continued)

System/Component	V A	β_c
<u>Normal Chilled Water</u>		
NCW chillers	8.41	0.42
NCW pumps	2.19	0.51
<u>Normal Plant Service Water</u>		
NPSW pumps	2.19	0.51
<u>Reactor Protection</u>		
Control rods (binding)	1.17	0.47
SGAHRs actuation logic	4.58	0.78
<u>Recirculating Gas Cooling</u>		
RGCS coolers	2.50	0.55
<u>Secondary Services Closed Cooling Water</u>		
SSCCW heat exchangers	8.41	0.42
SSCCW pumps	2.19	0.51
<u>Steam Generating System</u>		
Relief/safety valves (FTSD)	9.95	0.75
SGS module leak	1.03	0.52
SGS tube rupture	3.51	0.50
Steam piping	4.52	0.66
SWRPRS rupture disks	0.57	0.50

were assumed to be median values. Logarithmic standard deviations (β_{CS}) were estimated using LWR data.

3. The median reserve seismic margins for CRBRP components that have LWR counterparts were taken to be the average values reported in four PRAs (Zion, Indian Point, Oconee, and McGuire). The logarithmic standard deviations (β_{CS}) for these components were taken to be the average value reported in the four PRAs.
4. The PACCs are assumed to be strong (i.e., not fragile) based on discussions in the CRBRP Reserve Seismic Margins¹⁵ report.
5. The RGCS fragility parameters were taken to be the same as the Oconee reactor building ventilation system coolers.

CHAPTER IV

DEVELOPMENT OF CORE-DAMAGE SEQUENCES FOR SEISMIC EVENTS

Core-damage sequences resulting from seismic events were developed using the core-damage event tree and system-level fault trees used to develop core-damage sequences for internal initiators in the CRBRP risk assessment.¹³ The internal initiator core-damage logic model (event tree, top logic, and system-level fault trees) was modified by adding seismically induced failures where appropriate. This approach was selected for two reasons:

1. Insights about plant behavior and system interactions during accidents gained during the development of the internal sequence logic model were retained. In addition to expediting the analysis process, results of the seismic analysis were directly comparable to the internal initiator analysis results presented in the CRBRP risk assessment since both analyses are based on the same logic model.
2. The possibility of equipment failure due to nonseismic causes during a seismic event was included in the seismic logic model, as suggested by the PRA Procedures Guide.¹

The first step in the development of the seismic logic model was the identification of seismically induced common-cause failures. An earthquake may cause multiple component failures in two ways:

1. Building walls and roofs may spall such that the resulting debris crushes adjacent components, and

2. Components that share a common support may fail due to "coupling" effects.

The first type of common-cause failure is easy to conceive; however, the second type is not obvious. A group of components is considered to be coupled if the failure of any component in the group implies the failure of all components in the group. Recently completed works by the Seismic Safety Margins Research Program (SSMRP) imply that all plant components are somewhat coupled due to the common ground shaking which occurs during an earthquake.^{22,23} The PRA Procedures Guide¹ recommends that all similar components which share a common elevation and orientation (e.g., north-south) be coupled. In this study, a base case seismic logic model was defined according to the PRA Procedures Guide¹ recommendations. In addition, other possible coupling combinations were identified to assess the sensitivity of the final results to the coupling assumptions. Appendix B (Table B-1) lists the components which were coupled in the base case seismic logic model.

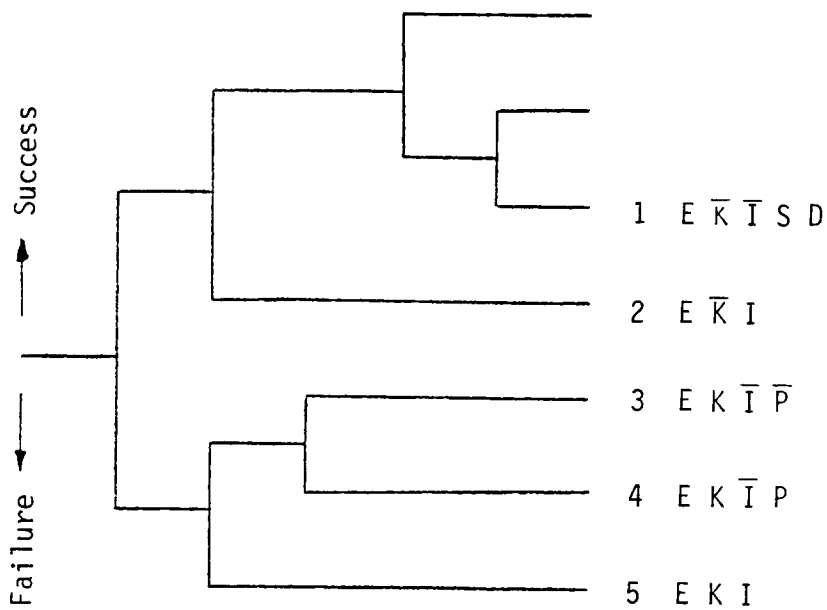
The second step in the development of the seismic logic model consisted of adding seismically induced failure events to the internal initiator logic model. This step was accomplished by modifying basic events in the top logic and fault trees that were prone to seismic failure with the appropriate events from Table III-1. For example, the event "SMPUMPA53," which represents hardware failures of the train A motor-driven auxiliary feedwater pump, was replaced with an "OR" gate with inputs "SMPUMPA53" and "MDAFWPS." The event

"MDAFWFS" represents the seismic failure of both motor-driven auxiliary feedwater pumps. Appendix B lists the modifications made to the top logic and system-level fault trees by this process (Table B-2) and all basic events added during development of the seismic logic model (Table B-3).

The third step in the development of the seismic logic model consisted of logic reduction and simplification. Two events contained in the internal initiator core-damage event tree were eliminated by this step. Event Q (initial reactor power normal) was removed since the addition of positive reactivity during an earthquake was taken to be probabilistically insignificant. [Neither WARD-D-0332¹⁵ nor any of the LWR PRAs¹⁸⁻²¹ included this event.] Event C (long-term cooling via the SGS) was removed since preliminary seismic sequence solution indicated that seismic failures of event C also implied the failure of event S (short-term cooling via the SGS). Figure IV-1 illustrates the seismic core-damage event tree.

The top logic for events I (maintenance of PHTS sodium inventory) and P (PHTS main pump trip) required modification to enable treatment of seismic events. The failure event "Failure to trip all PHTS main pump motors" (which is the top event of the event P top logic, and also appears in the event I top logic) is treated as a basic event in the internal initiator logic model (named "PHTS"). However, this failure event can only occur if (1) any pump fails to trip on demand and (2) the main PHTS pumps and their support systems (e.g., electric power) are not failed by the earthquake. Hence, new top logic was developed for event P (Figure IV-2).

E	K	I	P	S	D	SEQUENCE
---	---	---	---	---	---	----------



- E -- Earthquake
- K -- Reactor shutdown
- I -- Sodium inventory maintained
- P -- Main PHTS pump motors tripped
- S -- SGS cooling maintained
- D -- DHRS cooling maintained

Figure IV-1. Seismic Core-Damage Event Tree.

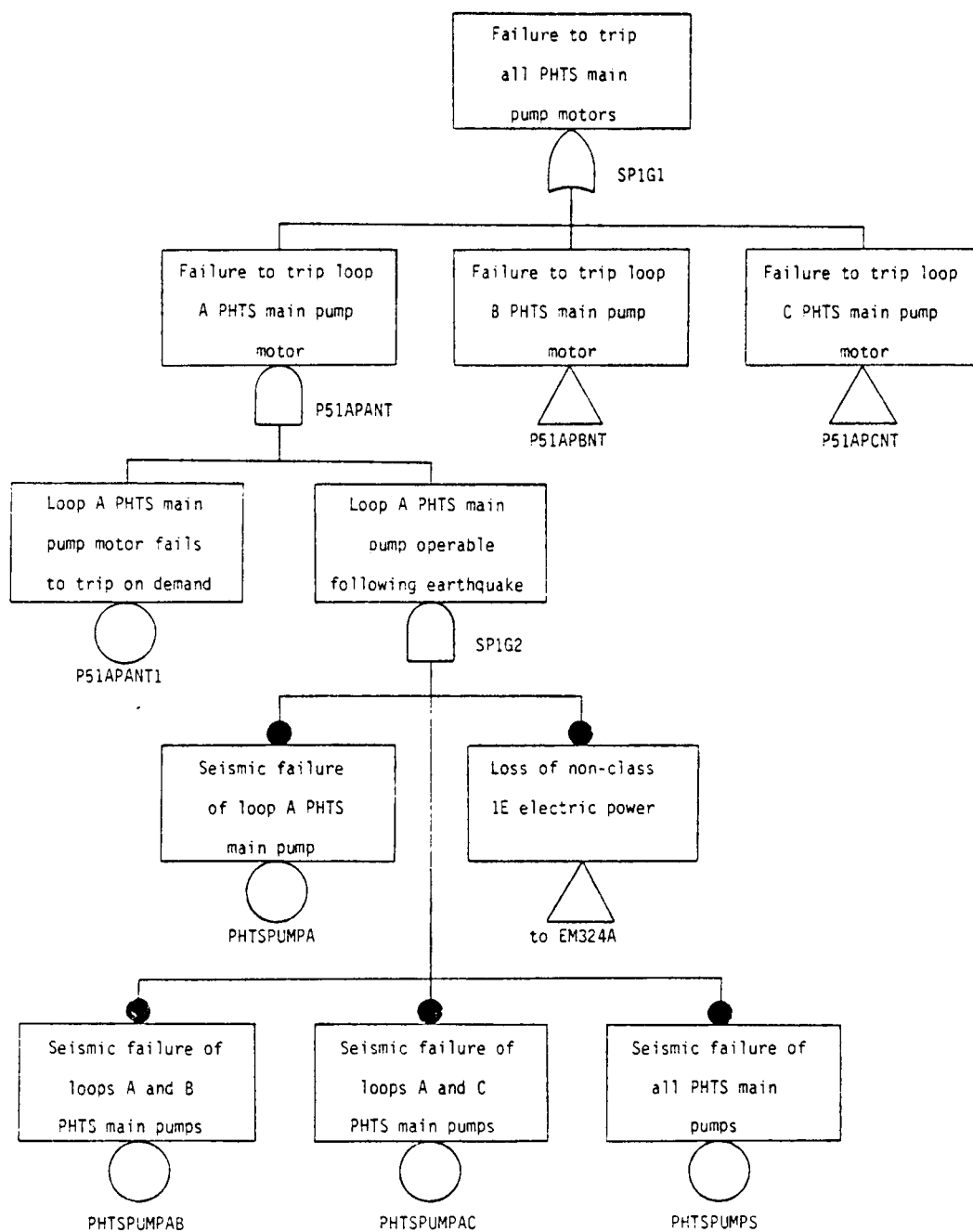


Figure IV-2. Seismic Event P Top Logic.

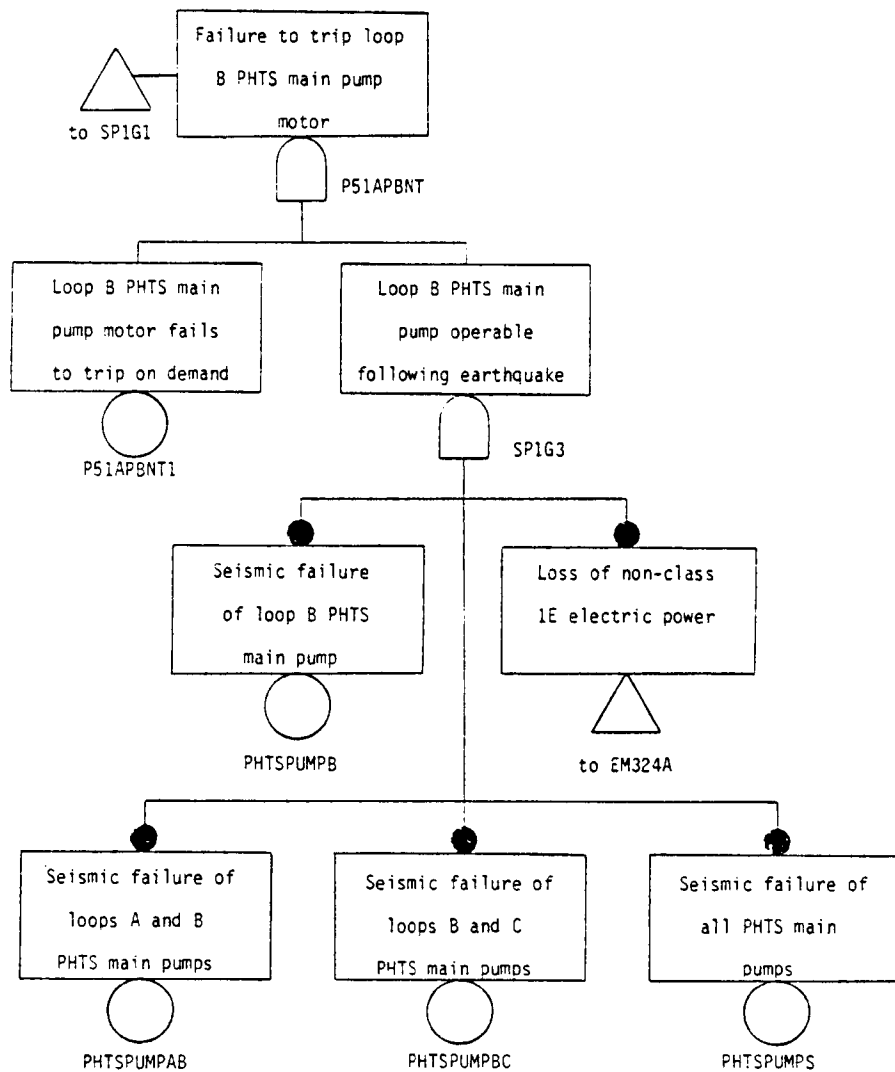


Figure IV-2. (continued).

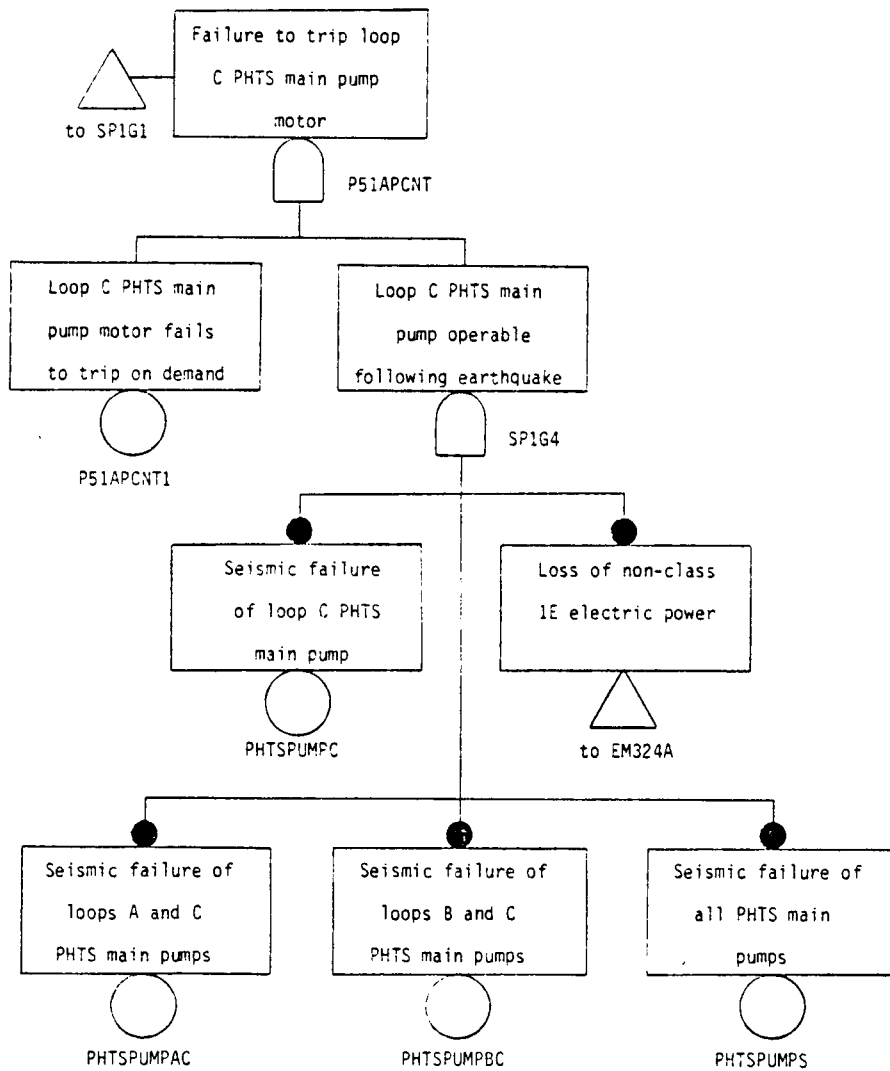


Figure IV-2. (continued).

CHAPTER V

EVALUATION OF CORE-DAMAGE SEQUENCES FOR SEISMIC EVENTS

The solution of a seismic core-damage sequence consists of two parts: a list of sequence prime implicants and the sequence occurrence frequency. Information necessary for sequence solution (site seismicity, component fragility, and the seismic logic model) has been previously discussed (Chapters II, III, and IV, respectively). The following section describes the methods used in the solution process.

V.1 SEISMIC SEQUENCE PRIME IMPLICANTS

A prime implicant of a seismic sequence is a collection of events (failures) or their complements (success) that are necessary and sufficient to cause occurrence of the sequence. That is, the prime implicants of a sequence are the roots of the sequence Boolean equation defined by the seismic logic model. (The term "prime implicant" is used rather than "minimal cut set" to reflect that the roots of the Boolean equation may contain complemented events).

The prime implicants for all sequences were determined using the SETS²⁴ computer program. Because of the complexity of these sequences, only prime implicants with occurrence frequencies greater than or equal to 1.0×10^{-7} per year were determined. The presence of complemented events in the logic model necessitated several SETS runs to ensure that all prime implicants with frequencies above 1.0×10^{-7} per year were determined. At low accelerations, component

fragilities (failure probabilities) approach zero; the probabilities of their complements approach unity. Alternatively, component fragilities approach unity at high accelerations. Thus, a prime implicant that contains both seismic events and complements would not be found if the failure probabilities supplied to SETS correspond to either a low or high acceleration. For example, suppose that inspection of the logic model shows one prime implicant to be A^*/B , where A and B are seismic events and $/B$ denotes the complement of B . The occurrence frequency of this prime implicant is given by:

$$f_{A^*/B}(a) = f_E(a)Pr\{A(a)\}[1-Pr\{B(a)\}]$$

where:

$f_{A^*/B}(a)$ = occurrence frequency of the prime implicant A^*/B at acceleration a

$f_E(a)$ = seismicity (earthquake occurrence frequency) at acceleration a

$Pr\{A(a)\}$ = fragility (failure probability) of event A at acceleration a

$Pr\{B(a)\}$ = fragility (failure probability) of event B at acceleration a

If $\bar{A}$ and B_c values are (3.6, 0.92) for event A and (3.6, 0.92) for event B and the truncation value given SETS is 1.0×10^{-7} , then the prime implicant A^*/B would not be reported for accelerations of 0.05 g and 0.80 g but would be reported for an acceleration of 0.25 g as shown in Table V-1 below.

Table V-1

SELECTION OF SETS TRUNCATION VALUES

Acceleration in g's	$f_E(a)$	$Pr\{A(a)\}$	$Pr\{B(a)\}$	$f_{A^*/B}(a)$
0.05	5.7-3	1.7-6	1.7-6	9.7-9
0.25	5.7-5	2.0-3	2.0-3	1.1-7
0.80	1.8-6	5.2-2	5.2-2	8.9-8

Hence, SETS runs were performed for several accelerations; the resulting prime implicants were then combined into one list.

V.2 SEISMIC SEQUENCES QUANTIFICATION

The objective of the seismic sequence quantification task was to generate an exceedance frequency curve for each sequence. The exceedance frequency curve is a plot of the sequence's occurrence frequency given an earthquake with acceleration greater than or equal to a specified value as a function of acceleration. This task was accomplished by discretizing the seismicity curve, calculating the occurrence probability of the sequence for various values of acceleration corresponding to the discretized seismicity curve, multiplying the sequence occurrence probability by the discretized seismicity to obtain the sequence occurrence frequency at fixed values of acceleration, and summing the sequence occurrence

frequencies at fixed accelerations to generate the sequence exceedance frequency.

The results of the site seismicity analysis (Chapter II) are presented as an exceedance frequency curve. That is,

$$\begin{aligned} F_E(a) &= \text{seismicity at acceleration } a, \\ &= \text{number of earthquakes per year that cause the CRBRP site} \\ &\quad \text{ground acceleration to exceed } a. \end{aligned}$$

Then

$$F_E(a_1) - F_E(a_2) = \text{number of earthquakes per year that cause the CRBRP site ground acceleration to fall between the interval } (a_1, a_2).$$

All of these earthquakes are assumed to cause a site acceleration equal to a_2 , the highest value. Thus, the discrete seismicity, $f_E(a_2)$ is equal to $F_E(a_1) - F_E(a_2)$. Fourteen acceleration values were selected; their corresponding discrete seismicities are given in Table V-2. (The discretization process is equivalent to stating that if an earthquake occurs, then the CRBRP site acceleration must be one of the fourteen values given in Table V-2).

The occurrence probability of a sequence at a specified acceleration may be expressed as a function of the seismic event fragilities at the specified acceleration by use of the axioms of probability theory. A sequence is the logical union of its prime implicants; thus,

$$\Pr\{S \mid a\} = \Pr\{I_1 + I_2 + \dots + I_n \mid a\} \quad (V-1)$$

Table V-2

CRBRP DISCRETE SEISMICITY

Acceleration in g's	Frequency in yrs ⁻¹
9.93-4	5.68-1
8.27-3	1.79-1
1.87-2	5.68-2
2.99-2	1.79-2
4.92-2	5.68-3
7.91-2	1.79-3
1.28-1	5.68-4
1.93-1	1.79-4
2.55-1	5.68-5
4.05-1	1.79-5
5.98-1	5.68-6
8.10-1	1.79-6
8.85-1	5.68-7
9.09-1	1.79-7

where I_i ($i = 1, 2, \dots, n$) represents one of n prime implicants for sequence S , and the symbol "+" denotes logical union. The sequence occurrence frequency may be approximated by:

$$\Pr\{S \mid a\} < 1 - \prod_{i=1}^n [1 - \Pr\{I_i \mid a\}] \quad (V-2)$$

Thus, given $f_E(a)$ and $\Pr\{S \mid a\}$ values, the exceedance frequency is given by:

$$f_S(A \geq a_i) = \sum_{j=i}^{14} f_E(a_j) \Pr\{S \mid a_j\} \quad (V-3)$$

All of the above computations were performed using the ESA²⁵ computer program.

CHAPTER VI

RESULTS

The frequencies of core-damage sequences due to earthquakes, calculated using the data and information in Chapters II through V, are presented in this section. Factors affecting the results are also discussed.

VI.1 QUALITATIVE RESULTS

Of the five seismic sequences that lead to core damage, four have prime implicants whose occurrence frequencies are greater than or equal to $1.0 \times 10^{-7}/\text{yr}$. Tables VI-1 through VI-4 summarize the prime implicants found for sequences 1 (EKISD), 2 ($\overline{\text{EKI}}$), 3 (EKIP), and 5 (EKI) using the base case seismic logic model as defined in Appendix B, Table B-1. All prime implicants for sequence 4 (EKIP) have frequencies less than $1.0 \times 10^{-7}/\text{yr}$. (Sequence EKIP is extremely unlikely since offsite power, which must remain intact to fail event P, is very fragile compared to the control rods. That is, failure to trip the reactor due to control rod mechanical binding will most likely be accomplished by a loss of offsite power.)

The major causes of SGS cooling mission failure (event S) are SWRPRS rupture disk failure and SGAHRS actuation logic failure. The DHRS cooling mission (event D) dominant failures include air blast heat exchanger failures, PHTS leaks, DHRS piping leaks, and loss of electrical power to the DHRS pumps. Electrical power to the DHRS

Table VI-1

SEQUENCE 1 (EKTSD) PRIME IMPLICANTS (BASE CASE)

-
-
1. (T8EVAPS + T8SH) * (RMPPIPE + ABHXS + EB416NS794 + RSB + RCB + EEB)
* /EVENTK

Following an earthquake, SGS cooling is lost due to the actuation of SWRPRS in all three SGS loops caused by the seismically induced failure of (1) the evaporator rupture disks (T8EVAPS), or (2) the superheater rupture disks (T8SH). DHRS cooling is also lost due to seismically induced (1) DHRS piping leaks (RMPPIPE), (2) failure of both air-blast heat exchangers (ABHXS), (3) loss of power to all DHRS components, or (4) missile impact with DHRS components located in the reactor service building (RSB). Power to all DHRS components may be lost by (1) missile impact with buses 32A and 32B which are located in the reactor containment building (RCB), (2) missile impact with buses 3A and 3B which are located in the electrical equipment building (EEB), or (3) the common-mode failure of buses 3A and 3B which are both located at elevation 794 feet in a north-south orientation (EB416NS794).

2. (T8EVAPS + T8SH) * NPSWPUMPS * /EVENTK

Following an earthquake, SGS cooling is lost due to the actuation of SWRPRS in all three SGS loops caused by the seismically induced failure of (1) the evaporator rupture disks (T8EVAPS), or (2) the superheater rupture disks (T8SH). DHRS cooling is also lost due to loss of cooling to the DHRS makeup pumps caused by the seismically induced failure of the normal plant service water pumps (NPSWPUMPS). (The DHRS makeup pumps are cooled by the recirculating gas cooling system whose heat sink is the chilled water system. In turn, the chilled water system's heat sink is the plant service water system. The systems are configured such that a loss of normal chilled water flow or a loss of offsite power will start both the emergency chilled water and emergency plant service water systems. The seismically induced failure of the normal plant service water pumps results in a loss of heat sink to the normal chilled water system and, hence, to the recirculating gas cooling system; the emergency plant service water and chilled water systems are not automatically placed into service since the earthquake does not cause a loss of normal chilled water flow or a loss of offsite power.)

Table VI-1 (Continued)

3. NPSWPUMPS * LOGICS * /EVENTK

Following an earthquake, SGS cooling using the main feedwater and turbine bypass systems is lost due to the seismically induced failure of the normal plant service water pumps (NPSWPUMPS), which provide the heat sink for the secondary services closed cooling water system (supplies main feedwater pump cooling). In addition, SGS cooling using the SGAHRS is lost due to the seismically induced failure of the SGAHRS actuation logic panels. DHRS cooling is lost due to the failure of the normal plant service water pumps, as discussed in (2) above.

4. (T8EVAPS + T8SH) * (T21AS + T21BS + T21CS) * [(/P51APANT * /P51APBNT * /P51APCNT) + T4 + EB161S] * /EVENTK

Following an earthquake, SGS cooling is lost due to the actuation of SWRPRS in all three SGS loops caused by the seismically induced failure of (1) the evaporator rupture disks (T8EVAPS), or (2) the superheater rupture disks (T8SH). DHRS cooling is lost due to a PHTS leak that lowers the primary sodium level below the minimum level needed to operate the DHRS (T21AS, T21BS, T21CS). Note that an extended loss of primary sodium does not occur since either (1) the main PHTS pumps are shut down following reactor trip (/P51APANT * /P51APBNT * /P51APCNT), or (2) the earthquake fails offsite power to the plant caused by ceramic insulator failure (T4) or 161 kV switchgear failure (EB161S).

5. (T8EVAPS + T8SH) * T4 * (EB4160816 + DGB + ECWPUMPS + EPSWPUMPS + EPH) * /EVENTK

Following an earthquake, SGS cooling is lost due to the actuation of SWRPRS in all SGS loops caused by the seismically induced failure of (1) the evaporator rupture disks (T8EVAPS), or (2) the superheater rupture disks (T8SH). In addition, a total loss of station power results when offsite power is failed (T4), and all emergency power is disabled by the earthquake. Emergency power may be lost due to (1) the common-mode failure of the diesel-generator buses (EB4160816), (2) missile impact with the diesel generators (DGB), or (3) loss of diesel generator cooling systems caused by pump failures (ECWPUMPS, EPSWPUMPS) or missile impact with the pumps (EPH).

Table VI-1 (Continued)

6. (T4 + EB161S) * (EB416NS794 + EB4160816) * LOGICS * /EVENTK

Following an earthquake, offsite power is lost due to (1) ceramic insulator failure (T4) or (2) 161 kV switchgear failure (EB161S). Emergency power is also failed due to the common-mode failures of all divisions (EB4160NS794, EB4160816). Thus, SGS cooling using either the main feedwater and turbine bypass systems or the motor-driven auxiliary feedwater pumps is precluded. In addition, DHRS cooling is also precluded by the total loss of power. SGS cooling using the turbine-driven auxiliary feedwater pump is failed due to the seismically induced failure of the SGAHRS actuation panels (LOGICS).

7. T4 * CB

Following an earthquake, offsite power is lost (T4); hence, SGS cooling using the main feedwater and turbine bypass systems is precluded. In addition, missiles generated by control building damage (CB) fail the SGAHRS actuation logic panels and buses 5A and 5B (which provide 125 V dc to the diesel sequencers). Thus, both SGS cooling using SGAHRS and DHRS cooling are precluded.

8. SGB

Following an earthquake, missiles generated by steam generator building damage (SGB) fail all SGS cooling. In addition, these missiles fail portions of the emergency chilled water system needed to operate the recirculating gas cooling system; hence, DHRS makeup pump cooling is lost.

Table VI-2

SEQUENCE 2 (E $\bar{K}$ I) PRIME IMPLICANTS (BASE CASE)

1. T10AS * TIGVLEAKAS * /EVENTK

The earthquake causes a PHTS leak into a loop A guard vessel (T10AS). The loop A guard vessel's integrity is lost (TIGVLEAKAS); hence, primary sodium is lost from the PHTS. Reactor shutdown is successfully achieved (/EVENTK).

2. T10BS * TIGVLEAKBS * /EVENTK

Same as (1) above, with loops A and B interchanged.

3. T10CS * TIGVLEAKCS * /EVENTK

Same as (1) above, with loops A and C interchanged.

4. T10R * RVGVLEAKS * /EVENTK

The earthquake causes a PHTS leak into the reactor vessel guard vessel (T10R). The reactor vessel guard vessel's integrity is lost (RVGVLEAKS); hence, primary sodium is lost from the PHTS. Reactor shutdown is successfully achieved (/EVENTK).

Table VI-3

SEQUENCE 3 (E $\overline{K}$ IP) PRIME IMPLICANTS (BASE CASE)

1. T4 * EVENTK

The earthquake causes a loss of offsite power to the plant due to failure of ceramic insulators in the switchyard (T4). Reactor shutdown is not achieved (EVENTK) due to mechanical binding of the control rods.

Table VI-4

SEQUENCE 5 (EKI) PRIME IMPLICANTS (BASE CASE)

1. T1OAS * TIGVLEAKAS * EVENTK

The earthquake causes a PHTS leak into a loop A guard vessel (T1OAS). The loop A guard vessel's integrity is failed (TIGVLEAKAS); hence, primary sodium is lost from the PHTS. Reactor shutdown is not achieved (EVENTK) due to mechanical binding of the control rods.

2. T1OBS * TIGVLEAKBS * EVENTK

Same as (1) above, with loops A and B interchanged.

3. T1OCS * TIGVLEAKCS * EVENTK

Same as (1) above, with loops A and C interchanged.

4. T1OR * RVGVLEAKS * EVENTK

The earthquake causes a PHTS leak into the reactor vessel guard vessel (T1OR). The reactor vessel guard vessel's integrity is lost (RVGVLEAKS); hence, primary sodium is lost from the PHTS. Reactor shutdown is not achieved (EVENTK) due to mechanical binding of the control rods.

pumps may be lost due to switchgear failures or a loss of offsite power coupled with diesel generator failures caused by loss of emergency plant service water. Building damage will fail either the DHRS (which depends on components in the reactor service building, reactor containment building, electrical equipment building, and emergency pump house) or both SGS and DHRS cooling (both cooling missions depend on components in the steam generator building and the control building). The major sources of PHTS sodium inventory loss (event I, which appears in sequences 2 and 5) are leaks into guard vessels coupled with guard vessel failure.

It should be recalled that seismically induced failures were added to the internal initiator logic model; hence, combinations of seismically induced failures, human errors, and random hardware failures may lead to core damage. Determination of the sequence prime implicants shows that only seismically induced failures are important (i.e., appear in prime implicants with occurrence frequencies above $1.0 \times 10^{-7}/\text{yr}$).

VI.2 QUANTITATIVE RESULTS

The frequency of seismically induced core-damage accidents from all sequences was calculated to be $3.2 \times 10^{-5}/\text{yr}$. Table VI-5 shows the contribution of each sequence to this total. Figures VI-1 through VI-4 show the exceedance frequency curves for the four sequences with prime implicant occurrence frequencies above $1.0 \times 10^{-7}/\text{yr}$ (sequences 1, 2, 3, and 5).

Table VI-5

SEISMIC SEQUENCE FREQUENCIES

Number	Sequence	Bin	Frequency (yr^{-1})
1	EKISD	LHSE	$1.5-5^a$
2	EKI	PLOS	$1.4-5$
3	EKIP	ULOF	$1.3-6$
4	EKIP	ULHS	ϵ^b
5	EKI	ULOS	$1.5-6$
TOTAL			$3.2-5$

^aThe notation "1.5-5" represents the value 1.5×10^{-5} .

^bThe notation " ϵ " is used to indicate that the occurrence frequency of the sequence is greater than zero but less than 1.0×10^{-7} per year.

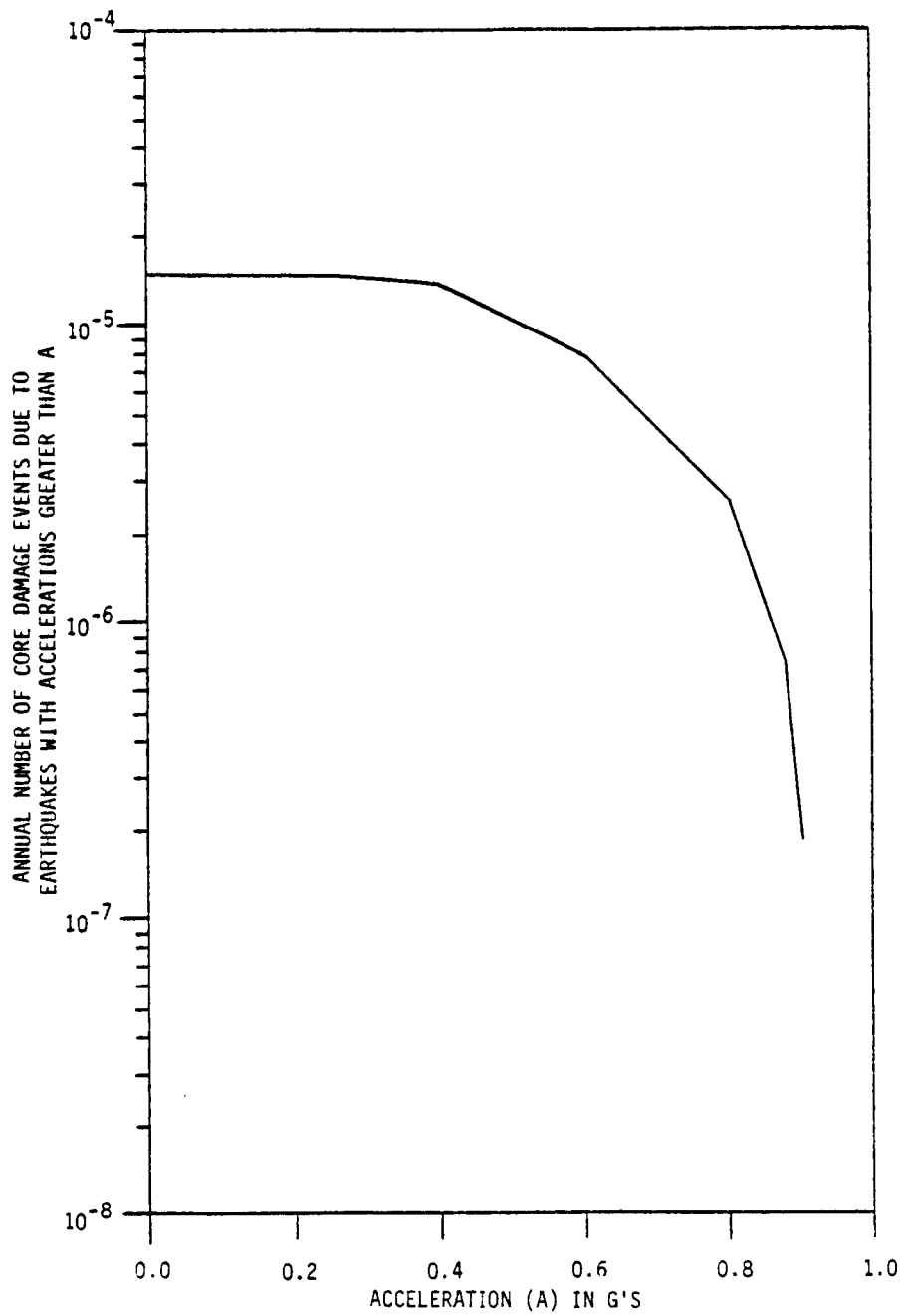


Figure VI-1. Exceedance Frequency Curve for Sequence 1: EKISD.

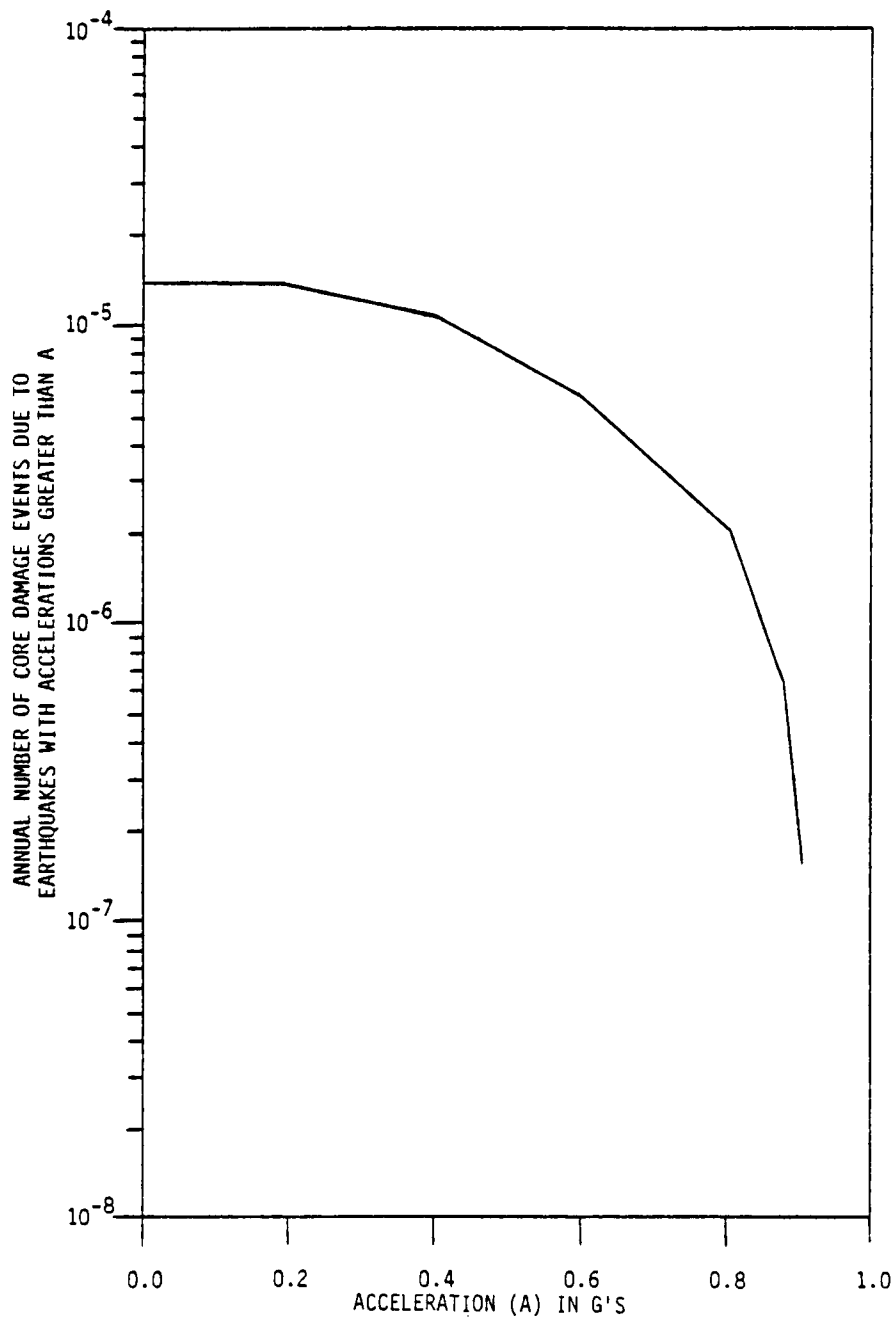


Figure VI-2. Exceedance Frequency Curve for Sequence 2: EK1.

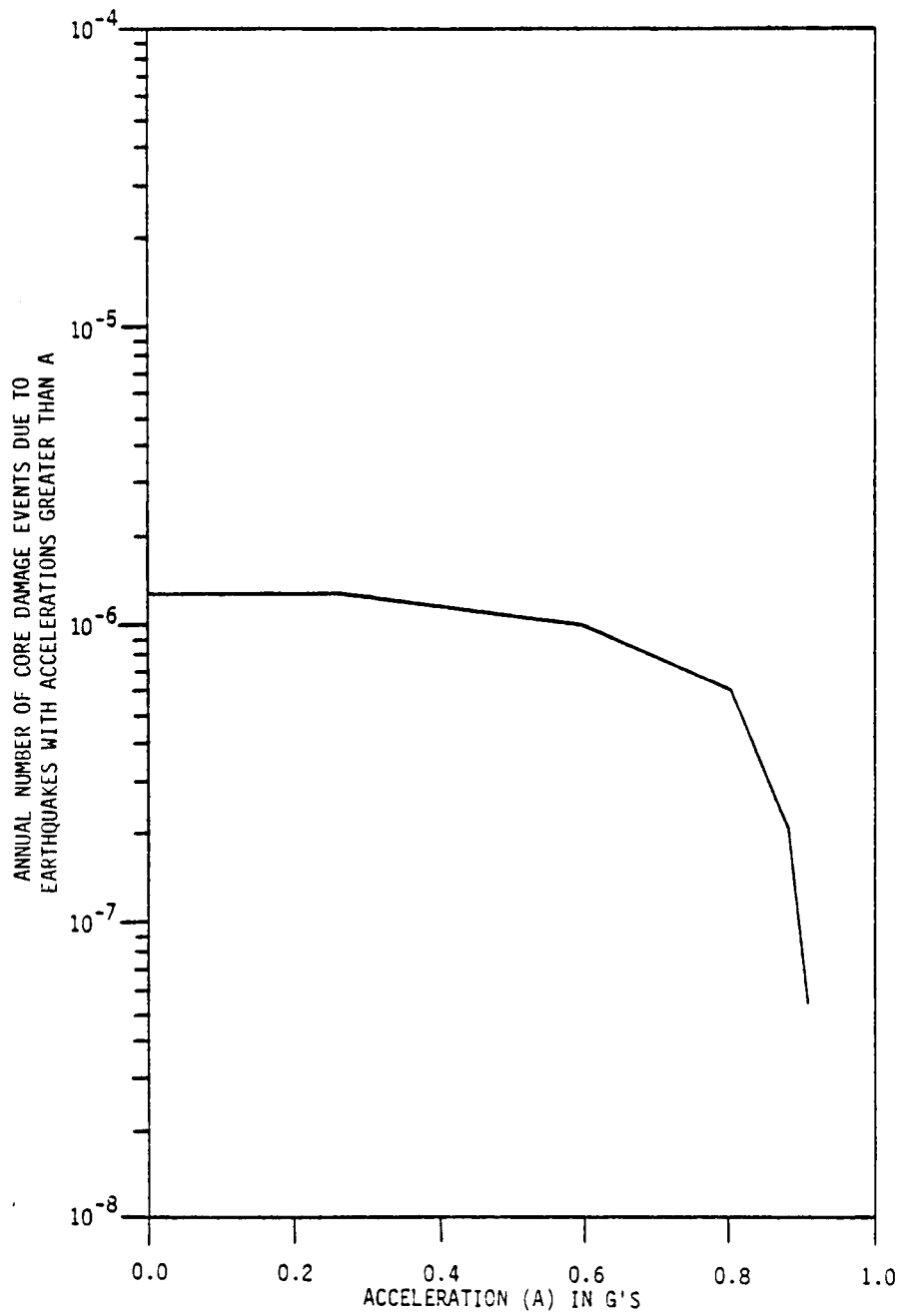


Figure VI-3. Exceedance Frequency Curve for Sequence 3: EKIP.

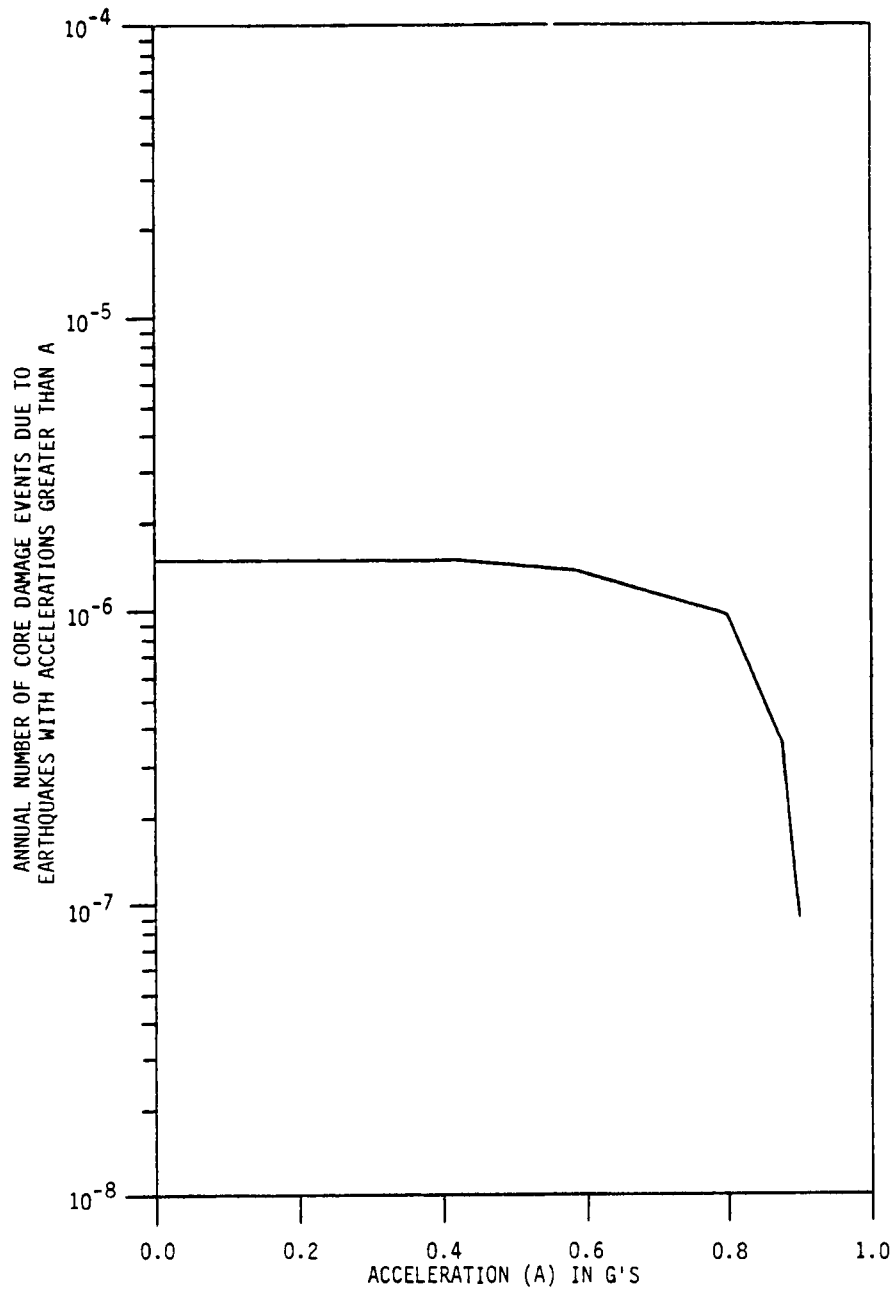


Figure VI-4. Exceedance Frequency Curve for Sequence 5: EKI.

The shape of each exceedance frequency curve suggests that the actual plant design (as modeled in this analysis) satisfies its seismic design objectives. A majority of the seismic risk is due to earthquakes with accelerations above the SSE limit of 0.25 g. (This is evident since each exceedance frequency curve is constant over the range of 0.0 to 0.25 g.) Hence, it is extremely unlikely that an earthquake with acceleration below the SSE will cause core damage.

VI.3 SENSITIVITY ANALYSIS

One of the more uncertain portions of this analysis concerns the assumptions made regarding component coupling. The previously reported results (Table VI-5) apply to the base case coupling assumptions (defined in Appendix B, Table B-3). Variations in the coupling assumption made for each component that appeared in the list of base case prime implicants were postulated to assess the effect of coupling on each sequence's occurrence frequency. The specific coupling assumptions selected for study concerned:

1. PHTS piping,
2. SWRPRS rupture disks,
3. Buildings,
4. Air blast heat exchangers,
5. Switchgears,
6. SGAHRS actuation logic, and
7. Balance-of-plant pumps (NPSW, EPSW, and ECW).

Components 2 through 7 only affect sequence 1 (EKISD); the first component affects sequences 1, 2, and 5 ($\overline{\text{EKISD}}$, $\overline{\text{EKI}}$, and EKI , respectively). Sequence 3 ($\overline{\text{EKIP}}$) has only one prime implicant (loss of offsite power * failure to trip); none of the components in this prime implicant are affected by coupling assumptions.

First, each of the above component's coupling assumptions were varied while maintaining the coupling assumptions on the remaining six components consistent with the base case. Then, several assumptions were simultaneously varied to determine the sets of assumptions that lead to the maximum and minimum total seismic core-damage frequencies. Table VI-6 summarizes the results of the sensitivity analysis.

As shown in Table VI-6, coupling assumptions about the PHTS piping and the SWRPRS rupture disks control the total seismically induced core-damage frequency. Variations in the coupling assumptions about buildings, air blast heat exchangers, switchgears, SGAHRS actuation logic, and balance-of-plant pumps do not cause any noticeable variation in either the total frequency or any sequence frequency. (These frequencies were computed to two significant digits.) In addition, the total seismically induced core-damage frequency was found to lie in the range of $2.0 \times 10^{-5}/\text{yr}$ to $6.2 \times 10^{-5}/\text{yr}$, depending on the particular set of coupling assumptions postulated. When compared to the base case frequency of $3.2 \times 10^{-5}/\text{yr}$, it was concluded that the seismic analysis results are rather insensitive to coupling assumptions.

Table VI-6

SENSITIVITY OF SEISMIC SEQUENCE FREQUENCIES TO COUPLING ASSUMPTIONS

PHTS Piping	SWRPRS Disks	Buildings	ABHX	Switchgear	SGAHS Logic	BOP Pumps	E*/K*/I*S*D	E*/K*I	E*K*/I*/P	E*K*I	TOTAL	Comments
B	B	B	B	B	B	B	1.5-5	1.4-5	1.3-6	1.5-6	3.2-5	basecase
C	B	B	B	B	B	B	1.8-5	6.8-6	1.3-6	7.5-7	2.7-5	
B	I	B	B	B	B	B	3.0-6	1.4-5	1.3-6	1.5-6	2.0-5	
B	C	B	B	B	B	B	1.2-5	1.4-5	1.3-6	1.5-6	2.9-5	
B	B	C	B	B	B	B	1.5-5	1.4-5	1.3-6	1.5-6	3.2-5	
B	B	B	I	B	B	B	1.5-5	1.4-5	1.3-6	1.5-6	3.2-5	
B	B	B	B	I	B	B	1.5-5	1.4-5	1.3-6	1.5-6	3.2-5	
B	B	B	B	B	I	B	1.5-5	1.4-5	1.3-6	1.5-6	3.2-5	
B	B	B	B	B	B	I	1.5-5	1.4-5	1.3-6	1.5-6	3.2-5	
G	B	R	B	B	B	B	1.5-5	4.4-5	1.3-6	1.7-6	6.2-5	maximum
B	I	B	I	I	I	I	3.0-6	1.4-5	1.3-6	1.5-6	2.0-5	minimum

B = basecase

C = coupled

G = guard vessels coupled to PHTS piping

I = independent

VI.4 QUALITATIVE UNCERTAINTY ANALYSIS

In addition to the assumptions made about component coupling, other assumptions were made that affect the values of the seismic sequence frequencies. The following assumptions affect the final results:

1. Earthquake occurrence may be modeled as a Poisson process. (That is, the probability of experiencing a given number of earthquakes in a specified time interval depends only on the duration of the time interval.)
2. Site seismicity may be determined by statistical analysis of historical earthquake data according to the distance between the earthquake's epicenter and the site. (This approach differs from traditional methods where data points are assigned to a particular earthquake generation source, such as a fault or tectonic region.)
3. The frequency-magnitude distribution of earthquakes may be obtained from the truncated Gutenberg-Richter correlation.
4. Reserve seismic margins obtained from the CRBRP Heat Transport System Incontainment Piping Reserve Seismic Margins¹⁴ and the CRBRP Reserve Seismic Margins¹⁵ reports are assumed to be median values. Chapter 8 of reference 15 discusses many conservatisms inherent in the calculated reserve seismic margins of CRBRP-specific components.

(For example, no credit was taken for ductility or energy absorption in the inelastic range.) Incorporation of these conservatisms would raise the reserve seismic margins of the CRBRP-specific components; this was not done due to lack of data. Further, the analyses in References 14 and 15 are based on "worst-case" components. This study assumed that the reserve seismic margins for the "worst-case" components applied to all similar components.

5. The logarithmic standard deviations (β_c values) for CRBRP-specific components (reserve seismic margins obtained from References 14 and 15) were obtained from analysis of LWR data. Quantification of the CRBRP-specific reserve seismic margin uncertainty could change (either increase or decrease) the logarithmic standard deviations used and, hence, could correspondingly change the final results.
6. The median reserve seismic margins and logarithmic standard deviations for CRBRP components that have LWR counterparts were taken to be the average values reported in four PRAs (Zion, Indian Point, Oconee, and McGuire). Detailed analysis of the specific components to be used in the CRBRP could change these parameters and, hence, could change the final results.

CHAPTER VII

CONCLUSIONS

The total core-damage frequency of the Clinch River Breeder Reactor Plant due to seismic events was estimated at 3.2×10^{-5} per year. Only earthquakes that cause the site peak ground acceleration to exceed 0.25 g (the CRBRP safe shutdown earthquake) were probabilistically significant contributors to the total seismically induced core-damage frequency. Seismically induced failures of the following components were found to be significant, regardless of any coupling effects:

1. Sodium-water reaction product relief system (SWRPRS) rupture disks;
2. Steam generator auxiliary heat removal system (SGAHRs) actuation logic cabinet;
3. Air blast heat exchangers;
4. Primary heat transport system (PHTS) piping;
5. Direct heat removal service (DHRS) piping;
6. Offsite power (161 kV) switchgear and ceramic insulators;
7. Normal plant service water (NPSW) pumps;
8. Emergency plant service water (EPSW) pumps;
9. Emergency chilled water (ECW) pumps;
10. 4160V buses 3A and 3A;
11. 4160V buses 3F, 3G, and 3H;
12. Reactor service building;

13. Reactor containment building;
14. Electrical equipment building;
15. Emergency pump house;
16. Diesel generator building;
17. Control building;
18. Steam generator building;
19. Reactor vessel guard vessel; and
20. Loop guard vessels.

Based on information in the CRBRP risk assessment¹³ (whose logic model was used to develop the seismic logic model in this study), seismically induced core-damage accidents comprise about 90% of the total core-damage frequency. Table VII-1 compares the seismically induced core-damage frequencies, along with their contributions to the total core-damage frequencies, determined in several risk assessments.

Based on the results summarized above, it is concluded that the CRBRP seismically induced core-damage frequency is similar to that assessed for most light water reactors. The actual design satisfies the seismic design criteria; only earthquakes that cause accelerations above 0.25 g (the safe shutdown earthquake) contribute to the total seismic risk. Since 90% of the total core-damage frequency was due to seismic events, it was concluded that the CRBRP is adequately designed to mitigate the effects of internal initiating events (e.g., loss of main feedwater).

Table VII-1

COMPARISON OF RESULTS WITH OTHER STUDIES

Study	Seismic Core-Damage Frequency	Percent of Total Core-Damage Frequency
1. This study	3.2-5	90
2. CRBRP-1 ¹²	5.6-6	28
3. Indian Point 2 ¹⁹	1.4-4	30
4. Indian Point 3 ¹⁹	3.1-6	<1
5. Oconee ²⁰	6.3-5	25
7. Zion (utility PRA) ¹⁸	5.6-6	90
8. Zion (SSMRP) ²³	3.0-5	--

Reduction of the seismically induced core-damage frequency may be achieved through a combination of component relocation and redesign. Results of the sensitivity analysis (Section VI.3) suggest that reorienting the sodium-water reaction product relief system (SWRPRS) rupture disks would lower the total frequency by a factor of 1.6. Additional strengthening (so as to raise the median acceleration capacity) of any or all of the components listed above would lower the total frequency.

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LIST OF REFERENCES

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APPENDICES

APPENDIX A

SUPPORTING DATA FOR COMPONENT FRAGILITY

This appendix contains data used to estimate component fragilities. Table A-1 contains information used to estimate primary and intermediary sodium piping reserve seismic margins. Reserve seismic margins for various portions of the reactor vessel guard vessel are listed in Table A-2. Estimates of the logarithmic standard deviation (β_c) for CRBRP-specific components using LWR data are given in Tables A-3 through A-8. Table A-9 shows the use of LWR data to obtain median acceleration capacities (A^v) and logarithmic standard deviations (β_c) for CRBRP components that have LWR counterparts.

Table A-1

ORIGINS OF SODIUM LEAK RESERVE SEISMIC MARGINS

Description of Initiator	Location of Minimum Capacity Component for Initiator	Reserve Seismic Margin*
Leak into Reactor Vessel Guard Vessel	Elbow in Cold Leg (See ES-LPD-83-001, Revision 1, Figure 3-3, Elbow #9) SOM = 3.953	5.73
PHTS Leak into Guard Vessel, Loops A, B, and C	Elbow in Guard Vessel Located on 6-inch Bubbler Line. (See ES-LPD-83-001, Revision 1, Figure 3-7, Elbow #1) SOM = 1.896	2.75
PHTS Leak Disables Loop and DHRS	Elbow in 24-inch PHTS Cold Leg Loop. (See ES-LPD-83-001, Revision 1, Figure 3-3, Elbow #8) SOM = 2.707	3.93
PHTS Leak Disables Loop but Allows DHRS Operation	Elbow in 24-inch PHTS Cold Leg Loop. (See ES-LPD-83-001, Revision 1, Figure 3-3, Elbow #4) SOM = 4.339	6.29
PHTS Loop Leak Disables DHRS but Allows Full PHTS Operation	Elbow in 6-inch Bubbler Line. (See ES-LPD-83-001, Revision 1, Figure 3-7, Elbow #2) SOM = 1.762	2.55
PHTS Loop Leak that Allows Full PHTS and DHRS Operation	Elbow in IHX Vent Return Line. (See ES-LPD-83-001, Revision 1, Figure 3-6, Elbow #23) SOM = 1.665	2.41
IHTS Leak	Leak in 24-inch IHTS Cold Leg. (See ES-LPD-83-001, Revision 1, Figure 3-5, Elbow # 5) SOM = 5.311	7.70

*Reserve Seismic Margin = Seismic-Only Margin X Conservatism Factor (1.45 for the CRBRP).

Table A-2

REACTOR VESSEL GUARD VESSEL RESERVE SEISMIC MARGIN

Part	Safety Margin (SM)	Seismic Stress to Nonseismic Stress Ratio (k)	Seismic Only Margin (SOM)	Conservatism Margin (CM)	Reserve Seismic Margin (RSM)	Median Capacity Acceleration in g's (A)
Head and Barrel	0.6	0.6	2.00	1.45	2.9	0.73
Support Skirt Bolts	0.01	0.6	1.02	1.45	1.47	0.37
Skirt Forging	1.2	0.6	3.00	1.45	4.35	1.09
Upper Flange	1.1	0.6	2.83	1.45	4.11	1.03
Inlet Nozzle and Elbow	0.07	0.6	1.12	1.45	1.62	0.40
Outlet Nozzle and Elbow	3.0	0.6	6.00	1.45	8.70	2.18

Table A-3

ESTIMATION OF LIQUID METAL PIPING FRAGILITY LOGARITHMIC
STANDARD DEVIATION FROM LWR DATA

Plant	Component	B_c
Zion		<u>0.67</u>
	Main Steam Piping and Supports (34")	0.71
	Main Feedwater Piping and Supports (16")	0.71
	Service Water Piping (supported, 12"-48")	0.65
	Service Water Piping (supported, 3"-8")	0.65
	Condensate Storage Piping (supported, 8")	0.64
	Condensate Storage Piping (supported, 10"-12")	0.65
	Containment Spray Piping (6"-8")	0.65
	Containment Spray Piping (10"-14")	0.66
	Reactor Coolant Loop Piping (27 1/2"-31")	0.59
	Reactor Coolant System Piping (1"-8")	0.70
	Reactor Coolant System Piping (10"-14")	0.71
	Residual Heat Removal Piping (8")	0.67
	Residual Heat Removal Piping (10"-14")	0.68
	Safety Injection Piping (4"-8")	0.67
	Safety Injection Piping (2"-4")	0.65
	Safety Injection Piping (10")	0.68
	Coolant Charging Piping (3"-8")	0.70
	Coolant Charging Piping (1 1/2"-3")	0.70
	Component Cooling Piping (3"-8")	0.70
	Component Cooling Piping (10"-16")	0.71
	Auxiliary Feedwater Piping (3"-6")	0.64
	Auxiliary Feedwater Piping (3")	0.65
Indian Point 3		<u>0.60</u>
	Main Steam Piping (28")	0.60
	Boiler Feedwater Piping (18")	0.63
	Service Water Piping (18")	0.60
	Service Water Piping (control building, 3")	0.61
	Service Water Piping (20")	0.60
	Service Water Piping (diesel generator, 10")	0.67
	Service Water Piping (containment, 3" and 10")	0.63
	Condensate Storage Piping (supported, 3"-12")	0.61
	Containment Spray (3"-4")	0.54

Table A-3 (Continued)

Plant	Component	β_c
	Containment Spray (containment, 8")	0.62
	Containment Spray (auxiliary building, 6"-8")	0.59
	Main Coolant Piping (27 1/2"-31")	0.55
	Reactor Coolant Piping (1"-4")	0.57
	Reactor Coolant Piping (6"-14")	0.61
	Residual Heat Removal Piping (auxiliary building, 8"-12")	0.63
	Residual Heat Removal Piping (containment, 8"-12", 14")	0.63
	Safety Injection Piping (auxiliary building, 6"-8")	0.59
	Safety Injection Piping (containment, 2"-4")	0.57
	Safety Injection Piping (containment, 6")	0.56
	Component Cooling Piping (auxiliary building, 3"-4")	0.54
	Component Cooling Piping (containment, 12")	0.63
	Component Cooling Piping (auxiliary, 6"-20")	0.63
	Auxiliary Feedwater Piping (3"-4")	0.55
	Auxiliary Feedwater Piping (3"-6")	0.63
Oconee		<u>0.56</u>
	Reactor Coolant Piping	0.54
	Class 2 and 3 Safety-Related Piping	0.58
Average (based on 48 evaluations)		0.63
Lowest Reported Value		0.54
Highest reported Value		0.71

Table A-4

ESTIMATION OF LIQUID METAL HEAT EXCHANGER FRAGILITY
LOGARITHMIC STANDARD DEVIATION FROM LWR DATA

Plant	Component	β_c
Zion		<u>0.51</u>
	Residual Heat Removal Heat Exchangers	0.58
	Component Cooling Heat Exchangers	0.44
Indian Point 3		<u>0.42</u>
	Residual Heat Removal Heat Exchangers	0.47
	Component Cooling Heat Exchangers	0.36
Oconee		<u>0.58</u>
	Letdown Cooler	0.56
	Reactor Coolant Pump Seal Return Cooler	0.55
	Low Pressure Injection Coolers	0.55
	Component Coolers	0.61
	Spent Fuel Pool Coolers	0.61
	Feedwater Heaters	0.60
	Reactor Building Coolers	0.55
McGuire		<u>0.54</u>
	Residual Heat Removal Heat Exchangers	0.60
	Regenerative Heat Exchanger	0.61
	Component Cooling Water Heat Exchangers	0.45
	Containment Spray Heat Exchangers	0.50
Average (based on 15 evaluations)		0.54
Lowest Reported Value		0.36
Highest Reported Value		0.61

Table A-5

ESTIMATION OF CONTROL ROD FRAGILITY LOGARITHMIC
STANDARD DEVIATION FROM LWR DATA

Plant	β_c
Zion (core geometry)	0.49
Indian Point 3 (core geometry)	0.45
Oconee (reactor internals including control rods)	0.47
Average (based on 3 evaluations)	0.47
Lowest Reported Value	0.45
Highest Reported Value	0.49

Table A-6

ESTIMATION OF STEAM GENERATING SYSTEM FRAGILITY
LOGARITHMIC STANDARD DEVIATION FROM LWR DATA

Plant	β_c
Zion	0.54
Indian Point 3	0.50
Average (based on 2 evaluations)	0.52

Table A-7

ESTIMATION OF LIQUID METAL EXPANSION TANK FRAGILITY
LOGARITHMIC STANDARD DEVIATION FROM LWR DATA

Plant	Component	β_c
Zion		<u>0.54</u>
	Condensate Storage Tank	0.40
	Refueling Water Storage Tank	0.41
	Pressurizer	0.59
	Reactor Vessel	0.59
	Reactor Coolant System Relief Tank	0.66
	Boron Injection Tank	0.56
Indian Point 3		<u>0.47</u>
	Condensate Storage Tank	0.33
	Refueling Water Storage Tank	0.36
	Containment Spray Additive Tank	0.60
	Pressurizer	0.46
	Pressurizer Relief Tank	0.61
	Boron Injection Tank	0.47
Oconee		<u>0.53</u>
	Pressurizer	0.34
	Letdown Boric Acid Storage Tank	0.46
	Reactor Coolant Bleed Hold Up	0.40
	Core Flooding Tank	0.65
	Upper Surge Tank	0.54
	High Pressure Service Water Storage Tank	0.60
	Component Cooling Water Surge Tank	0.60
	Recirculating Cooling Water Surge Tank	0.60
	Condensate Storage Tank	0.60

Table A-7 (Continued)

Plant	Component	B_c
McGuire		<u>0.57</u>
	Upper Surge Tank	0.60
	Auxiliary Feedwater Condensate Tank	0.60
	Diesel Generator Fuel Oil Day Tank	0.30
	Diesel Generator Starting Air Tank	0.62
	Boron Injection Tank	0.63
	Cold Leg Accumulators	0.61
	Upper Head Injection Tanks	0.55
	Chemical and Volume Control Tank	0.63
Average (based on 30 evaluations)		0.53
Lowest Reported Value		0.30
Highest Reported Value		0.66

Table A-8

ESTIMATION OF CRBRP BUILDING FRAGILITY LOGARITHMIC
STANDARD DEVIATION FROM LWR DATA

Plant	Component	β_c
Zion		<u>0.48</u>
	Reactor Building (soil failure beneath foundation slab)	0.44
	Reactor Building (collapse of pressurizer enclosure roof)	0.52
	Reactor Building (shear failure of containment wall)	0.54
	Reactor Building (vertical shear failure at buttress plates)	0.51
	Reactor Building (flexural failure of containment wall)	0.50
	Reactor Building (failure of internal structure shear anchors)	0.50
	Reactor Building (shear failure of foundation slab)	0.48
	Auxiliary Building (failure of concrete shear walls)	0.41
	Auxiliary Building (failure of concrete roof diaphragm)	0.46
Indian Point		<u>0.37</u>
	Shear Failure of Containment Wall	0.40
	Buried Concrete Structures (failure of walls)	0.33
Oconee		<u>0.39</u>
	Shear Failure of Containment	0.39
	Flexural Failure of Containment	0.40
	Shear Failure of Primary Shield Wall	0.38
	Flexural Failure of Primary Shield Wall	0.37
	Shear Failure of Secondary Shield Wall	0.38
	Shear Failure of Auxiliary Building	0.33
	Shear Walls	0.33

Table A-8 (Continued)

Plant	Component	β_c
	Failure of Auxiliary Building Concrete Moment Frames	0.36
	Failure of North-South Braced Frames	0.45
	Failure of East-West Moment Frames	0.42
	Failure of Standby Shutdown Facility Shear Walls	0.41
McGuire		
	Shear Failure of Reactor Building Wall	0.45
	Flexural Failure of Crane Wall	0.46
	Flexural Failure of Auxiliary Building Shear Walls	0.45
	Average (based on 24 evaluations)	0.43
	Lowest Reported Value	0.33
	Highest Reported Value	0.54

Table A-9

LWR DATA USED TO ESTIMATE CRBRP COMPONENT FRAGILITY

CRBRP Component	Zlon (SSE=.17 g)			Indian Point (SSE=.15 g)			Oconee (SSE=.1 g)			McGuire (SSE=.15 g)		
	V	A	RSM	β_c	V	A	RSM	β_c	V	A	RSM	β_c
<u>Auxiliary Feedwater</u>												
Motor-Driven Pumps	6.85		40.29	.56	15.0		100.0	.55		.69	4.6	.36
Turbine-Driven Pumps	6.85		40.29	.56	15.0		100.0	.55		.48	3.2	.38
PWST	1.19		7.0	.57	1.37		9.13	.61				
<u>Electric Power</u>												
Switchgear (4160V)	.72		4.23	.59	1.5		10.0	.58	1.2	.4	2.66	.50
Diesel Generator	.86		5.06	.51	1.3		8.66	.57	1.42			
Transformers	1.39		8.18	.65	1.07		7.13	.67	.40			
Switchgear (480V)	.72		4.23	.59	1.51		10.06	.58				
Motor Control Centers	.72		4.23	.59	1.17		7.8	.57	1.20	1.68	11.2	.54
Cable Trays	4.74		27.88	.71	2.20		14.66	.59				
Inverters	2.69		15.82	.48	4.30		26.86	.48				
Distribution Panels	.60		3.52	.62	1.19		7.93	.56				
Batteries and Racks	1.01		3.94	.69	1.17		7.8	.66	1.59	.95	6.33	.47
DC Bus Work	.60		3.52	.62								
D/G Day Tanks									1.95	1.44	9.6	.30
Ceramic Insulators	.2		1.17	.32	.2		1.33	.32	.2	.2	1.33	.32

Table A-9 (Continued)

ORBRP Component	Zion (SSE=.17 g)			Indian Polnt (SSE=.15 g)			Oconee (SSE=.1 g)			McGuire (SSE=.15 g)		
	V	A	RSM	β_c	V	A	RSM	β_c	V	A	RSM	β_c
<u>Emergency Chilled Water</u>												
Chillers	8.32		48.94	.44	6.13		40.87	.36		1.66	11.07	.45
Pumps	6.63 ^a		39.00	.56	6.60		44.0	.54		1.74	11.6	.55
	.63 ^b		3.71	.39	2.47		16.47	.79		.92	6.13	.35
<u>Emergency Service Water</u>												
Pumps	6.63 ^a		39.00	.56	6.60		44.0	.54		1.74	11.6	.55
	.63 ^b		3.71	.39	2.47		16.47	.79		.92	6.13	.35
<u>Heat Rejection</u>												
Pumps	6.63 ^a		39.00	.56	6.60		44.0	.54		1.74	11.6	.55
	.63 ^b		3.71	.39	2.47		16.47	.79		.92	6.13	.35
<u>Heat Transport</u>												
PHTS Pumps	3.29 ^c		19.35	.54	3.04		20.27	.50				
IHTS Pumps	3.29 ^c		19.35	.54	3.04		20.27	.50				
Reactor Vessel	7.31		43.0	.59	3.80		25.33	.55				
<u>Main Feedwater</u>												
Feed Pumps	6.63		39.0	.56	6.60		44.0	.54		1.74	11.6	.55
Condensate Pumps	6.63		39.0	.56	6.60		44.0	.54		1.74	11.6	.55

Table A-9 (Continued)

CRBRP Component	Zion (SSE=.17 g)			Indian Point (SSE=.15 g)			Oconee (SSE=.1 g)			McGuire (SSE=.15 g)					
	V	RSM	β_c	V	A	RSM	β_c	V	A	RSM	β_c	V	A	RSM	β_c
<u>Normal Chilled Water</u>															
Pumps	.63 ^a	3.70	.39	2.47		16.46	.79	---	.92				.92	6.13	.35
	6.63 ^b	39.0	.56	6.60		44.0	.54	1.12	1.74	11.2	.61		1.74	11.6	.55
Chillers	8.32	48.94	.44	6.13		40.87	.36	---	1.66	---	---		1.66	11.07	.45
<u>Normal Service Water</u>															
Pumps	.63 ^a	3.70	.39	2.47		16.46	.79	---	.92				.92	6.13	.35
	6.63 ^b	39.0	.56	6.60		44.0	.54	1.12	1.74	11.2	.61		1.74	11.6	.55
<u>Reactor Protection</u>															
ESF Logic Panels	2.67	15.8	.48	5.07		33.8	.48	.80	---	5.33	1.37		---	---	---
Instrumentation	2.67	15.8	.48	5.07		33.8	.48	---	---	---	---		---	---	---
<u>SSCCW</u>															
Pumps	.63	3.70	.39	2.47		16.46	.79	---	.92				.92	6.13	.35
Heat exchangers	8.32	48.94	.44	6.13		40.87	.36	---	1.66	---	---		1.66	11.07	.45
<u>SGS</u>															
Steam Piping	3.84	22.59	.71	2.03		13.53	.6	---	---	---	---		---	---	---
Relief/Safety Valves (FTSD)	6.69	39.35	.79	6.04		40.27	.70	---	---	---	---		---	---	---

^aComponent cooling^bService water^cReactor coolant pump data

APPENDIX B

SUPPORTING DATA FOR SEISMIC LOGIC MODEL

This appendix contains three tables that comprise the seismic logic model used to generate core-damage sequences. Table B-1 defines the base case seismic logic model by listing the components assumed to be coupled and uncoupled. Changes to the internal initiator fault trees¹³ are given in Table B-2. Table B-3 lists all basic events added to the logic model.

Table B-1

BASE CASE SEISMIC LOGIC MODEL

Coupled Components

1. Air blast heat exchangers (755', E-W, RSB)
2. Batteries (765', E-W, CB)
3. Buses 3A, 3B, 3E (4160 V ac, 794', N-S, EEB)
4. Buses 5A, 5B (125 V dc, 765', E-W, CB)
5. Buses 8A, 8B, 8C (120/280 V ac, 765', E-W, CB)
6. Buses 16C, 28A, 28B, 52B, (480 V ac, 794', E-W, SGB)
7. Buses 32A, 32B, 41B, (480 V ac, 733', N-S, RSB)
8. Buses 44C, 52A (480 V ac, 836', E-W, SGB)
9. Condensate pumps (816', vertical, TGB)
10. DHRS pumps (733', E-W, RSB)
11. Emergency chilled water chillers (733', E-W, SGB)
12. Emergency chilled water pumps (733', E-W, DGB)
13. Emergency diesel generators (816', E-W, DGB)
14. Emergency diesel generator day tanks (814', vertical, DGB)
15. Emergency plant service water pumps (832', vertical, EPH)
16. Evaporator modules (806', vertical, SGB)
17. Evaporator SWRPRS rupture disks (816', E-W, SGB)
18. Ex-vessel storage tank coolers (779', E-W, RSB)
19. Heat rejection pumps (818', vertical, CPH)
20. Inverters (765', E-W, CB)
21. Main feedwater pumps (816', E-W, TGB)
22. Motor-driven auxiliary feedwater pumps (733', E-W, SGB)
23. NaK expansion tanks (792', E-W, RSB)
24. Normal plant service water pumps (818', vertical, CPH)
25. Recirculating gas cooling system coolers (752', N-S, CB)
26. SGAHRS logic cabinets (816', H-S, CB)
27. Secondary services closed cooling water coolers (816', N-S, TGB)
28. Secondary services closed cooling water pumps (816', E-W, TGB)
29. Superheater modules (806', vertical, SGB)
30. Superheater SWRPRS rupture disks (816', N-S, SGB)
31. Transformers 28A, 28B (779', N-S, RSB)
32. Transformers 40A, 40B, 40C (765', E-W, CB)

Table B-1 (Continued)

Non-coupled Components

1. Buildings (all buildings independent)
 2. Bus 16C (480 V ac, 794', E-W, SGB)
 3. Bus 41A (480 V ac, 755', E-W, RSB)
 4. Bus 42A (480 V ac, 863', N-S, CB)
 5. Bus 42B (480 V ac, 847', N-S, CB)
 6. DHRS piping (all trains independent)
 7. IHTS piping (all loops independent)
 8. PHTS guard vessels (guard vessels independent of piping they protect)
 9. PHTS piping (all loops independent)
 10. PHTS pumps (all loops independent)
 11. Protected water storage tank (748', E-W, SGB)
 12. Turbine-driven auxiliary feedwater pump (733', E-W, SGB)
-
-

Table B-2

INCORPORATION OF SEISMIC EVENTS INTO
THE INTERNAL INITIATOR FAULT TREE

Event Definition

Auxiliary Feedwater

SMPUMPA53 = SUPUMPA531 + SMPUMPA53S + MDAFWPS + NIB + SGB
 SMPUMPB53 = SUPUMPB531 + SMPUMPB53S + MDAFWPS + NIB + SGB
 SMTDPCH53 = SMTDPCH531 + SMTDPCH53S + NIB + SGB
 S52TKPWST = S52TKPWST1 + 552TKPWSTS + NIB + SGB

Direct Heat Removal Service

RMPIPE = RMPIPE1 + RMPICES + RSB + NIB
 LMABHXA = LMABHXA1 + LMABHXAS + ABHXS + RSB + NIB
 LMABHXB = LMABHXB1 + LMABHXBS + ABHXS + RSB + NIB
 LMLEAKA = LMLEAKA1 + EVSCoolERa + EVSCoolS + NKEXPTANKA +
 NKEXPTANKS + RSB + NIB
 LMLEAKB = LMLEAKB1 + EVSCoolERB + EVSCoolS + NKEXPTANKB +
 NKEXPTANKS + RSB + NIB
 LMPUMP3A = LMPUMP3A1 + DHRSPUMPA + DHRSPUMPS + RSB + NIB
 LMPUMP3B = LMPUMP3B1 + DHRSPUMPB + DHRSPUMPS + RSB + NIB

Electric Power

EBS3AG = EBS3AG1 + EB41603AG + 4160S + NIB + EB416NS794 + EEB
 EBS3BG = EBS3BG1 + EB41603BG + 4160S + NIB + EB416NS794 + EEB
 EBS3EG = EBS3EG1 + EB41603EG + 4160S + NIB + EB416NS794 + SGB
 EMBATTERY = EMBATTERY1 + EBATTERY + EBATTS + NIB + CB
 EMBATT10A = EMBATT10A1 + EBATT10A + EBATTS + NIB + CB
 EMBATT10B = EMBATT10B1 + EBATT10A + EBATTS + NIB + CB
 EMBUS5A = EMBUS5A1 + EB1255A + EB125S + NIB + CB
 EMBUS5B = EMBUS5B1 + EB1255B + EB125S + NIB + CB
 EMEDG22C = EMEDG22C1 + EDGC + EDGS + NIB + DGB
 EMFUELA = EMFUELA1 + EFUELA + FUELS + NIB + DGB
 EMFUELB = EMFUELB1 + EFUELB + FUELS + NIB + DGB
 EMFUELC = EMFUELC1 + EFUELC + FUELS + NIB + DGB
 EMBUS8A = EMBUS8A1 + ESEIS8A + ESEISS + NIB + CB
 EMBUS8B = EMBUS8B1 + ESEIS8B + ESEISS + NIB + CB

Table B-2 (Continued)

Event Definition	
<u>Electric Power (Continued)</u>	
EMEDGAL	= EMEDGALI + EB41603GG + 4160S + EB4160816 + DGB + NIB
EMEDGBL	= EMEDGBLI + EB41603HG + 4160S + EB4160816 + DGB + NIB
EMEDGCL	= EMEDGCLI + EB41603FG + 4160S + EB4160816 + DGB + NIB
EMBUS8C	= EMBUS8C1 + ESEIS8C + ESEISS + NIB + CB
EMDCB	= EMDCB1 + EINVERT14B + EINVERTS + NIB + CB
EMDCA	= EMDCA1 + EINVERT14A + EINVERTS + NIB + CB
EMDCC	= EMDCC1 + EINVERT14C + EINVERTS + NIB + CB
EMAC	= EMAC1 + EXFMRAC + XF765EW + EXFMR + NIB + CB
EMDC480A	= EMDC480A1 + EXFMR40A + XF765EW + EXFMR + NIB + CB
EMDC480B	= EMDC480B1 + EXFMR40B + XF765EW + EXFMR + NIB + CB
EMXFMR28A	= EMBUS28AX + EXFMR28A + XF780NS + EXFMR + NIB + RSB
EMXFMR28B	= EMBUS28BX + EXFMR28B + XF780NS + EXFMR + NIB + RSB
EMBUS16C	= EMBUS16C1 + EB480V16C + EB794EW + EB480EW + EB480794 + EB480V + NIB + SGB
EMBUS28AX	= EMBUS28A1 + EB480V28A + EB794EW + EB480EW + EB480794 + EB480V + NIB + RSB
EMBUS28BX	= EMBUS28B1 + EB480V28B + EB794EW + EB480EW + EB480794 + EB480V + NIB + RSB
EMBUS52B	= EMBUS52B1 + EB480N52B + EB794EW + EB480EW + EB480794 + EB480V + NIB + SGB
EMBUS61C	= EMBUS61C1 + EB480V61C + EB768EW + EB480EW + EB480768 + EB480V + NIB + CB
EMBUS44C	= EMBUS44C1 + EB480V44C + EB836EW + EB480EW + EB480836 + EB480V + NIB + SGB
EMBUS41A	= EMBUS41A + EB480V41A + EB755EW + EB480EW + EB480755 + EB480V + NIB + RSB
EMBUS32B	= EMBUS32B1 + EB480V32B + EB733NS + EB480NS + EB480733 + EB480V + NIB + RCB
EMBUS41B	= EMBUS41B1 + EB480V41B + EB733NS + EB480NS + EB480733 + EB480V + NIB + RCB
EMBUS42A	= EMBUS42A1 + EB480V42A + EB863NS + EB480NS + EB480863 + EB480V + NIB + CB
EMBUS42B	= EMBUS42B1 + EB480V42B + EB847NS + EB480NS + EB480847 + EB480V + NIB + CB
EMBUS52A	= EMBUS52A1 + EB480V52A + EB836NS + EB480NS + EB480836 + EB480V + NIB + SGB
EMBUS32A	= EMBUS32A1 + EB480V32A + EB733NS + EB480NS + EB480733 + EB480V + EXFMR32A + XF780NS + EXFMR + NIB + RCB

Table B-2 (Continued)

Event Definition

Electric Power (Continued)

EMEDGAS = EMEDGASI + EDGA + EDGS + DGB + NIB
 EMEDGBS = EMEDGBSI + EDGB + EDGS + DGB + NIB
 EMEDGCS = EMEDGCSI + EDGC + EDGS + DGB + NIB

EMXFMR5A = EMXFMR5A1 + EXFMR5A + EXFMR + EXFMR816 + EB1611 + EB161S
 EMXFMR6A = EMXFMRGA1 + EXFMRGA + EXFMR + EXFMR816 + EB16129 + EB161S
 EMXFMR5B = EMXFMR5B1 + EXFMR5B + EXFMR + EXFMR816 + EB1611 + EB161S
 EMXFMR6B = EMXFMR6B1 + EXFMR6B + EXFMR + EXFMR816 + EB16129 + EB161S
 EMBUS58A = EMBUS58A1 + EB48058A + EB733NS + EB480733 + EB480NS +
 EB480V + RCB + NIB
 EMBUS58B = EMBUS58B1 + EB48058B + EB733NS + EB480733 + EB480NS +
 EB480V + RCB + NIB
 EMBUS50A = EMBUS50A1 + EB48050A + EB480836 + EB480NS + EB480V +
 SGB + NIB
 EMBUS50B = EMBUS50B1 + EB48050B + EB480836 + EB480NS + EB480V +
 SGB + NIB

Emergency Chilled Water

XM1ECWA = XM1ECWA1 + ECWPUMPA + ECWPUMPS + ECWCOOLA + ECWCOOLS +
 SGB + NIB
 XM1ECWB = XM1ECWB1 + ECWPUMPB + ECWPUMPS + ECWCOOLB + ECWCOOLS +
 SGB + NIB

Emergency Plant Service Water

WMA4L = WMA4L1 + EPSWPUMPA + EPSWPUMPS + EPH + NIB
 WMB4L = WMB4L1 + EPSWPUMPB + EPSWPUMPS + EPH + NIB
 WMC3 = WMC3 + EPSWPUMPC + EPSWPUMPS + EPH + NIB

Heat Rejection

NMCIRC2A73 = NMC1RC2A731 + HRPUMPA + HRPUMPS + NIB + TGB
 NMCIRC2B73 = NMC1RC2B731 + HRPUMPB + HRPUMPS + NIB + TGB
 NMCIRC2C73 = NMC1RC2C731 + HRPUMPC + HRPUMPS + NIB + TGB

Table B-2 (Continued)

Event Definition

Main Feedwater and Condensate

$\text{FMCDPMA272} = \text{FMCDPMA2721} + \text{CDSPUMPA} + \text{CDSPUMPS} + \text{NIB} + \text{TGB}$
 $\text{FMCDPMB272} = \text{FMCDPMB2721} + \text{CDSPUMPB} + \text{CDSPUMPS} + \text{NIB} + \text{TGB}$
 $\text{FMCDPMC272} = \text{FMCDPMC2721} + \text{CDSPUMPC} + \text{CDSPUMPS} + \text{NIB} + \text{TGB}$
 $\text{FMFWPMA272} = \text{FMFWPMA2721} + \text{MFWPUMPA} + \text{MFWPUMPS} + \text{NIB} + \text{TGB}$
 $\text{FMFWPMB272} = \text{FMCDPMB2721} + \text{MFWPUMPB} + \text{MFWPUMPS} + \text{NIB} + \text{TGB}$
 $\text{FMFWPMC272} = \text{FMFWPMC2721} + \text{MFWPUMPC} + \text{MFWPUMPS} + \text{NIB} + \text{TGB}$

Normal Chilled Water

$\text{XM1NCW} = (\text{A*B*C} + \text{A*B*D} + \text{A*B*F} + \text{A*C*D} + \text{A*C*E} + \text{A*C*F} + \text{A*D*E} +$
 $\text{A*D*F} + \text{B*C*D} + \text{B*C*E} + \text{B*C*F} + \text{C*D*E} + \text{C*D*F} + \text{D*E*F}) +$
 $\text{NCWPMS} + \text{NCWCHS} + \text{NCWPMNS} + \text{NCWPMEW} + \text{NCWCHNS} + \text{NCWCHEW} +$
 $\text{NCWCH794} + \text{NCWPM794} + \text{NCWPM794EW} + \text{NCWCH794EW} + \text{RCB} + \text{NIB}$

$\text{A} = \text{NCWPMA} + \text{NCWCHA}$
 $\text{B} = \text{NCWPMB} + \text{NCWCHB}$
 $\text{C} = \text{NCWPMC} + \text{NCWCHC}$
 $\text{D} = \text{NCWPMD} + \text{NCWCHD}$
 $\text{E} = \text{NCWPME} + \text{NCWCHE}$
 $\text{F} = \text{NCWPMF} + \text{NCWCHF}$

Normal Plant Service Water

$\text{WMN1} = \text{WMN11} + \text{NPSWCOOLA} + \text{CPH} + \text{NIB}$
 $\text{WMN2} = \text{WMN21} + \text{NPSWPUMPA} + \text{NPSWPUMPS} + \text{CPH} + \text{NIB}$
 $\text{WMN3} = \text{WMN31} + \text{NPSWPUMPB} + \text{NPSWPUMPS} + \text{CPH} + \text{NIB}$

Reactor Protection

$\text{S56CLDID} = \text{S56CLDID1} + \text{LOGICI} + \text{LOGICS} + \text{CB} + \text{NIB}$
 $\text{S56CLDIID} = \text{S56CLDIID1} + \text{LOGICII} + \text{LOGICS} + \text{CB} + \text{NIB}$
 $\text{S56CLDIIID} = \text{S56CLDIIID1} + \text{LOGICIII} + \text{LOGICS} + \text{CB} + \text{NIB}$

Table B-2 (Continued)

Event Definition
<u>Recirculating Gas Cooling System</u>
MMRGC10328 = MMRGC103281 + RGC103COOL + RGCS + RCB + NIB
MMRGC10428 = MMRGC104281 + RGC104COOL + RGCS + RCB + NIB
<u>Secondary Services Closed Cooling Water</u>
WSM3 = WSM31 + SSCCW PUMPA + SSCCW PUMPS + SSCCW PUMPB + SSCCW COOLA + SSCCW COOLB + SSCCW COOLS + NIB + TGB
<u>Steam Generating System</u>
QMEXITA = QMEXITA1 + SGSEX1TA + SGSEXITS + SGSMODULES + SGB + NIB
QMEXITB = QMEXITB1 + SGSEX1TB + SGSEXITS + SGSMODULES + SGB + NIB
QMEXITC = QMEXITC1 + SGSEX1TC + SGSEXITS + SGSMODULES + SGB + NIB
QMEEVAPA = QMEEVAPA1 + SGSEVAPA + SGSEVAPS + SGSMODULES + SGB + NIB
QMEEVAPB = QMEEVAPB1 + SGSEVAPB + SGSEVAPS + SGSMODULES + SGB + NIB
QMEEVAPC = QMEEVAPC1 + SGSEVAPC + SGSEVAPS + SGSMODULES + SGB + NIB
QMWEVAPA = QMWEVAPA1 + SGSWVAPA + SGSEVAPS + SGSMODULES + SGB + NIB
QMWEVAPB = QMWEVAPB1 + SGSWVAPB + SGSEVAPS + SGSMODULES + SGB + NIB
QMWEVAPC = QMWEVAPC1 + SGSWVAPC + SGSEVAPS + SGSMODULES + SGB + NIB
<u>Top Logic</u>
TD1G1 = TD1G1 + NIB + RSB
TS1G1 = TS1G1 + NIB + SGB
T2 = CDSPUMPS + MFWPUMPS + HRPUMPS + NIB + TGB + CPH
T3 = CDSPUMPS + MFWPUMPS + NIB + TGB
T6 = PHTSPUMPA + PHTSPUMPB + PHTSPUMPC + PHTSPUMPAB + PHTSPUMPAC + PHTSPUMPBC + PHTSPUMPS + NIB + RCB
T7A = T7AEE + T7AWE + T7ASH + T7EE + T7WE + T7SH + T7EVAPS + T7AS + T7 + SGB + NIB
T7B = T7BEE + T7BWE + T7BSH + T7EE + T7WE + T7SH + T7EVAPS + T7BS + T7 + SGB + NIB
T7C = T7CEE + T7CWE + T7CSH + T7EE + T7WE + T7SH + T7EVAPS + T7CS + T7 + SGB + NIB
T8A = T8AEE + T8AWE + T8ASH + T8EE + T8WE + T8SH + T8EVAPS + T8AS + T8 + SGB + NIB
T8B = T8BEE + T8BWE + T8BSH + T8EE + T8WE + T8SH + T8EVAPS + T8BS + T8 + SGB + NIB
T8C = T8CEE + T8CWE + T8CSH + T8EE + T8WE + T8SH + T8EVAPS + T8CS + T8 + SGB + NIB

Table B-2 (Continued)

Event Definition

Top Logic (Continued)

$T9A = T9AS + T9AB + T9AC + T9 + NIB + RCB + SGB$
 $T9B = T9BS + T9AB + T9BC + T9 + NIB + RCB + SGB$
 $T9C = T9CS + T9AC + T9BC + T9 + NIB + RCB + SGB$
 $T10A = T10AS + T10AB + T10AC + T10 + NIB + RCB$
 $T10B = T10BS + T10AB + T10BC + T10 + NIB + RCB$
 $T10C = T10CS + T10AC + T10BC + T10 + NIB + RCB$
 $T11A = T11AS + T11AB + T11AC + T11 + NIB + RCB$
 $T11B = T11BS + T11AB + T11BC + T11 + NIB + RCB$
 $T11C = T11CS + T11AC + T11BC + T11 + NIB + RCB$
 $T12A = T12AS + T12AB + T12AC + T12 + NIB + RCB$
 $T12B = T12BS + T12AB + T12BC + T12 + NIB + RCB$
 $T12C = T12CS + T12AC + T12BC + T12 + NIB + RCB$
 $T14A = T14AEE + T14AWE + T14ASH + T14EE + T14WE + T14SH + T14AS +$
 $T14 + SGB + NIB$
 $T14B = T14BEE + T14BWE + T14BSH + T14EE + T14WE + T14SH + T14AS +$
 $T14 + SGB + NIB$
 $T14C = T14CEE + T14CWE + T14CSH + T14EE + T14WE + T14SH + T14AS +$
 $T14 + SGB + NIB$
 $T15 = XM1NCW$
 $T16 = NPSWPUMPS + NPSWPUMPA + NPSWPUMPB + CPH + NIB$
 $T17 = (EMXFMR5A + EMXFMR6A) * (EMXFMR5B + EMXFMR6B) + NIB$
 $T20 = (EMDCA + EMBUS5A + EMDCB + EMBUS5B + EMDCC) + NIB$
 $T21A = T21AS + T21AB + T21AC + T21 + NIB + RCB$
 $T21B = T21BS + T21AB + T21BC + T21 + NIB + RCB$
 $T21C = T21CS + T21AC + T21BC + T21 + NIB + RCB$
 $T22A = T22AS + T22AB + T22AC + T22 + NIB + RCB$
 $T22B = T22BS + T22AB + T22BC + T22 + NIB + RCB$
 $T22C = T22CS + T22AC + T22BC + T22 + NIB + RCB$
 $SP1G2A = /T4*/PHTSPUMPA*/PHTSPUMPAB*/PHTSPUMPAC*/PHTSPUMPS$
 $SP1G2B = /T4*/PHTSPUMPB*/PHTSPUMPAB*/PHTSPUMPBC*/PHTSPUMPS$
 $SP1G2C = /T4*/PHTSPUMPC*/PHTSPUMPAC*/PHTSPUMPBC*/PHTSPUMPS$
 $P51APANT = P51APBNT1 * SP1G2A.$
 $P51APBNT = P51APBNT1 * SP1G2B.$
 $P51APCNT = P51APCNT1 * SP1G2C.$
 $SP1G1 = P51APANT + P51APBNT + P51APCNT.$
 $TI2G5A = SP1G1 + TIGVLEAKA$
 $TI2G5B = SP1G1 + TIGVLEAKB$
 $TI2G5C = SP1G1 + TIGVLEAKC$

Table B-2 (Continued)

Event Definition
<u>Top Logic (Continued)</u>
$TI2G4A = T11A + T12A + T21A + T22A$
$TI2G4B = T11B + T12B + T21B + T22B$
$TI2G4C = T11C + T12C + T21C + T22C$
$TI2G3A = T10A * TI2G5A$
$TI2G3B = T10B * TI2G5B$
$TI2G3C = T10C * TI2G5C$
$TI2G2A = TI2G4A * SP1G1$
$TI2G2B = TI2G4B * SP1G1$
$TI2G2C = TI2G4C * SP1G1$
$TI2G1A = TI2G2A + TI2G3A.$
$TI2G1B = TI2G2B + TI2G3B.$
$TI2G1C = TI2G2C + TI2G3C.$
$TI1G3 = SP1G1 + TIRVGVLEAK.$
$TI1G2 = T10R * TI1G3.$
$TI1G1 = TI1G2 + TI2G1A + TI2G1B + TI2G1C$
$NOT1 = /TI1G1$

Table B-3

LIST OF BASIC EVENTS ADDED TO THE INTERNAL INITIATOR LOGIC MODEL

Event	Description of Event	Parameters					Failure Probability			
		Building	Elevation	Heading	$\frac{V}{A}$	H_c	.125g	.250g	.500g	.750g
<u>Auxiliary Feedwater</u>										
MOAFPS	Seismic failure of both motor driven AFW pumps	SCB	755	E-W	12.07	0.49	0.0	0.0	0.0	0.0
SMIDPCN53	Seismic failure of turbine driven AFW pump	SCB	755	E-W	11.96	0.47	0.0	0.0	0.0	0.0
552TKPWSTS	Seismic failure of PWST	SCB	755	E-W	2.02	0.59	1.19E-06	1.99E-04	8.98E-03	4.65E-02
<u>Buildings</u>										
CB	Seismic failure of the control building	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
CPH	Seismic failure of the circulating water pump house	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
DOB	Seismic failure of the diesel generating building	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
EEB	Seismic failure of the electrical equipment building	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
EPH	Seismic failure of the emergency pump house	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
NIB	Seismic failure of all OHRP buildings	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
RCB	Seismic failure of the reactor containment building	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
RSB	Seismic failure of the reactor service building	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
SCB	Seismic failure of the steam generating building	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
TGB	Seismic failure of the turbine generating building	NA	NA	NA	1.00	0.43	6.56E-07	6.32E-04	5.33E-02	2.52E-01
<u>Direct Heat Removal Service</u>										
LMABHXAS	Seismic failure of air blast heat exchanger A	RSB	755	E-W	0.72	0.54	5.93E-04	2.51E-02	2.50E-01	5.50E-01
LMABHXBS	Seismic failure of air blast heat exchanger B	RSB	755	E-W	0.72	0.54	5.93E-04	2.51E-02	2.50E-01	5.50E-01
ABHX	Seismic failure of both air blast heat exchangers	RSB	755	E-W	0.72	0.54	5.93E-04	2.51E-02	2.50E-01	5.50E-01
EVSCOOLERA	Seismic failure of EVS cooler A	RSB	779	E-W	0.93	0.54	1.01E-04	7.49E-03	1.25E-01	3.45E-01
EVSCOOLERB	Seismic failure of EVS cooler B	RSB	779	E-W	0.93	0.54	1.01E-04	7.49E-03	1.25E-01	3.45E-01
EVSCOOLS	Seismic failure of both EVS coolers	RSB	779	E-W	0.93	0.54	1.01E-04	7.49E-03	1.25E-01	3.45E-01
NREXTANKA	Seismic failure of NaK expansion tank A	RSB	792	E-W	0.98	0.53	5.11E-05	4.98E-03	1.02E-01	3.07E-01
NREXTANKB	Seismic failure of NaK expansion tank B	RSB	792	E-W	0.98	0.53	5.11E-05	4.98E-03	1.02E-01	3.07E-01
NREXTANKC	Seismic failure of both NaK expansion tanks	RSB	792	E-W	0.98	0.53	5.11E-05	4.98E-03	1.02E-01	3.07E-01
DIRSPUMP A	Seismic failure of DIRS pump A	RSB	755	E-W	2.30	0.63	1.91E-06	2.14E-04	7.71E-03	3.76E-02
DIRSPUMP B	Seismic failure of DIRS pump B	RSB	755	E-W	2.30	0.63	1.91E-06	2.14E-04	7.71E-03	3.76E-02
DIRSPUMPS	Seismic failure of 2/2 DIRS pumps	RSB	755	E-W	2.30	0.63	1.91E-06	2.14E-04	7.71E-03	3.76E-02
HMIPES	Seismic failure of UHRP piping	NA	NA	NA	0.62	0.63	5.51E-03	7.47E-02	3.66E-01	6.19E-01
<u>Electric Power System</u>										
EB41003AG	Seismic failure of 4160V bus 3AG	EEB	794	N-S	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01
EB41003BG	Seismic failure of 4160V bus 3BG	EEB	794	N-S	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01
EB41603EG	Seismic failure of 4160V bus 3EG	SCB	794	N-S	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01

Event	Description of Event	Parameters					Failure Probability				
		Building	Elevation	Heading	V A	Bc	.125g	.750g	.500g	.750g	
41A05	Seismic failure of all 4160V buswork	NA	NA	NA	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01	
BATTERY	Seismic failure of battery 10C	CR	765	E-W	2.12	0.60	1.19E-06	1.83E-04	8.03E-03	4.17E-02	
ERAT10A	Seismic failure of battery 10A	CR	765	E-W	2.12	0.60	1.19E-06	1.83E-04	8.03E-03	4.17E-02	
ERAT10B	Seismic failure of battery 10B	CR	765	E-W	2.12	0.60	1.19E-06	1.83E-04	8.03E-03	4.17E-02	
ERAT15	Seismic failure of all batteries	CR	765	E-W	2.12	0.60	1.19E-06	1.83E-04	8.03E-03	4.17E-02	
ERAT25A	Seismic failure of bus 5A	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ERAT25B	Seismic failure of bus 5B	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ERAT25C	Seismic failure of bus 5C	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ERAT25D	Seismic failure of all 125V buswork	NA	NA	NA	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01	
ERAT25E	Seismic failure of EDG fuel oil day tank A	CR	814	VERT	3.64	0.45	0.0	0.0	5.13E-06	2.24E-04	
ERAT25F	Seismic failure of EDG fuel oil day tank B	CR	814	VERT	3.64	0.45	0.0	0.0	5.13E-06	2.24E-04	
ERAT25G	Seismic failure of EDG fuel oil day tank C	CR	814	VERT	3.64	0.45	0.0	0.0	5.13E-06	2.24E-04	
ERAT25H	Seismic failure of 3/3 EDG fuel oil day tanks	CR	814	VERT	3.64	0.45	0.0	0.0	5.13E-06	2.24E-04	
FUELS	Seismic failure of bus RA	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159A	Seismic failure of bus RB	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159B	Seismic failure of bus RC	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159C	Seismic failure of bus RD	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159D	Seismic failure of bus RE	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159E	Seismic failure of bus RF	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159F	Seismic failure of bus RG	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159G	Seismic failure of bus RH	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159H	Seismic failure of bus RI	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159I	Seismic failure of bus RJ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159J	Seismic failure of bus RK	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159K	Seismic failure of bus RL	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159L	Seismic failure of bus RM	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159M	Seismic failure of bus RN	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159N	Seismic failure of bus RO	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159O	Seismic failure of bus RP	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159P	Seismic failure of bus RQ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159Q	Seismic failure of bus RR	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159R	Seismic failure of bus RS	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159S	Seismic failure of bus RT	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159T	Seismic failure of bus RU	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159U	Seismic failure of bus RV	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159V	Seismic failure of bus RW	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159W	Seismic failure of bus RX	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159X	Seismic failure of bus RY	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159Y	Seismic failure of bus RZ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF159Z	Seismic failure of bus RA	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160A	Seismic failure of bus RB	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160B	Seismic failure of bus RC	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160C	Seismic failure of bus RD	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160D	Seismic failure of bus RE	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160E	Seismic failure of bus RF	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160F	Seismic failure of bus RG	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160G	Seismic failure of bus RH	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160H	Seismic failure of bus RI	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160I	Seismic failure of bus RJ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160J	Seismic failure of bus RK	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160K	Seismic failure of bus RL	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160L	Seismic failure of bus RM	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160M	Seismic failure of bus RN	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160N	Seismic failure of bus RO	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160O	Seismic failure of bus RP	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160P	Seismic failure of bus RQ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160Q	Seismic failure of bus RR	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160R	Seismic failure of bus RS	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160S	Seismic failure of bus RT	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160T	Seismic failure of bus RU	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160U	Seismic failure of bus RV	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160V	Seismic failure of bus RW	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160W	Seismic failure of bus RX	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160X	Seismic failure of bus RY	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160Y	Seismic failure of bus RZ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF160Z	Seismic failure of bus RA	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161A	Seismic failure of bus RB	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161B	Seismic failure of bus RC	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161C	Seismic failure of bus RD	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161D	Seismic failure of bus RE	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161E	Seismic failure of bus RF	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161F	Seismic failure of bus RG	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161G	Seismic failure of bus RH	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161H	Seismic failure of bus RI	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161I	Seismic failure of bus RJ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161J	Seismic failure of bus RK	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161K	Seismic failure of bus RL	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161L	Seismic failure of bus RM	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161M	Seismic failure of bus RN	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161N	Seismic failure of bus RO	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161O	Seismic failure of bus RP	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161P	Seismic failure of bus RQ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161Q	Seismic failure of bus RR	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161R	Seismic failure of bus RS	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161S	Seismic failure of bus RT	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161T	Seismic failure of bus RU	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161U	Seismic failure of bus RV	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161V	Seismic failure of bus RW	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161W	Seismic failure of bus RX	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161X	Seismic failure of bus RY	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161Y	Seismic failure of bus RZ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF161Z	Seismic failure of bus RA	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162A	Seismic failure of bus RB	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162B	Seismic failure of bus RC	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162C	Seismic failure of bus RD	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162D	Seismic failure of bus RE	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162E	Seismic failure of bus RF	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162F	Seismic failure of bus RG	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162G	Seismic failure of bus RH	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162H	Seismic failure of bus RI	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162I	Seismic failure of bus RJ	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162J	Seismic failure of bus RK	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162K	Seismic failure of bus RL	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162L	Seismic failure of bus RM	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162M	Seismic failure of bus RN	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ESF162N	Seismic failure of bus RO	CR	765	E-W	1.79	0.59	3.22E-06	4.24E-04	1.5		

Table B-3 (Continued)

Event	Description of Event	Parameters					Failure Probability				
		Building	Elevation	Heading	γ	B_c	.125q	.250q	.500q	.750q	
Electric Power System (Cont.)											
ED4RDV41A	Seismic failure of 480V bus 41A	RSB	755	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV52B	Seismic failure of 480V bus 52B	RSR	733	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV41B	Seismic failure of 480V bus 41B	RSR	733	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV42A	Seismic failure of 480V bus 42A	CR	863	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV42B	Seismic failure of 480V bus 42B	CR	847	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV52A	Seismic failure of 480V bus 52A	SCR	836	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV52A	Seismic failure of 480V bus 52A	PSR	733	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED794EW	Seismic failure of all 480V buswork at 794' with E-W heading	NA	794	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV794	Seismic failure of all 480V buswork with E-W heading	NA	NA	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV794	Seismic failure of all 480V buswork at 794'	NA	794	NA	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED1611	Seismic failure of all 480V buswork	Yard	816	NA	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED1611	Seismic failure of 161 kV bus 1	Yard	816	NA	1.10	0.64	3.39E-04	1.03E-02	1.09E-01	2.75E-01	
ED16129	Seismic failure of 161 kV bus 29	Yard	816	NA	1.10	0.64	3.39E-04	1.03E-02	1.09E-01	2.75E-01	
ED1615	Seismic failure of all 161 kV buses	NA	816	NA	1.10	0.64	3.39E-04	1.03E-02	1.09E-01	2.75E-01	
ED416M5794	Seismic failure of all 4160V buswork at 794' with N-S heading	NA	794	N-S	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01	
ED416M35G	Seismic failure of 4160V bus 35	DCR	816	E-W	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01	
ED416M35G	Seismic failure of 4160V bus 35	DCR	816	E-W	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01	
ED416M35G	Seismic failure of 4160V bus 31	DCR	816	E-W	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01	
ED416M35G	Seismic failure of all 4160V buswork at 816'	NA	816	NA	1.81	0.84	7.32E-04	9.22E-03	6.28E-02	1.47E-01	
ED416M35G	Seismic failure of transformer 5A	Yard	816	NA	1.10	0.64	3.39E-04	1.03E-02	1.09E-01	2.75E-01	
ED416M35G	Seismic failure of transformer 5B	Yard	816	NA	1.10	0.64	3.39E-04	1.03E-02	1.09E-01	2.75E-01	
ED416M35G	Seismic failure of transformer 6A	Yard	816	NA	1.10	0.64	3.39E-04	1.03E-02	1.09E-01	2.75E-01	
ED416M35G	Seismic failure of transformer 6B	Yard	816	NA	1.10	0.64	3.39E-04	1.03E-02	1.09E-01	2.75E-01	
ED416M35G	Seismic failure of all transformer at 816'	NA	816	NA	1.10	0.64	3.39E-04	1.03E-02	1.09E-01	2.75E-01	
ED4RDV58A	Seismic failure of 480V bus 58A	RCB	733	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV58B	Seismic failure of 480V bus 58B	RCB	733	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV58C	Seismic failure of 480V bus 58C	RCB	836	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV58D	Seismic failure of 480V bus 58D	RCB	836	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED768EW	Seismic failure of all 480V buswork at 768' with E-W heading	NA	768	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV768	Seismic failure of all 480V buswork at 768'	NA	768	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV768	Seismic failure of all 480V buswork at 836' with E-W heading	NA	836	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV768	Seismic failure of all 480V buswork at 836'	NA	836	NA	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	
ED4RDV755	Seismic failure of all 480V buswork at 755' with E-W heading	NA	755	E-W	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.02E-02	

Table B-3 (Continued)

Event	Description of Event	Parameters					Failure Probability			
		Building	Elevation	Heading	γ A	B _c	.125g	.250g	.500g	.750g
<u>Electric Power System (Cont.)</u>										
EBAR0755	Seismic failure of all 480V buswork at 755'	NA	755	NA	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
EB755NS	Seismic failure of all 480V buswork at 755' with N-S heading	NA	755	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
EB480NS	Seismic failure of all 480V buswork with N-S heading	NA	NA	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
EBAR0733	Seismic failure of all 480V buswork at 733'	NA	733	NA	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
EB863NS	Seismic failure of all 480V buswork at 863' with N-S heading	NA	863	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
EBAR0863	Seismic failure of all 480V buswork at 863'	NA	863	NA	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
EB847NS	Seismic failure of all 480V buswork at 847' with N-S heading	NA	847	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
EBAR0847	Seismic failure of all 480V buswork at 847'	NA	847	NA	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
EB836NS	Seismic failure of all 480V buswork at 836' with N-S heading	NA	836	N-S	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
EBAR0836	Seismic failure of all 480V buswork at 836'	NA	836	NA	1.79	0.59	3.22E-06	4.24E-04	1.53E-02	7.07E-02
<u>Emergency Chilled Water</u>										
ECW01MFA	Seismic failure of ECW pump A	SGR	733	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
ECW01MFB	Seismic failure of ECW pump B	SGR	733	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
ECW02OLA	Seismic failure of ECW chiller A	SGR	733	E-W	8.41	0.42	0.0	0.0	0.0	0.0
ECW02OLB	Seismic failure of ECW chiller B	SGR	733	E-W	8.41	0.42	0.0	0.0	0.0	0.0
ECW01MFC	Seismic failure of both ECW pumps	SGR	733	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
ECW02OLS	Seismic failure of both ECW HXS	SGR	733	E-W	8.41	0.42	0.0	0.0	0.0	0.0
<u>Emergency Plant Service Water</u>										
EPSP01MFA	Seismic failure of EPSW pump A	EPH	823	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
EPSP01MFB	Seismic failure of EPSW pump B	EPH	823	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
EPSP01MFC	Seismic failure of EPSW pump C	EPH	823	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
EPSP01MFS	Seismic failure of 3/3 EPSW pumps	EPH	823	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
<u>Heat Rejection</u>										
HR01MFA	Seismic failure of circulating water pump A	CPH	818	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
HR01MFB	Seismic failure of circulating water pump B	CPH	818	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
HR01MFC	Seismic failure of circulating water pump C	CPH	818	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
HR01MFS	Seismic failure of 3/3 circulating water pumps	CPH	818	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02

Table B-3 (Continued)

Event	Description of Event	Parameters					Failure Probability			
		Building	Elevation	Heading	V A	B _C	.125q	.250q	.500q	.750q
<u>Heat Transport</u>										
<u>PHYSICS</u>										
<u>Main Feedwater and Condensate</u>										
CCSPM1A	Seismic failure of a all IHTS and PHTS Pumps	NA	NA	NA	3.91	0.52	0.0	5.96E-08	3.83E-05	7.48E-04
CCSPM1B	Seismic failure of condensate system pump A	TGB	816	Vert	6.61	0.57	0.0	0.0	2.98E-06	6.73E-05
CCSPM1C	Seismic failure of condensate system pump B	TGB	816	Vert	6.61	0.57	0.0	0.0	2.98E-06	6.73E-05
CCSPM1D	Seismic failure of condensate system pump C	TGB	816	Vert	6.61	0.57	0.0	0.0	2.98E-06	6.73E-05
CCSPM1E	Seismic failure of 3/3 condensate system pumps	TGB	816	Vert	6.61	0.57	0.0	0.0	2.98E-06	6.73E-05
MCWPM1A	Seismic failure of main feedwater pump A	TGB	816	E-W	6.61	0.57	0.0	0.0	2.98E-06	6.73E-05
MCWPM1B	Seismic failure of main feedwater pump B	TGB	816	E-W	6.61	0.57	0.0	0.0	2.98E-06	6.73E-05
MCWPM1C	Seismic failure of main feedwater pump C	TGB	816	E-W	6.61	0.57	0.0	0.0	2.98E-06	6.73E-05
MCWPM1D	Seismic failure of 3/3 main feedwater pumps	TGB	816	E-W	6.61	0.57	0.0	0.0	2.98E-06	6.73E-05
<u>Normal Chilled Water</u>										
NCHW1A	Seismic failure of NCHW chiller A	RCB	816	N-S	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1B	Seismic failure of NCHW chiller B	RCB	794	E-W	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1C	Seismic failure of NCHW chiller C	RCB	816	N-S	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1D	Seismic failure of NCHW chiller D	RCB	794	E-W	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1E	Seismic failure of NCHW chiller E	RCB	794	E-W	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1F	Seismic failure of NCHW chiller F	RCB	794	N-S	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1G	Seismic failure of NCHW pump A	RCB	816	N-S	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
NCHW1H	Seismic failure of NCHW pump B	RCB	794	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
NCHW1I	Seismic failure of NCHW pump C	RCB	816	N-S	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
NCHW1J	Seismic failure of NCHW pump D	RCB	794	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
NCHW1K	Seismic failure of NCHW pump E	RCB	794	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
NCHW1L	Seismic failure of NCHW pump F	RCB	794	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
NCHW1M	Seismic failure of 6/6 NCHW chillers	RCB	NA	NA	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1N	Seismic failure of all NCHW chillers with N-S heading	RCB	NA	N-S	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1O	Seismic failure of all NCHW chillers with E-W heading	RCB	NA	E-W	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1P	Seismic failure of all NCHW chillers with E-W heading 794'	RCB	794	NA	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1Q	Seismic failure of all NCHW chillers & elevation 794'	RCB	794	E-W	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1R	Seismic failure of all NCHW chillers & elevation 794' with E-W heading	RCB	794	E-W	8.41	0.42	0.0	0.0	0.0	0.0
NCHW1S	Seismic failure of 6/6 NCHW pumps	RCB	NA	NA	2.19	0.51	0.0	0.0	1.04E-05	1.89E-03
NCHW1T	Seismic failure of all NCHW pumps with N-S heading	RCB	NA	N-S	2.19	0.51	0.0	0.0	1.04E-05	1.89E-03
NCHW1U	Seismic failure of all NCHW pumps with E-W heading	RCB	NA	E-W	2.19	0.51	0.0	0.0	1.04E-05	1.89E-03
NCHW1V	Seismic failure of all NCHW pumps at Elevation 794'	RCB	794	NA	2.19	0.51	0.0	0.0	1.04E-05	1.89E-03
NCHW1W	Seismic failure of all NCHW pumps at Elevation 794' with E-W heading	RCB	794	E-W	2.19	0.51	0.0	0.0	1.04E-05	1.89E-03

Table B-3 (Continued)

Event	Description of Event	Parameters					Failure Probability			
		Building	Elevation	Heading	γ	Bc	.125g	.250g	.500g	.750g
<u>Normal Plant Service Water</u>										
NPSWMPMPA	Seismic failure of NPSW pump A	CPH	B18	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
NPSWMPMPB	Seismic failure of NPSW pump B	CPH	B18	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
NPSWMPMS	Seismic failure of 2/2 NPSW pumps	CPH	B18	Vert	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
<u>Reactor Protection</u>										
LOGICI	Seismic failure of SGARS logic panel I	CB	B16	N-S	4.58	0.78	1.97E-06	9.64E-05	2.26E-03	1.02E-02
LOGICII	Seismic failure of SGARS logic panel II	CB	B16	N-S	4.58	0.78	1.97E-06	9.64E-05	2.26E-03	1.02E-02
LOGICIII	Seismic failure of SGARS logic panel III	CB	B16	N-S	4.58	0.78	1.97E-06	9.64E-05	2.26E-03	1.02E-02
LOGICS	Seismic failure of 3/3 SGARS logic panels	CB	B16	N-S	4.58	0.78	1.97E-06	9.64E-05	2.26E-03	1.02E-02
<u>Recirculating Gas Cooling System</u>										
RGC10XCOOL	Seismic failure of RGCS cooler for Cell 103 (Subsystem 280a)	RCB	752	N-S	2.50	0.55	0.0	1.42E-05	1.72E-03	1.43E-02
RGC104COOL	Seismic failure of RGCS cooler for Cell 104 (Subsystem 290a)	RCB	752	N-S	2.50	0.55	0.0	1.42E-05	1.72E-03	1.43E-02
RGCS	Seismic failure of RGCS coolers serving Cells 103 and 104	RCB	752	N-S	2.50	0.55	0.0	1.42E-05	1.72E-03	1.43E-02
<u>Secondary Services Closed Cooling Water</u>										
SSCOWMPA	Seismic failure of SSCOW pump A	TGB	B16	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
SSCOWMPB	Seismic failure of SSCOW pump B	TGB	B16	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
SSCOWMPS	Seismic failure of 2/2 SSCOW pumps	TGB	B16	E-W	2.19	0.51	0.0	1.04E-05	1.89E-03	1.78E-02
SSCOWCOOLA	Seismic failure of SSCOWHX A	TGB	B16	N-S	8.41	0.42	0.0	0.0	0.0	0.0
SSCOWCOOLB	Seismic failure of SSCOWHX B	TGB	B16	N-S	8.41	0.42	0.0	0.0	0.0	0.0
SSCOWCOOLS	Seismic failure of 2/2 SSCOW HXS	TGB	B16	N-S	8.41	0.42	0.0	0.0	0.0	0.0
<u>Steam Generating System</u>										
SGSEX11A	Seismic failure of superheater A	SGR	R06	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSEX11B	Seismic failure of superheater B	SGR	R06	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSEX11C	Seismic failure of superheater C	SGR	R06	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSEX11S	Seismic failure of 3/3 superheaters	SGR	R06	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSEVAPA	Seismic failure of east evaporator A	SGR	R06	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSEVAPB	Seismic failure of east evaporator B	SGR	R06	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSEVAPC	Seismic failure of east evaporator C	SGR	R06	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSWAPA	Seismic failure of west evaporator A	SGR	R06	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01

Table B-3 (Continued)

Event	Description of Event	Parameters					Failure Probability			
		Building	Elevation	Heading	γ	B _c	.125g	.250g	.500g	.750g
SGSVAPEB	Seismic failure of west evaporator B	SCB	806	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSVAPEC	Seismic failure of west evaporator C	SCB	806	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSEVAPS	Seismic failure of 6/6 evaporators	SCB	806	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
SGSMODULES	Seismic failure of 9/9 SGS superheaters and evaporators	SCB	806	N-S	1.03	0.52	2.50E-05	3.24E-03	8.23E-02	2.71E-01
<u>Top Logic</u>										
Event K	Mechanical binding of control rods due to an earthquake	NA	NA	NA	1.17	0.47	950E-07	5.17E-04	3.52E-02	1.72E-01
TIRGVLEAK	Seismic failure of a guard vessel	NA	NA	NA	0.40	0.50	1.00E-02	1.74E-01	6.72E-01	8.96E-01
TY	Seismic failure of the electric power system ceramic insulators	NA	NA	NA	0.20	0.32	7.09E-02	7.57E-01	9.98E-01	1.00E-00
T7*	Seismic failure of steam generator tubes	NA	NA	NA	3.51	0.50	0.0	5.96E-08	4.86E-05	1.01E-03
T8*	Seismic failure of SWGRS rupture disks	NA	NA	NA	0.57	0.50	1.20E-03	4.96E-02	3.97E-01	7.08E-01
T9*	Seismic failure of RHIS piping	NA	NA	NA	1.93	0.63	6.97E-06	5.89E-04	1.60E-02	6.68E-02
T10*	Seismically induced RHIS leak inside a loop guard vessel	NA	NA	NA	0.69	0.63	335E-03	5.35E-02	3.09E-01	5.53E-01
T10R	Seismically induced leak inside reactor vessel guard vessel	NA	NA	NA	1.43	0.63	5.48E-05	2.87E-03	4.77E-02	1.53E-01
T11*	Seismic failure of RHIS piping which disables both SGS and DRS cooling	NA	NA	NA	0.98	0.63	5.40E-04	1.51E-02	1.43E-01	3.56E-01
T12*	Seismic failure of RHIS piping which disables SGS cooling but allows DRS operation	NA	NA	NA	1.57	0.63	2.99E-05	1.77E-03	3.47E-02	1.20E-01
T14*	Seismic failure of steam piping	NA	NA	NA	4.52	0.66	0.0	5.76E-06	4.25E-04	3.25E-03
T21*	Seismic failure of RHIS piping which disables RHIS but allows SGS operation	NA	NA	NA	0.64	0.63	4.77E-03	6.78E-02	5.48E-01	5.99E-01
T22*	Seismic failure of RHIS piping which allows both DRS and SGS operation	NA	NA	NA	1.93	0.63	6.97E-06	5.89E-04	1.60E-02	6.68E-02

*Event prefix only. This event is a loop dependent qualifier (i.e., A,B,C for loop 1,2,3 respectively not shown)

VITA

Martin Allen Stutzke was born in Louisville, Kentucky on July 3, 1954. He attended elementary and secondary schools in that city and was graduated from Iroquois High School in June 1972. The following September, he entered The University of Tennessee, Knoxville; in June 1976, he received a Bachelor of Science degree in Nuclear Engineering.

In December 1976, he was commissioned as an Ensign in the United States Navy. Stationed onboard the USS Enterprise, he served as the ship's Electrical Division Officer, Reactor Training Assistant, and Aft Propulsion Group Station Officer. He resigned his commission in January 1981 to accept a position as technical advisor to the Operations Division at Oak Ridge National Laboratory. Since February 1982, he has been employed by Technology for Energy Corporation.

In July 1981, he entered The Graduate School at The University of Tennessee. He received the Master of Science degree with a major in Nuclear Engineering in December 1984.

The author is a member of Tau Beta Pi and the American Nuclear Society.