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Three Mile Island Unit 1
Probabilistic Risk Assessment

ENVIRONMENTAL AND EXTERNAL HAZARDS REPORT

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SUMMARY OF CONTENTS

CONTENTS

CONTENTS (continued)

CONTENTS (continued)

BOOK 2 OF 2

LIST OF TABLES

Table Page

BOOK 1 OF 2

LIST OF TABLES (continued)

LIST OF FIGURES

BOOK 1 OF 2

LIST OF ACRONYMS

Abbreviation	Definition
ACR	air-cooled reactor
ADV	atmospheric dump valve
AEC	U.S. Atomic Energy Commission
AOV	air-operated valve
ATOG	abnormal transient operational guidelines
ATWS	anticipated transient without scram
B&W	Babcock & Wilcox Company
BOP	balance of plant
BRP	Big Rock Point
Btu	British thermal unit
BWR	boiling water reactor
BWST	borated water storage tank
CAR	corrective action report
CARS	condenser air removal system
CAS	chemical addition system
CBVS	control building ventilation system
CCF	common cause failure
CDF	cumulative distribution function
CFT	core flooding tank
CIV	containment isolation valve
CSF	conditional split fraction
CST	condensate storage tank
CRO	control room operator
CWS	circulating water system
DHCCW	decay heat closed cooling water
DHR	decay heat removal
DHRS	decay heat removal system
DHRW	decay heat river water
DPD	discrete probability distribution
EFW	emergency feedwater
EEHR	Environmental and External Hazards Report
EHC	electrohydraulic control
EOF	emergency operations facility
EPRI	Electric Power Research Institute
ESD	event sequence diagram
ESAS	engineered safeguards actuation system
ETC	event tree code
FAA	Federal Aviation Administration
FHA	fire hazards analysis
FSAR	Final Safety Analysis Report
FTAP	Fault Tree Analysis Program

LIST OF ACRONYMS (continued)

Abbreviation	Definition
GCR	gas-cooled reactor
GE	General Electric Company
GPUN	GPU Nuclear Corporation
HCR	human cognitive reliability
HCLPF	high confidence low probability of failure
HIA	Harrisburg International Airport
HPI	high pressure injection
HPIS	high pressure injection system
HSPS	heat sink protection system
HTM	high trajectory missile
HVAC	heating, ventilating, and air conditioning
ICCS	intermediate closed cooling system
ICCW	intermediate closed cooling water
ICS	integrated control system
IREP	Interim Reliability Evaluation Program
LBIS	line break isolation system
LCO	limiting condition for operation
LER	Licensee Event Report
LOCA	loss of coolant accident
LOFW	loss of main feedwater
LONS	loss of nuclear services
LORI	loss of reactor coolant system inventory
LORW	loss of river water
LOSP	loss of offsite power
LPI	low pressure injection
LPIS	low pressure injection system
LSS	low speed stop
LTM	low trajectory missile
MCC	motor control center
MFPT	main feedwater pump trip
MFW	main feedwater
MGL	multiple Greek letter
MOV	motor-operated valve
MSIV	main steam isolation valve
MSLB	main steam line break
MSS	main steam system
MSSV	main steam safety valve
MSV	main steam valve
MUP	makeup and purification
MUT	makeup tank

LIST OF ACRONYMS (continued)

Abbreviation	Definition
NPE	Nuclear Power Experience
NRC	U.S. Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSCCS	nuclear services closed cooling system
NSCCW	nuclear services closed cooling water
NSRW	nuclear services river water
NSSS	nuclear steam supply system
NTSB	National Transportation Safety Board
NUS	NUS Corporation
OPM	Operations Plant Manual
OTSG	once-through steam generator
P&ID	piping and instrumentation drawing
PCL	panel center left
PCR	panel center right
PDF	probability density function
PDS	plant damage state
PLF	panel left front
PLG	Pickard, Lowe and Garrick, Inc.
PMF	probable maximum flood
PMR	Plant Model Report
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PRF	panel right front
PSHX	primary to secondary heat transfer
PSV	pressurizer safety valve
PWR	pressurized water reactor
RBCU	reactor building cooler unit
RBEC	reactor building emergency cooling
RBD	reliability block diagram
RBS	reactor building spray
RBSS	reactor building spray system
RCDT	reactor coolant drain tank
RCP	reactor coolant pump
RCS	reactor coolant system
RPS	reactor protection system
RSS	Reactor Safety Study
RSSM	Reactor Safety Study Methodology Application Program
SAI	Science Applications, Inc.
SCCW	secondary closed cooling water
SCM	subcooled margin
SGTR	steam generator tube rupture
SLB	steam line break
SLRDS	steam line rupture detection system
SRO	senior reactor operator

LIST OF ACRONYMS (continued)

Abbreviation	Definition
SRV	safety relief valve
SRW	secondary river water
SSCCS	secondary services closed cooling system
SSE	safe shutdown earthquake
SSS	support state system
STA	shift technical advisor
TBV	turbine bypass valve
TLV	toxic limit valve
TMI-1	Three Mile Island Nuclear Generating Station, Unit 1
TMI-2	Three Mile Island Nuclear Generating Station, Unit 2
TPRA	Three Mile Island Probabilistic Risk Assessment
TVA	Tennessee Valley Authority
UCS	The Union of Concerned Scientists
ULD	unit load demand

1. INTRODUCTION

A probabilistic evaluation of the impact of environmental and external hazards on Three Mile Island Unit 1 is presented in this report. This study was performed as part of the TMI-1 probabilistic risk assessment. By environmental hazards, we mean those equipment failure causes whose sources are within the plant boundaries and may simultaneously affect several components. Examples are fire, internal flood, and steam. By external hazards, we mean those buildings and equipment failure causes whose sources are outside the plant boundaries. Examples are earthquake, external flood, and aircraft crash. One external hazard, loss of river water (principally due to screen clogging), is included in the Plant Model Report.

A long list of environmental and external hazards were considered for the TMI-1 PRA. Most of this list came from a compiled list found in the PRA Procedures Guide.* Many hazards from that list were judged to be of little significance or relevance to TMI-1 and, therefore, are not analyzed further. Table 1-1 gives this list of hazards and summarizes the reasons for including in or excluding from this analysis.

This report is divided into 8 sections. Each of the following seven sections is dedicated to one type of environmental hazard, except for Section 3. Section 3 addresses environmental hazards that may potentially be generated within the plant. This analysis is called "spatial interaction analysis." Examples of this type of hazard are fire, flood from internal sources, steam, and high energy pipe movement. The analysis of missiles from the main turbine-generator set is treated separately in Section 6. The title of each section is listed below:

Section Number	Title
1	Introduction
2	Seismic Analysis
3	Analysis of Spatial Interactions
4	Analysis of Flooding from External Sources
5	Analysis of Extreme Weather Phenomena
6	Turbine Missile Analysis
7	Aircraft Crash Analysis
8	Hazardous Chemicals Analysis

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*American Nuclear Society and Institute of Electrical and Electronics Engineers, "PRA Procedures Guide; A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," sponsored by the U.S. Nuclear Regulatory Commission and the Electric Power Research Institute, NUREG/CR-2300, April 1983.

TABLE 1-1. ENVIRONMENTAL AND EXTERNAL HAZARDS CONSIDERED FOR TMI-1

Sheet 1 of 4

Hazard Type	Source Exists	Included in this Analysis	Remarks
Aircraft Impact	Yes	Yes	See Section 7.
Avalanche	No	--	No nearby hills or mountains.
Coastal Erosion	Yes	No	Very slow process; long lead time to put plant in cold shutdown.
Drought	Yes	No	Very slow process; long lead time to put plant in cold shutdown.
Explosion (internal)	Yes	Yes	See Section 3.
External Flooding	Yes	Yes	See Section 4.
Extreme Winds and Tornadoes	Yes	Yes	See Section 5.
Falling Objects	Yes	Yes	See Section 3.
Fire	Yes	Yes	See Section 3.
Fog	Yes	No	Indirect impact of fog, such as impact on aircraft crash frequency, is addressed as part of other hazards.
Forest Fire	Yes	No	Plant is on an island on Susquehanna river; a fire involving the vegetation on the island or on the mainland is judged to only threaten the offsite power. This scenario is included in the loss of offsite power frequency evaluation.
Frost	Yes	No	Impact of frost on diesel generator availability is treated as part of diesel generator failure data. Impact of frost on transmission lines is included in the loss of offsite power frequency. Impact of frost on screen house water

TABLE 1-1 (continued)

Sheet 2 of 4

Hazard Type	Source Exists	Included in this Analysis	Remarks
Frost (continued)			availability is included in loss of river water data analysis (see Data Analysis Report).
Hail	Yes	No	Impact of hail on offsite power is included in the frequency of loss of offsite power analysis. Contribution to the overall risk is judged to be negligible.
High Tide, or High Lake Level	No	No	--
High River Stage	Yes	Yes	Same as external flooding, see Section 4.
High Energy Line Break	Yes	Yes	See Section 3.
High Summer Temperature	Yes	No	The impact of high temperature environment on equipment performance is included in equipment failure data.
Hurricane	Yes	Yes	See Section 5.
Ice Cover	Yes	No	See discussion on frost.
Industrial or Military Facility Accident	Yes	Yes	See Section 7.
Internal Flooding	Yes	Yes	See Section 3.
Jets (water)	Yes	Yes	See Section 3.
Landslide	No	--	--
Lightning	Yes	No	Plant is equipped with lightning protection. Contribution to the overall risk judged to be negligible. Impact on offsite power included in loss of offsite power frequency evaluation.
Low Lake or River Water Level	Yes	No	Included in loss of river water frequency evaluation.

TABLE 1-1 (continued)

Sheet 3 of 4

Hazard Type	Source Exists	Included in this Analysis	Remarks
Low winter temperature	Yes	No	Impact on equipment has been included through component (independent and common cause) failure rates.
Meteorite	Yes	No	Likelihood of occurrence is very small.
Missiles (internal)	Yes	Yes	See Section 3.
Pipeline Accident (gas, etc.)	Yes	Yes	See Section 8.
Intense Precipitation	Yes	Yes	See Section 4.
Release of Chemicals in Onsite Storage	Yes	Yes	See Section 8.
River Diversion	No	No	Intake screen blockage is part of loss of river water frequency evaluation.
Sandstorm	No	No	--
Seiche	No	No	--
Seismic activity	Yes	Yes	See Section 2.
Snow	Yes	No	Included in external flood analysis and loss of offsite power frequency evaluation.
Soil Shrink-Swell Consolidation	Yes	No	Very slow process.
Smoke	Yes	Yes	See Section 3.
Spray (water)	Yes	Yes	See Section 3.
Steam	Yes	Yes	See Section 3.
Storm Surge	Yes	Yes	See Section 4.
Transportation Accidents	Yes	Yes	See Section 8.
Tsunami	No	No	--

TABLE 1-1 (continued)

Sheet 4 of 4

Hazard Type	Source Exists	Included in this Analysis	Remarks
Toxic Gas	Yes	Yes	See Section 8.
Turbine-Generated Missile	Yes	Yes	See Section 6.
Volcanic Activity	No	No	No volcanic mountains nearby.
Waves	No	No	River cannot generate tall waves.

2. SEISMIC EVENTS

This section describes the analysis of potential seismically initiated events at the plant site. A seismic risk analysis consists of five main steps:

1. Seismicity Analysis. Determination of the frequency of ground motions of various sizes at the site.
2. Fragility Analysis. Determination of the seismically initiated ground acceleration at which plant structures and components are predicted to fail.
3. Plant Logic Analysis. Development of a logic model that includes the seismically induced events that may cause one or more different classes of initiating events and one or more failures of components or systems needed to respond to the initiating event as well as the consideration of nonseisimic failures that can combine with seismically induced failures to produce an accident sequence.
4. Initial Assembly. Quantification and assembly of the seismicity, component fragility, and plant logic to obtain point estimates of the frequencies of core melt and various plant damage states that might result from seismic initiating events.
5. Final Assembly. After comparing point estimates of plant damage state frequencies from other initiators with those for seismically initiated scenarios that are major frequency contributors, calculation of the probability distribution of plant damage state frequencies and combining the results with the probability distribution of frequencies from other initiating events.

2.1 SEISMIC HAZARD

A site seismic hazard study was performed by Risk Engineering, Inc., a subcontractor on this project, and is incorporated in Appendix A. A summary of the analysis techniques and results from that study are presented here.

Earthquake motions to which structures and equipment might be subjected can be characterized by a single parameter, the peak ground acceleration, a. Structures and equipment require several cycles of strong acceleration in order to develop damaging motions. Low magnitude earthquakes often do not contain sufficient energy or duration to generate several such cycles. Therefore, in order to correlate structure and equipment fragilities with damaging ground accelerations, it is necessary to differentiate between instrumental peak acceleration and the sustained based peak acceleration that encompasses at least several cycles of motion. The seismic hazard analysis presents the likelihood of peak ground accelerations in terms of their annual exceedance frequencies, $\phi(a)$; i.e., the frequency of exceeding various accelerations. Multiple curves reflect the uncertainty in the seismicity and result from the generation of different seismologic hypotheses as described below.

The first step in the seismic hazard analysis is to delineate zones of potential earthquake occurrences using seismicity, geology, and tectonic evidence. Then, for each zone, data on historical earthquake occurrences are gathered, earthquake magnitudes are determined from prior measures or are estimated from earthquake intensities, and the number of earthquakes per unit of time occurring in specific magnitude intervals is determined. The third step is to adopt an attenuation function which estimates peak acceleration as a function of earthquake magnitude and distance between the source and site. Finally, an integration is made over all possible earthquake magnitudes and locations to obtain the annual frequencies that various levels of acceleration will be exceeded.

Seven sets of seismogenic zones were examined in the study, based on NUREG/CR-3756, with each set representing one hypothesis. Equal weight was assigned to each set to reflect the likelihood of it being the correct one for describing the seismic hazard.

For statistical data analysis, earthquakes identified with an epicentral Modified Mercalli (MM) intensity, I_e, but without a magnitude estimate were converted to a body-wave magnitude, m_b, using the relation

$$m_b = 1.75 + 0.5I_e \tag{2.1}$$

Equation (2.1) was derived for the central United States and is considered reliable for the eastern region as well.

The annual number, n, of earthquakes equal to or greater than earthquakes of body wave magnitude, m_b, was determined from the expression

$$\log_{10} n(m_b) = a - bm_b \tag{2.2}$$

where $n(m_b)$ is the annual number of earthquakes of body-wave magnitude m_b and a and b are parameters fit to seismicity data.

The rate of earthquake occurrence was determined for each seismogenic zone using the historical information in that zone.

The best estimate of maximum possible magnitude, $m_{b,max}$, in each zone was taken to be about one MM intensity, or about 0.5 magnitude units, above the maximum historical value in the zone. Alternative values of ± 0.5 magnitude units from the best estimate were examined to represent the uncertainty in this value, with the alternative values assigned a weight of 0.3 each, and the best estimate assigned 0.4.

Four hypotheses were used to estimate peak horizontal ground acceleration at the site as a function of magnitude or moment magnitude, and distance from the epicenter. These are described in Appendix A. Each of the three main hypothesis was assigned equal weight and one of these, in turn, divided in half. In each case, a lognormal distribution of attenuation about the mean value was assumed. The distribution of peak ground acceleration was truncated to reflect the notion that small or moderate earthquakes can only cause a limited amount of damage to real structures.

Considering the variations described above, 756 seismic hazard curves were generated (7 sets of seismogenic zones, 4 attenuation functions, 3 activity rates, 3 values of $m_{b\ max}$, and 3 b-values). These were aggregated into 10 representative curves, which are seen in Figure 2-1. Shown also is the confidence level, or probability, for each curve, where the weight of each curve is the sum of the weights of the contributing curves.

For point estimate computations, mean frequencies were determined from the weighted seismic hazard curves (Figure 2-1) for peak ground acceleration intervals corresponding to discrete accelerations up to 0.6g where the frequency is extremely low and of no practical significance. Actually, the frequency of each discrete acceleration is determined by the difference in frequency at accelerations half-way to the next discrete value and therefore considers the frequency of accelerations greater than 0.6g. Table 2-1 shows the mean values which use will be described later.

2.2 FRAGILITY

A seismic fragility or failure vulnerability analysis was conducted by Structural Mechanics Associates, Inc. (currently NTS Engineering), and is included in Appendix B. The approach adopted in assigning peak ground acceleration capacities to safety related structures, equipment, and other components was to first determine the median factor of safety against failure and its statistical variability under the safe shutdown earthquake. From this safety factor and variability, the median ground acceleration capacity and its variability were determined. For nonsafety systems, capacities were calculated and then keyed back to the SSE for results presentation. For the TMI site, the SSE ground motion used in design of the facility was 0.12g free field peak ground acceleration.

In general, the factor of safety against failure of a structure or component from seismically initiated ground motion can be defined as the ratio of the ground motion causing failure to the maximum ground motion used in design to maintain acceptable elastic stress limits in the component's materials. The overall safety factor was determined by evaluating the factors for a number of parameters which fell into two categories: capacity and response. For structures, parameters influencing the factor on structural capacity are the strength of the structure compared to the design stress level, and the inelastic energy absorption capacity (ductility) of a structure to carry load beyond yield and the earthquake duration to account for the expected duration compared to that assumed in determining the energy absorption factor. The most significant parameters for response to a given ground acceleration include:

1. The response spectra required to be used in the design compared to a median centered spectra more typical for the site.

2. Energy dissipation (damping).

3. Methods for combining dynamic response modes.

4. Combination of earthquake components.

5. Modeling accuracy.

6. Soil-structure interaction effects.

The overall safety factor for equipment and other plant components is derived from similar factors for the component. However, their response also depends on the building in which they are located and their location within the building. Therefore, the overall safety factor for components is made up of component strength capacity relative to the floor acceleration, earthquake duration, component response, and building response that resulted in the floor spectra used in the component design.

A best estimate of limiting value for each parameter was established as being the median of a distribution of possible values. The ratio between that value and the value used in the plant design (by analysis or qualification test) was determined for each critical plant building, equipment, or component and represents the safety factor. A combination of generic and plant specific information was used for these estimates.

The derivation of each factor considered variability. Section 2.2.2 discusses how, in each case, a median safety factor was assigned along with a variability. When combining these median factors for contributing parameters, variabilities were also combined to define the variability in overall safety factor. From this overall safety factor, the median acceleration capacity, or peak ground acceleration at failure, was determined by multiplying the safety factor by 0.12g, the SSE ground motion.

2.2.1 DEFINITION OF FAILURE

For purposes of this study, seismic Category I structures are considered to have failed when inelastic deformations of the structure under seismic load potentially interfere with the operability of equipment attached to the structure. These limits on inelastic energy absorption capacity (ductility limits) are estimated to correspond to the onset of significant structural damage, not necessarily structure collapse.

Piping, electrical, mechanical, and electromechanical equipment vital to mitigating the effects of earthquakes are considered to fail when they can no longer perform their designated functions. Also, ruptures of pressure boundaries are considered failures. In most cases, however, the equipment will lose its ability to function at lower accelerations before pressure boundaries fail because these pressure boundaries for equipment such as pumps and valves are usually very conservatively designed.

2.2.2 FRAGILITY CURVE FORMULATION

Seismic-induced failure data are generally unavailable for specific plant components or structures. Thus, fragility curves which plot the peak ground acceleration at which the component is expected to fail must be developed primarily from analysis and engineering judgment supported by limited test data. Such fragility curves will contain a good deal of uncertainty; therefore, great precision in attempting to define the shape of these curves is impossible to attain.

Earthquakes causing the same peak ground acceleration at the plant site can have different energy contents and durations. These factors vary randomly and affect the fragility of structures and components. So, while the median acceleration capacity can be determined for structures and components, even if their strengths and responses are well known, it is still necessary to assign a random variability to the capacity just due to differences in earthquake characteristics. In addition, the strengths and response characteristics of the structures and components are not exactly known, so our uncertainty about these also needs to be expressed. The median acceleration capacity, random variability, and uncertainty can be expressed by $\tilde{\tilde{a}}$, ε_R, and ε_U, respectively. Then, the acceleration capacity, a, at designated levels of confidence is given by

$$a = \tilde{\tilde{a}}\,\varepsilon_R\varepsilon_U \tag{2.3}$$

As discussed in Appendix B, the statistical variations of many material properties and seismic response variables are represented as well by logarithmic as by other distributions. Therefore, it is assumed that both ε_R and ε_U are lognormally distributed with logarithmic standard deviations of β_R and β_U, respectively. This representation is believed to be inappropriate near the tails of the distributions because experience tells us that there are practical lower and upper bounds on capacity. (This will be discussed shortly.) The random variability, ε_R, about a median acceleration capacity can be expressed by

$$\varepsilon_R = \exp\,(f \cdot \beta_R) \tag{2.4}$$

where f is the standardized Gaussian random variable.

Then

$$a = \tilde{a} \cdot \exp\,(f \cdot \beta_R) \tag{2.5}$$

where a is any acceleration capacity on a curve and $\tilde{a}$ is the median on that curve. This distribution is seen for each of the curves shown in Figure 2-2. The uncertainty variability, ε_U, about the median is expressed by

$$\varepsilon_U = \exp\,(f' \cdot \beta_U) \tag{2.6}$$

and

$$\tilde{a} = \tilde{\tilde{a}} \cdot \exp\,(f' \cdot \beta_U) \tag{2.7}$$

where $\tilde{\tilde{a}}$ is effectively the median of the distribution on acceleration capacity. Therefore, Equation (2.3) becomes

$$a = \tilde{\tilde{a}}\,\exp\,(f' \cdot \beta_U) \cdot \exp\,(f \cdot \beta_R) \tag{2.8}$$

which for f' = 0 becomes the median curve.

The solid curve in Figure 2-2 is the median fragility curve. The 5th and 95th percentile curves are also shown, as indicated by the left and right dashed curves in the figure, respectively, reflecting the uncertainty in the median curve. These percentiles indicate the level of confidence that for a given failure fraction, F, of earthquakes, the component will fail at accelerations greater than indicated by the curve. There actually exists a family of curves representing designated cumulative percentiles of confidence. Figure 2-2 therefore can be thought of as representing a family of fragility curves expressing the fraction of earthquakes of a given peak ground acceleration at which the component is expected to fail. Within this family, a mean value at discrete accelerations is obtained for point estimate calculations, discussed later.

As previously stated, the fragility descriptions are based on a logarithmic distribution because the data fit that as well as other possible distributions. However, the data do not fit well in the tails of the distribution below failure fractions of 0.01 to 0.02. At these levels, the curves are considered to be very conservative (see Appendix B). For example, conventional components such as piping and conduits routinely withstand static vertical 0.1g loads without failing. Small dynamic loads resulting from cranes, forklifts, and other component handling equipment regularly occur without causing structures to fail. For low acceleration levels, say below 0.05g, it is inconceivable that well engineered structures will have even a small chance of failure.

It is therefore expected that below some acceleration threshold, there is virtually no chance of failure due to seismic excitation. Material strength and damping, for instance, do not have infinitely low and high values but instead have some lower and upper thresholds. Further, extensive studies have been conducted to develop response spectra from available earthquake records and, while dispersion exists about the median values, spectra with essentially zero or infinite response do not occur. For these as well as other variables contributing to the seismic fragility of a given structure or component, it is apparent that some lower and upper cutoffs on the tails of the dispersion exist. Since the overall fragility curves are based on a combination of these variables, it is expected a lower threshold exists below which no failures will occur. This is supported by experience. Although quantitative data are lacking, this lower threshold value for the median fragility curve is judged in Appendix B to be

$$\overset{\approx}{a} \exp(-2\beta_c) \tag{2.9}$$

where

$$\beta_C = \sqrt{\beta_R^2 + \beta_U^2} \tag{2.10}$$

The cutoff for the lower tails of the other fragility curves is then

$$\overset{\approx}{a} \exp(-2\beta_c) \cdot \exp\left[\left(\frac{x}{1.65}\right)\beta_c\right] \tag{2.11}$$

or

$$\tilde{\tilde{a}} \exp \left[\left(\frac{x}{1.65} \right) - 2 \beta_c \right] \qquad (2.12)$$

where x is the ratio of the deviation for the curve of interest to the standard deviation. For the 5th percentile curve, this value becomes

$$\tilde{\tilde{a}} \exp (-3 \beta_c)$$

and for the 95th percentile curve

$$\tilde{\tilde{a}} \exp (- \beta_c)$$

The upper threshold value for all curves is judged in Appendix B to be

$$\tilde{\tilde{a}} \exp (3 \beta_c)$$

However, there is some speculation about the suitability of these cutoff points. Therefore, when the full family of fragility curves is used truncations are only made in the seismic analysis when the failure fractions are less than 1×10^{-6}, regardless of acceleration.

In the following sections, the specifics of the fragility analysis, as they apply to the TMI-1 plant, are described.

2.2.3 STRUCTURES AND EQUIPMENT FRAGILITIES

The key fragility parameters, which resulted from the fragility analysis for structures at TMI-1, are tabulated in Table 2-2. Similar results for mechanical and electrical equipment whose failure can result in the initiation of a scenario or in the degradation of the plant response to such accidents are tabulated in Table 2-3.

The seismic hazard curves in Figure 2-1 indicated that the upper bound on ground acceleration was less than 1.0g, certainly for the mean and for most of the curves. Further, the annual exceedance frequency for these high accelerations is extremely small. As a step in reducing the number of structures and equipment listed in Tables 2-2 and 2-3 that have to be considered, those that have median acceleration capacities greater than 1.0g were excluded. The remaining components are seen in Table 2-4, which is a summary of the key plant components having the lower median acceleration capacities. The table lists the components in order of increasing $\tilde{\tilde{a}}$ and includes the random and uncertainty variables, β_R and β_U, respectively.

Two capacities are shown in Table 2-4 for some of the electrical components, such as the diesel generator control panel, switchgear, and 480V motor control centers, transformers, and buses. The first is for either chatter or relay trip failures that are automatically or manually recoverable. The second capacity of these components, as indicated in Appendix B and seen in the table, is for structural or nonrecoverable failure and is significantly greater.

The failure fractions of the critical key components at various mean discrete acceleration levels are tabulated in Table 2-5. These values are used for point estimate quantification. Blank spaces represent accelerations below the lower bound cutoff for which no failure is predicted. This lower bound is the high confidence low probability of failure point equal to the 5th percentile of randomness on the 5th percentile uncertainty curve.

2.3 SYSTEMS AND PLANT LOGIC

The occurrence of a seismic event could initiate a sequence at TMI-1 in any of several ways. Failure of the offsite power transformer insulators, item ①, seen in Table 2-4, would result in offsite power to the plant being lost. Also, at higher accelerations, combinations of river water pumps, nuclear service river water pumps, and nuclear service, ICCW, or decay heat component cooling water heat exchangers, items ⑬ , ⑯ , ㉓ , ㉔ , or ㉕ , respectively, could cause a loss of river water and reactor shutdown. Other failures, such as instrument buses, item ⑨, that would cause a transient type event would occur at accelerations higher than those that would already have caused a loss of offsite power and would result in similar sequences.

Event trees used for internal analyses are also used to describe the plant response to seismic initiators. The event tree that closely models the course of scenarios initiated by the above events is the general transient tree with a turbine trip as the initiating event for any earthquake. Component failures that result in the unavailability of the systems or actions represented by the top events in this tree and that would affect the scenarios were considered.

If a seismic event should occur, it is possible for mitigating structures and equipment to fail from the earthquake effects. They might also be unavailable from such nonseismic causes as random failures, testing, or maintenance. These other causes were therefore also included in the seismic analysis.

The methodology applied in this project for analyzing the consequences of possible seismic failures using the logic of the event trees is summarized as follows:

1. Dependencies were identified between possible seismic failures of components modeled for internal events analyses, or of passive components not modeled (such as structures, piping, or cable trays), and failure of the equipment and systems that could mitigate accident scenarios.

2. The seismic structures not previously in the plant model were added to the first top event that they would affect. In addition, a seismic failure term was added for equipment and other components already in the model.

3. The split fractions (the likelihood for success or failure of top events) used in the plant model were modified to include the probable seismic failures at discrete accelerations as well as the nonseismic failures normally considered.

4. Mean values of the seismic hazard frequency at discrete accelerations were used as point estimates of initiating event frequencies; in this case, turbine trip..

5. With the initiating event frequencies and split fractions established for discrete accelerations as input, the ETC code runs were made for each of four seismic events (accelerations) and the results of these runs were assembled using the MAXIMA code to obtain the seismic contribution to plant damage state frequency.

6. The frequencies of these point estimates of seismically initiated plant damage states were compared with plant damage state frequencies from other initiating events to determine if any seismically initiated plant damage state frequencies are significant enough to warrant the uncertainty analysis and more precise quantification using the SEIS code. Since seismic contribution was not significant, the SEIS code was not executed.

Table 2-6 lists the impact relationship between the seismic failure of plant components and the consequential failure of top events reflected in the transient event tree. Using the potential seismic failures indicated in Table 2-6, a Boolean equation was developed for each top event in the event tree. For each such numbered top event seen in Table 2-6, the Boolean is indicated in Table 2-7. Also shown in the table is the conditional mean seismic failure fraction for each top event (as represented by each Boolean) in the event tree at the same discrete mean accelerations indicated in Table 2-1. The nonseismic unavailability of all top events in the logic model is not included in the values seen in Table 2-7, but for the event tree quantification was obtained from the systems analyses for internal events. The total unavailability of each top event is then determined at each discrete acceleration by adding the values of seismic failure fractions to the nonseismic conditional failure probabilities (which are constant over all acceleration ranges).

Two types of dependencies were considered in the analysis: statistical and functional. Where indicated in Table 2-6, failures affecting more than one top event in the event tree (statistically dependent component failures) were accounted for in the support system tree. Also, as indicated in the table footnotes, certain component failures are considered to be highly dependent functionally. That is, because the components are similar; are in the same building or location; and have common modeling assumptions, if the stronger one fails, there is a high likelihood the weaker ones will have also failed. Where it was conservative to assume these dependencies, such as between the reactor river water and nuclear service river water pumps, or between heat exchangers, total dependency for seismic failure was assumed.

The unconditional plant damage state frequencies for seismic initiated events were calculated using the event trees as was done for the internal events analysis. As in the internal events analysis, the support system tree was quantified using the event tree code (ETC), but in this case the initiating event frequency was each of the discrete mean acceleration values seen in Table 2-1. The main event tree was also quantified for each of the support system states. The calculations from the support and main event trees were combined for each of the discrete acceleration values to obtain the unconditional mean frequencies for each scenario in each plant damage state.

The frequencies of major contributing scenarios were binned for each plant damage state using the MAXIMA code. The results for each acceleration are given in Table 2-8. The frequencies of the seismic initiated plant damage states were then compared with those resulting from other initiators to determine if any seismic scenarios are major contributors to a plant damage state and to core damage.

Something can be said of the results from the seismic point estimate analysis. As seen in Table 2-8, the total seismic contribution to overall core melt frequency is negligible (2.6×10^{-6} versus 4.7×10^{-4}, respectively). As seen in the table, only plant damage state 5E has a significant contribution from seismic events. The major contributors to PDS 5E are the loss of offsite power, item ①, and a loss of DC power due to seismic failure of the DC battery chargers, item ⑩, and batteries, item ㉚, resulting in an eventual loss of DC power to control the diesel generators, thereby losing power to the BWST pumps. With this or seismic failure of the BWST, item ⑫, there would be no core cooling; thus, core damage would be ensured.

It is seen in Table 2-8 that seismically initiated PDS 5E and 5F contribute about 90% of the seismic initiated core damage. However, as is also seen in the table, total seismically initiated core damage is less than 1% of the total. Therefore, further refinement of the calculations and performance of an uncertainty analysis is unnecessary.

TABLE 2-1. MEAN ACCELERATION FREQUENCIES

Discrete Acceleration Levels			
.15g	.25g	.4g	.6g
1.09×10^{-3}	1.67×10^{-4}	4.77×10^{-5}	5.59×10^{-6}

TABLE 2-2. SEISMIC CAPACITY OF STRUCTURES

Structure	Critical Element	Median Acceleration Capacity $\tilde{\tilde{a}}$(g)	β_R	β_U
Reactor Building	Shear Wall Failure	5.5	0.32	0.36
	Secondary Shield Wall Failure	2.4	0.25	0.35
	Primary Shield Wall Failure	2.6	0.25	0.37
Control Building	Shear Wall Failure	1.0	0.27	0.36
Auxiliary Building	Shear Wall Failure	1.7	0.24	0.35
Intake Screen House	Shear Wall Failure	1.4	0.12	0.29
Intermediate Building	Shear Wall Failure	1.3	0.21	0.33
Diesel Generator Building	Impact Due to Sliding	1.3*	0.23	0.42
Borated Water Storage Tank	Wall Buckling	0.62	0.24	0.43
Condensate Storage Tank	Anchor Bolts Failure	2.0	--	--

*Lower bound cutoff at 0.66g.

TABLE 2-3. SUMMARY OF KEY COMPONENT FRAGILITIES

Sheet 1 of 4

System	Component	$\tilde{\tilde{a}}(g)^{(1,2)}$	β_R	β_U
Emergency Core Cooling System	BWST	0.62	0.24	0.43
	HPI Makeup Pumps	>1.0	--	--
	Isolation Valves	>1.0	--	--
	LPI/DHR Pumps	>1.0	--	--
	DHR Heat Exchangers	0.75	0.25	0.31
	Isolation Valves	>1.0	--	--
	Dropline Valves	>1.0	--	--
	Piggyback Valves	>1.0	--	--
	Reactor Building Sump	>1.0	--	--
	Isolation Valves	>1.0	--	--
Reactor Building Spray	Reactor Building Spray Pumps	>1.0	--	--
	Spray Header and Nozzles	>1.0	--	--
	Motor-Operated Valves	>1.0	--	--
Reactor Building Emergency Cooling System	Reactor River Pumps	0.58	0.39	0.39
	Cooling Coils	0.9	0.25	0.42
	Isolation Valves	>1.0	--	--
	Fans and Motors	>1.0	--	--
Emergency Feedwater System	Motor-Driven Pumps	>1.0	--	--
	Turbine-Driven Pumps	>1.0	--	--
	Flow Control Valves	>1.0	--	--
	Block Valves (MOVs)	>1.0	--	--
Engineered Safeguards Actuation System	Sensors	0.88	0.25	0.40
	Actuation Cabinets A and B	0.4/0.8$^{(3)}$	0.25/0.25$^{(3)}$	0.48/0.34$^{(3)}$
	Engineered Safeguards Relay Cabinets	0.4/0.8$^{(3)}$	0.25/0.25$^{(3)}$	0.48/0.34$^{(3)}$
	Bistable Cabinets	0.4/0.8$^{(3)}$	0.25/0.25$^{(3)}$	0.48/0.34$^{(3)}$
Reactor Protection System	CRDMs and Assemblies	>1.0	0.25	0.34

NOTE: Notes (1) through (3) are on Sheet 4 of this table.

TABLE 2-3 (continued)

Sheet 2 of 4

System	Component	$\tilde{\tilde{a}}(g)$(1,2)	β_R	β_U
Electric Power				
A. AC Power	4,160V Switchgear	0.4/0.8(3)	0.25/0.25(3)	0.48/0.34(3)
	4,160V/480V Transformer	0.73	0.25	0.29
	480V Switchgear	0.4/0.8(3)	0.25/0.25(3)	0.48/0.34(3)
	480V MCC	0.4/0.8(3)	0.25/0.25(3)	0.48/0.34(3)
B. DC Power	Batteries	0.95	0.25	0.56
	Chargers	0.49	0.25	0.60
	Inverters	0.49	0.25	0.60
	DC Distribution Panels 1A and 1B	>1.0	--	--
	DC Subpanels 1E,1C,1H,1D, 1F, and 1J	>1.0	--	--
	Vital AC Instrument Buses VBA/B/C/D, ATA/B, TRA, and PRB	0.4/0.8(3)	0.25/0.25(3)	0.48/0.34(3)
	120V Transformers	0.73	0.25	0.29
C. Offsite Power	Ceramic Insulators, etc.	0.3	0.25	0.50
D. Emergency Power	Diesel Generators (everything on the skid)	0.75	0.25	0.44
	Air Receiver Tank	0.68	0.25	0.25
	Fuel Oil Transfer Pump	>1.0	--	--
	Air Start Compressor	>1.0	--	--
	Batteries for Air Start Compressor	0.3	0.25	0.31
	Diesel Generator Control/ Breaker Panel	0.37/>1.0(3)	0.25	0.42
	Fuel Oil Day Tank	0.6	0.25	0.42
Reactor Coolant System	Reactor Pressure Vessel	>1.0	--	--
	Reactor Coolant Pumps	>1.0	--	--
	Pressurizer	>1.0	--	--
	Steam Generator	>1.0	--	--
	RPV Internals	0.86	0.29	0.50

NOTE: Notes (1) through (3) are on Sheet 4 of this table.

TABLE 2-3 (continued)

Sheet 3 of 4

System	Component	$\overset{\approx}{a}(g)^{(1,2)}$	β_R	β_U
	Pressurizer Safety Valves	>1.0	--	--
	PORV	>1.0	--	--
	Reactor Coolant Drain Tank	0.7	0.25	0.40
	Auxiliary Spray Line	>1.0	--	--
Control Building Ventilation System	Normal Supply Fans	>1.0	--	--
	Emergency Supply Fans	>1.0	--	--
	Chilled Water Supply Pumps	>1.0	--	--
	Air-Operated Dampers	>1.0	--	--
	Booster Fans	>1.0	--	--
	Return Fans	>1.0	--	--
Nuclear Service River and Closed Cooling Water Systems	Nuclear Service River Water Pumps	0.68	0.39	0.39
	Nuclear Service Heat Exchangers	0.75	0.25	0.31
	Intermediate Closed Cooling Water Heat Exchangers	0.75	0.25	0.31
	Nuclear Service Cooling Water Pumps	>1.0	NA	NA
	Nuclear Service Surge Tank	0.7	0.25	0.40
	Supply and Return Isolation Valves	>1.0	--	--
Decay Heat River and Closed Cooling Water Systems	Decay Heat River Pumps	1.16	0.39	0.39
	Decay Heat Removal Heat Exchanger	0.75	0.25	0.31
	Decay Heat Closed Cooling Water Pumps	>1.0	NA	NA
	Decay Heat Closed Cooling Water Heat Exchangers	0.75	0.25	0.31
	Decay Heat Surge Tanks	0.7	0.25	0.40
	Supply and Return Isolation Valves	>1.0	--	--
Main Steam System	Main Steam Safety Valves	>1.0	--	--
	Atmospheric Dump Valves	>1.0	--	--
	Turbine Bypass Valves	>1.0	--	--

NOTE: Notes (1) through (3) are on Sheet 4 of this table.

TABLE 2-3 (continued)

Sheet 4 of 4

System	Component	$\tilde{\tilde{a}}(g)^{(1,2)}$	β_R	β_U
	MSIV	>1.0	--	--
	Main Steam Lines	>1.0	--	--
	Turbine Stop Valves	>1.0	--	--
	Turbine Control Valves	>1.0	--	--
Containment Isolation System	Containment Purge Valves	>1.0	--	--
	Letdown Isolation Valves	>1.0	--	--
	RCP Seal Isolation Valves	>1.0	--	--
Air Systems	Air Bottles (2-hour emergency)	>1.0	--	--
	Regulating Valves	>1.0	--	--
	Piping	>1.0	--	--
Intermediate Closed Cooling Water System	Intermediate Cooling Pumps	>1.0	--	--
	Surge Tanks	0.7	0.25	0.40
	Intermediate Closed Cooling Water Heat Exchangers	0.75	0.25	0.31
	Isolation Valves	>1.0	--	--

NOTES:

1. $\tilde{\tilde{a}}$ is a conservative median capacity level which has been derived from the results of past SMA PRAs, except where otherwise noted.
2. Fragilities labeled ">1.0" are not expected to influence the risk, based on PLG's assessment of the hazard curves. These components or structures have a high confidence (95%) of a low probability of failure (5%) at 0.4 g's or greater.
3. Electrical components may have two values given in the attached table; i.e., "a/b". These two values "a" and "b" represent recoverable (chatter and trip) and nonrecoverable failures, respectively.

TABLE 2-4. KEY STRUCTURES/COMPONENTS FOR SEISMIC ANALYSIS

Sheet 1 of 2

Component	$\overset{\approx}{a}$(g)	β_R	β_U	β_C
1 Ceramic Insulators	.30	.25	.50	.56
2 Diesel Generator Control/Breaker Panel	.37/>1.0	.25	.42	.49
3 Actuation Cabinets	.40/.80	.25	.48/0.34	.42
4 Engineered Safeguards Relay Cabinets	.40/.80	.25	.48/0.34	.42
5 Bistable Cabinets	.40/.80	.25	.48/0.34	.42
6 4,160V Switchgear	.40/.80	.25	.48/0.34	.42
7 480V Switchgear	.40/.80	.25	.48/0.34	.42
8 480V MCC	.40/.80	.25	.48/0.34	.42
9 Vital AC Instrument Buses VBA/B/C/D	.40/.80	.25	.48/0.34	.42
10 DC Power Chargers	.49	.25	.60	.65
11 DC Power Inverters	.49	.25	.60	.65
12 BWST	.53	.25	.44	.51
13 Reactor River Pumps	.58	.39	.39	.55
14 Fuel Oil Day Tank	.60	.25	.42	.49
15 Air Receiver Tank	.68	.25	.25	.35
16 Nuclear Service River Water Pumps	.68	.39	.39	.55
17 NSS Tank	.70	.25	.40	.47
18 Decay Heat Surge Tanks	.70	.25	.40	.47
19 Surge Tanks	.70	.25	.40	.47
20 4,160V/480V Transformer	.73	.25	.29	.38

TABLE 2-4 (continued)

Sheet 2 of 2

Component	$\tilde{\tilde{a}}$ (g)	β_R	β_U	β_C
21 120V Transformer	.73	.25	.29	.38
22 DHR Heat Exchangers	.75	.25	.31	.40
23 Nuclear Service Heat Exchangers	.75	.25	.31	.40
24 Intermediate Closed Cooling Water Heat Exchanger	.75	.25	31	.40
25 Decay Heat Component Cooling Water Heat Exchanger	.75	.25	.31	.40
26 Diesel Generators	.75	.25	.44	.51
27 RPV Internals	.86	.29	.50	.58
28 Sensors	.88	.25	.40	.47
29 Cooling Coils	.90	.25	.42	.49
30 DC Power Battery	.95	.25	.56	.61
31 Control Building	1.00	.27	.36	.45

TABLE 2-5. MEAN SEISMIC FAILURE FRACTIONS OF KEY STRUCTURES/COMPONENTS

Sheet 1 of 2

Component	Acceleration			
	.15g	.25g	.4g	.6g
① Ceramic Insulators	6.17-2	3.49-1	6.28-1	8.73-1
② Diesel Generator Control/Breaker Panel	--	--	--	--
③ Actuation Cabinets	--	--	3.67-2	2.19-1
④ Engineered Safeguards Relay Cabinets	--	--	3.67-2	2.19-1
⑤ Bistable Cabinets	--	--	3.67-2	2.19-1
⑥ 4,160V Switchgear	--	--	3.67-2	2.19-1
⑦ 480V Switchgear	--	--	3.67-2	2.19-1
⑧ 480V MCC	--	--	3.67-2	2.19-1
⑨ Vital AC Instrument Buses VBA/B/C/D	--	--	3.67-2	2.19-1
⑩ DC Power Chargers	1.45-2	1.41-1	3.24-1	5.94-1
⑪ DC Power Inverters	1.45-2	1.41-1	3.24-1	5.94-1
⑫ BWST	1.73-3	6.39-2	2.33-1	5.60-1
⑬ Reactor River Pumps	1.79-3	5.91-2	2.03-1	4.91-1
⑭ Fuel Oil Day Tank	--	3.43-2	1.59-1	4.63-1
⑮ Air Receiver Tank	--	4.83-4	4.79-2	3.19-1
⑯ Nuclear Service River Water Pumps	3.08-4	3.25-2	1.33-1	3.79-1
⑰ NSS Tank	--	1.06-2	8.93-2	3.38-1
⑱ Decay Heat Surge Tanks	--	1.06-2	8.93-2	3.38-1

NOTE: Exponential notation is indicated in abbreviated form; i.e., 6.17-2 = 6.17 x 10^{-2}.

TABLE 2-5 (continued)

Sheet 2 of 2

Component	Acceleration			
	.15g	.25g	.4g	.6g
⑲ Surge Tanks	--	1.06-2	8.93-2	3.38-1
⑳ 4,160V/480V Transformer	--	4.73-4	4.20-2	2.69-1
㉑ 120V Transformer	--	4.73-4	4.20-2	2.69-1
㉒ DHR Heat Exchangers	--	5.01-4	4.16-2	2.54-1
㉓ Nuclear Service Heat Exchangers	--	5.01-4	4.16-2	2.54-1
㉔ Intermediate Closed Cooling Water Heat Exchanger	--	5.01-4	4.16-2	2.54-1
㉕ Decay Heat Component Cooling Water Heat Exchanger	--	5.01-4	4.16-2	2.54-1
㉖ Diesel Generators	--	9.31-3	8.20-2	3.00-1
㉗ RPV Internals	--	1.10-2	7.23-2	2.44-1
㉘ Sensors	--	--	3.51-2	1.86-1
㉙ Cooling Coils	--	5.76-4	3.62-2	1.82-1
㉚ DC Power Battery	--	8.51-3	6.21-2	2.07-1
㉛ Control Building	--	--	1.19-2	1.12-1

NOTE: Exponential notation is indicated in abbreviated form; i.e., 1.06-2 = 1.06 x 10^{-2}.

TABLE 2-6. SEISMIC IMPACTS

EVENT NUMBER	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
COMPONENT	OFFSITE POWER OP	RIVER WATER HA/HB/NS	INSTRUMENT AIR AM	CONTROL BUILDING VENTILATION CV	DC POWER DA/DB	VITAL INSTRUMENT BUSES VA/VB	ATA BUS AA	CLASS IE SWITCHGEAR GA/GB, GIVEN OP	CLASS IE SWITCHGEAR GA/GB, GIVEN OP FAILED	ESAS EA/EB	EFW EF+/EF−	MFW MF+/MF−	MAIN STEAM SL	BWST BW	RBEC CF	RPS/RODS RT
① CERAMIC INSULATORS	X															
② DIESEL GENERATOR CONTROL/BREAKER																
③ ESAS ACTUATION CABINETS A AND B										X						
④ ESAS RELAY CABLES										X						
⑤ BISTABLE CABINETS										X						
⑥ 4,160V SWITCHGEAR								X	X							
⑦ 480V SWITCHGEAR								X	X							
⑧ 480V MCC								X	X							
⑨ VITAL AC INSTRUMENT BUSES VBA/B/C/D, ATA/ATB, TRA, AND TRB						X										
⑩ DC POWER CHARGERS					X											
⑪ DC POWER INVERTERS						X										
⑫ BWST														X		
⑬ REACTOR RIVER PUMPS															X	
⑭ FUEL OIL DAY TANK									X							
⑮ AIR RECEIVER TANK			X													
⑯ NSRW PUMPS		X														
⑰ NSCCW SURGE TANK		X														
⑱ DHCCW SURGE TANK																
⑲ ICCW SURGE TANK																
⑳ 4,160V/480V TRANSFORMERS								X	X							
㉑ 120V TRANSFORMERS							X									
㉒ DHR HEAT EXCHANGERS																
㉓ NUCLEAR SERVICES HEAT EXCHANGERS		X														
㉔ ICCW HEAT EXCHANGERS																
㉕ DHCCW HEAT EXCHANGERS																
㉖ DIESEL GENERATORS									X							
㉗ RPV INTERNALS																X
㉘ SENSORS										X	X	X	X			X
㉙ REACTOR BUILDING ECW COOLING COILS															X	
㉚ STATION BATTERIES					X											
㉛ CONTROL BUILDING				X												

TABLE 2-7. SEISMIC BOOLEANS AND CONDITIONAL SEISMIC FAILURE FRACTION CONTRIBUTIONS

Event Number*	Boolean Expression	Acceleration			
		0.15g	0.25g	0.4g	0.6g
1	①	6.17-2	3.49-1	6.28-1	8.73-1
2	⑯ ∨ ⑰ ∨ ㉓	3.08-4	4.33-2	2.39-1	9.40-1
3	⑮	--	4.83-4	4.79-2	3.19-1
4	㉛	--	--	1.19-2	1.12-1
5	⑩ ∨ ㉚	1.45-2	1.38-1	1.85-1	6.79-1
6	⑨ ∨ ⑪	1.45-2	1.41-1	3.51-1	6.83-1
7	㉑	--	4.73-4	4.20-2	2.69-1
8	⑥ ∨ ⑦ ∨ ⑧ ∨ ⑳ (Given OP)	--	4.73-4	1.44-1	6.52-1
9	⑥ ∨ ⑦ ∨ ⑧ ∨ ⑭ ∨ ⑳ ∨ ㉖ (Given Loss of OP)	--	4.38-2	3.39-1	8.69-1
10	③ ∨ ④ ∨ ⑤ ∨ ㉘	--	--	1.37-1	6.12-1
11	㉘	--	--	3.51-2	1.86-1
12	㉘				
13	㉘				
14	⑫	1.73-3	6.39-2	2.33-1	5.60-1
15	⑬ ∨ ㉙	--	5.97-2	2.32-1	5.84-1
16	㉗ ∨ ㉘	--	1.10-2	1.05-1	3.85-1

*Referenced to Table 2-6.

NOTES:

1. Exponential notation is indicated in abbreviated form; i.e., 6.17-2 = 6.17 x 10^{-2}.
2. Above values do not include the conditional unavailability of the top events due to nonseismic events.

TABLE 2-8. SEISMIC-INITIATED PLANT DAMAGE STATE FREQUENCIES

Plant Damage State	Acceleration					PDS Total Frequency
	0.15g	0.25g	0.40g	0.60g	Total	
1A	--	--	--	--	--	4.0-10
1C	--	--	--	--	--	1.5-16
1D	--	--	--	--	--	4.1-12
1F	--	--	--	--	--	2.6-13
1H	--	--	--	--	--	1.6-11
2A	--	--	--	--	--	2.6-5
2B	--	--	--	--	--	2.8-7
2C	--	--	--	--	--	4.9-7
2D	--	--	--	--	--	1.4-9
2E	--	--	--	--	--	2.3-9
2F	--	--	--	--	--	3.0-8
2G	--	--	--	--	--	4.8-9
2H	--	--	--	--	--	1.2-6
3A	--	--	--	--	--	4.6-6
3C	3.8-16	1.5-15	3.1-15	2.8-10	2.8-10	6.5-6
3D	--	--	--	--	--	6.4-9
3E	2.8-16	1.1-15	2.3-15	9.3-14	9.6-14	2.1-9
3F	2.4-16	1.0-15	2.0-15	1.1-12	1.1-12	5.9-7
3H	--	--	--	--	--	6.9-7
4A	2.5-8	1.2-7	7.2-8	1.7-8	2.3-7	6.9-5
4B	--	--	--	--	--	0.0
4C	--	--	--	--	--	2.0-4
4D	1.2-12	5.3-12	3.3-12	8.0-13	1.0-11	1.1-6
4E	--	--	--	--	--	0.0
4F	--	--	--	--	--	0.0
4G	--	--	--	--	--	5.5-7
4H	--	--	--	--	--	1.5-7
5A	9.5-10	1.0-9	2.4-10	1.4-10	2.3-9	3.1-5
5B	1.3-10	5.7-10	3.5-10	1.7-8	1.8-8	4.5-5
5C	2.3-12	3.6-12	3.5-12	5.4-9	5.4-9	2.3-5
5D	3.0-13	3.2-13	7.7-14	9.6-15	7.0-13	1.7-8
5E	2.9-11	8.8-11	4.8-11	1.3-6	1.3-6	2.3-6
5F	1.9-10	2.0-10	5.0-11	1.0-6	1.0-6	2.2-5
5G	1.9-11	7.7-11	1.5-10	8.1-10	1.0-9	1.1-6
5H	1.0-9	4.6-9	2.7-9	7.2-8	8.0-8	3.5-6
Total	2.6-8	1.2-7	7.5-8	2.4-6	2.6-6	4.7-4

NOTE: Exponential notation is indicated in abbreviated form; i.e., 4.0-10 = 4.0×10^{-10}.

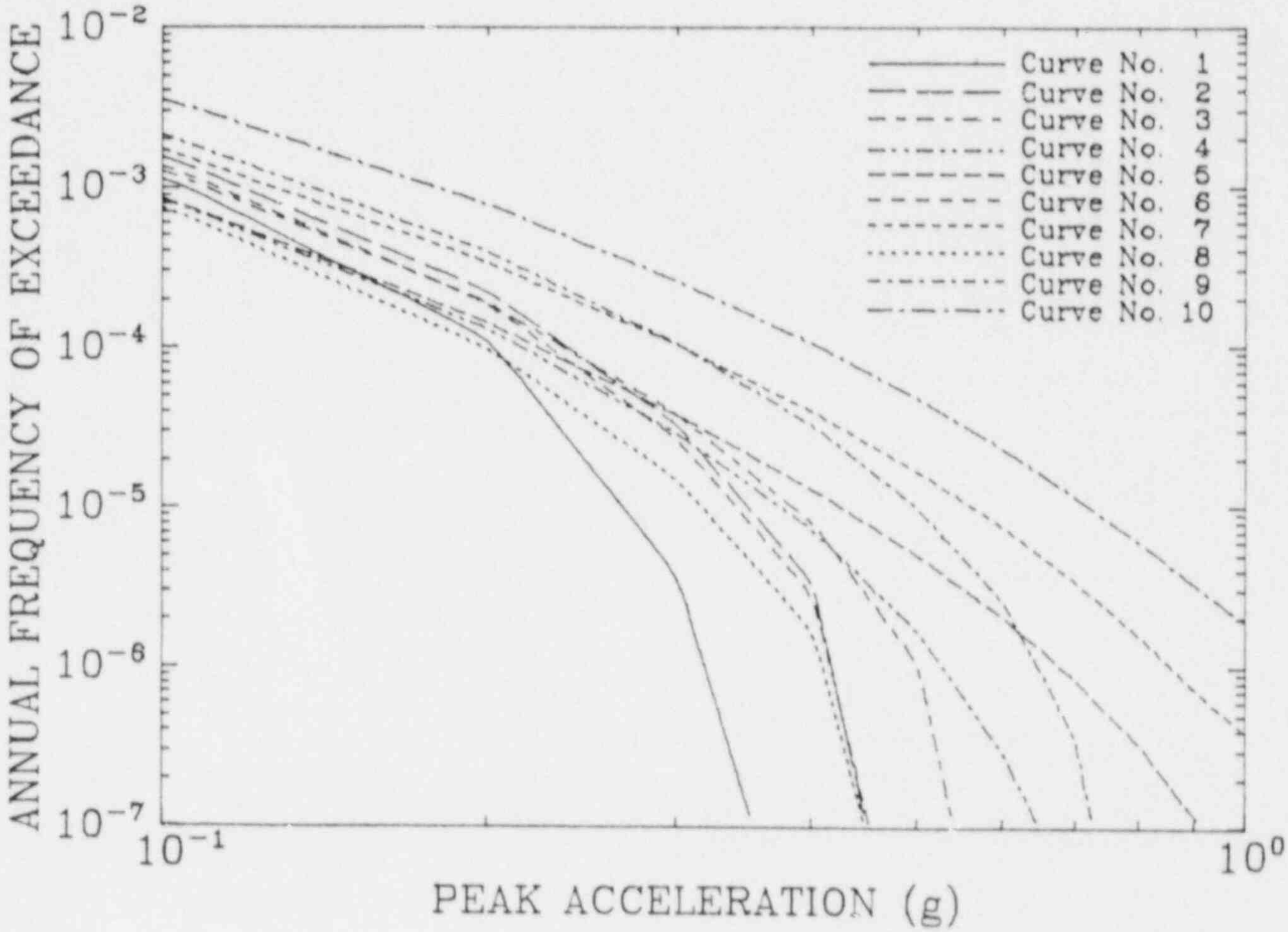

FIGURE 2-1. FAMILY OF SEISMIC HAZARD CURVES FOR THE TMI-1 SITE

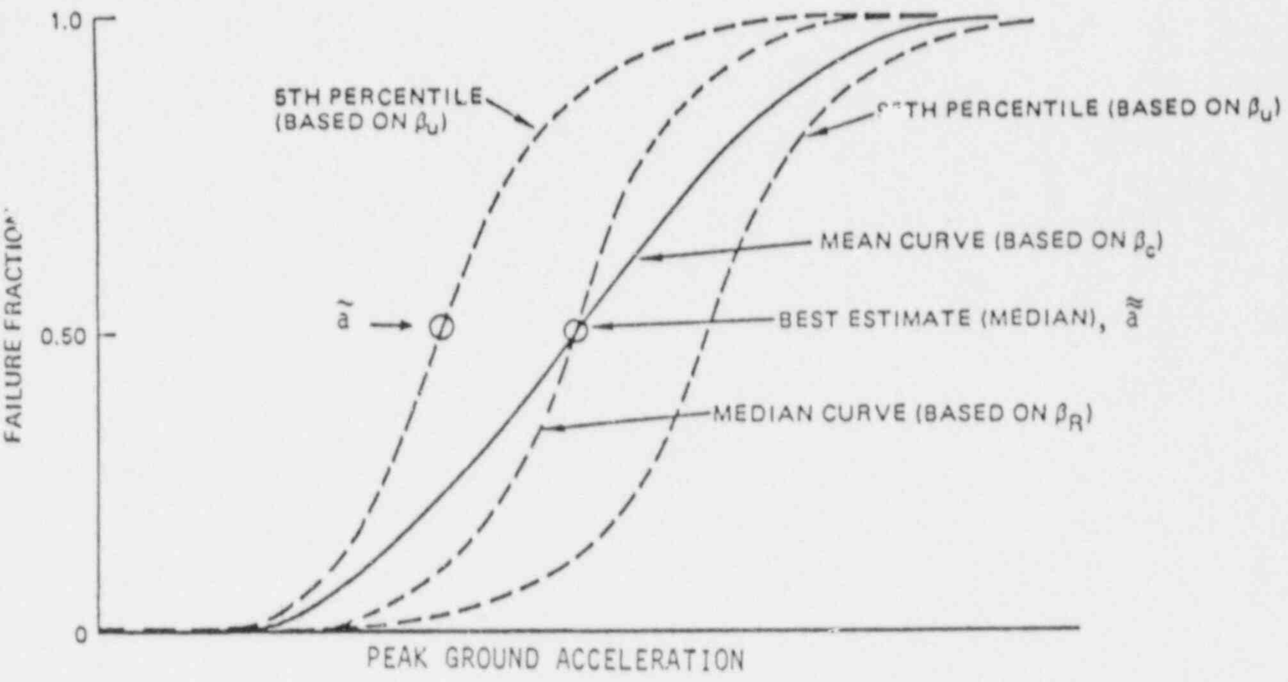

FIGURE 2-2. TYPICAL FRAGILITY CURVE

3. ANALYSIS OF SPATIAL INTERACTIONS

The primary objective of the TMI-1 spatial interaction analysis is to identify those physical interactions involving environmental hazards, such as fire, flood, and steam, that contribute significantly to risk. These hazards can cause an initiating event, fail or degrade the performance of one or more systems, or cause intersystem dependent failures. The spatial interaction scenarios considered to be the most important are evaluated further as initiating events in the plant model report or as contributors to individual system failures in the systems analysis report.

The analysis of the spatial interactions can be divided into two parts: (1) identification of environmental hazard scenarios and (2) assessment of their importance relative to the other contributors to risk. Several sets of tables are developed for the first part, which catalog the information needed for scenario identification and plant impact evaluation. In the first set, the inventory of the components is listed for each location in the plant. In the second set, the potential sources of hazards are identified within each location. In the third set, again for each location, a series of hazard propagation scenarios are developed and listed. Finally, in the last set, the impact of the hazard propagation scenarios on plant systems is evaluated and documented. The complete set of these tables is contained in Appendix C.

The relative importance of each hazard scenario is evaluated and compared to other contributors. The hazard scenarios are classified into two categories: (1) scenarios that impact more than one system and (2) scenarios whose impact is limited to only one system. The results of this evaluation and comparison are documented in the final part of this section. In the following sections, the steps taken for spatial interaction evaluation are described in detail. The final conclusions are given in Section 3.7.

In general, the limitations noted in Reference 3-1 (see Table 3-1) for the analysis of fire scenarios also apply to this analysis of spatial interactions.

3.1 COMPONENT INVENTORY

To determine the significant spatial interactions, it is necessary to know the component inventory of each location in the plant. These are the components that are included in the PRA model. The inventory information was collected and documented in a table for each selected location. Table 3-2 shows an example. The headings for the columns in the tables delineate the system, pump, valve, electrical cabinet, and three types of cables (power, control, and instrumentation) that are contained in the designated location. The "other items" column groups together other components, such as tanks, pipes, etc. The sources of information are referenced, and a column for remarks is also provided for items, such as assumptions, the function of the component, etc. Appendix C contains all the location inventory tables, which are organized by building and fire zone designators.

There were three primary sources for gathering component location information:

1. Fire hazard analysis for Appendix R compliance (Reference 3-2).
2. Drawings: architectural, piping isometrics, heating, ventilation, air conditioning schematics, color-coded electrical cable drawings, and cable tray and conduit drawings.
3. Plant walk-throughs.

The fire hazard analysis report (Reference 3-2) provides an appropriate breakdown of the plant by fire zones and fire areas, both of which are referred to as "locations" in this study. A fire area is defined as one enclosed by 3-hour fire barriers. A fire zone, on the other hand, is a conveniently defined region within the plant that may have openings to other zones. The location names and designators listed on the location inventory tables can be either fire zones or fire areas.

The equipment and cables indicated for each location are identified by "safe shutdown analysis." The safe shutdown analysis (Reference 3-2) was limited to those systems that provide a reactor core safety function. It employs a success-oriented logic that is the logical complement of the failure-oriented logic of fault trees and event trees. The fault tree/event tree logic of this PRA questions the containment safety functions as well as the reactor core; therefore, more systems are considered for accident mitigation than for the case of safe shutdown analysis. The components or systems that this PRA study considers that the safe shutdown analysis of Reference 3-2 does not are:

- Reactor Building Spray System
- PORVs and Their Block Valves
- Emergency Safeguards Actuation Circuits
- Reactor Protection System
- Condensate Pumps
- Instrument Air System
- Turbine Stop and Control Valves
- BWST
- Condensate Storage Tanks
- Reactor Building Isolation
- Control Building HVAC Units AH-E-17A and AH-E-17B
- Offsite Power

This limitation is minimized by assuming that suspected areas indeed contain equipment or cables related to the above items. For example, the reactor building spray pumps are located next to the DHR pumps and are powered from the same switchgear as the DHR pumps. Therefore, the control and power cables to the building spray pumps are judged to be routed the same way as the DHR pump cables.

Two systems are not addressed in this spatial interaction analysis. They are the reactor protection system and the reactor building isolation system. From an evaluation of the RPS, it is concluded that it is highly unlikely for any of the hazards considered in this analysis to fail the RPS so that the control rods would be prevented from inserting or the reactor trip circuit would be prevented from being deenergized.

Similarly, it is highly unlikely that reactor building isolation can be failed by an environmental hazard scenario. All the pipes penetrating the reactor building boundary have two valves for isolating the path. One of the two is always an air-operated valve of the fail-closed type. Therefore, those hazards that do not cause direct damage to the valves and lead to control power deenergization, would lead to reactor building isolation. Furthermore, inducing a spurious signal that would keep the air-operated valve open for a long time is deemed to be an unlikely event. Direct damage to an air-operated valve from an environmental hazard that would prevent its closure is also unlikely. Of course, it is possible for a valve to fail to close. Such failures would have a cause independent from the environmental hazard and are unlikely. Their frequency is less than 10^{-2} per occurrence. Multiplying this frequency times the frequency of the hazard scenario (generally less than 10^{-3} per year), the overall frequency can be concluded as very unlikely.

Certain areas in the plant were reviewed but not analyzed in detail in this study. These areas are the underground duct banks for the emergency power cables from the diesel generators to the switchgears, the yard area, the service building, the transformer area behind the turbine building, the circulating pump house, the cooling towers, and the air intake tunnel. None of these areas except the air duct banks, the air intake tunnel and the yard area contain equipment important to plant safety. Potential adverse phenomena originating in these areas that may propagate to other important areas are deemed to be very unlikely or of little significance. It is envisioned that the emergency power cables in the duct banks are separated by concrete ducts and, therefore, any failures would be limited to only one cable.

The safety equipment in the yard are the BWST and the two condensate storage tanks. They are on two sides of the diesel generator and service buildings, and a single hazard is deemed very unlikely to affect both tanks. The air intake tunnels contain several vital cables. These tunnels are not entered under normal conditions and are protected by very fast acting fire protection systems. Also, they do not contain sources of hazard that could propagate to other parts of the plant. Fires involving the cooling towers, the fuel oil tanks, or the warehouse are not deemed to affect the important equipment or buildings within the plant.

3.2 SOURCE AND MITIGATION TABLES

In the first step of the scenario identification task, a series of tables are put together, one for each location, that are called "source tables." Table 3-3 shows an example source table. See Appendix C for the tables put together for TMI-1 fire zones. The intent in filling out this table is to make a reasonably complete list of sources of environmental hazards that exist in each location. As with the location inventory tables, architectural drawings, piping isometrics, the FSAR, and the fire hazard analysis report are reviewed in conjunction with the walkdown to establish a list of the sources and relevant factors affecting propagation and mitigation. Identification of specific scenarios is performed in conjunction with another type of table, as explained below. Table 3-4 lists the environmental hazard types that are used in this study with examples of each.

In the first column of the source table (Table 3-3), the hazard type is recorded. For each type, there may be several sources that are described by the second through the fourth columns. In the second column, a description of the source is given. For this, the level of detail varies widely. It could range from a certain pipe section of a certain system to a blanketing statement, such as "transient fuels" for fires. It must be noted that detailed source description is needed only when the hazard scenario is important. In the third column, relevant assumptions are listed. For example, in location AB-FZ-5, it is assumed that no river water piping is in the area. This assumption is a reasonable one considering the type of equipment in this location, and it allows us to simplify our search for sources of environmental hazards. In the fourth column, all the references are given, such as drawing numbers and document names or numbers.

In columns five and six of the source tables, the information on mitigative factors for each source is recorded. In the fifth column, all the available systems, components, and equipment that can be used to either contain, totally stop, or retard the phenomenon of concern are mentioned. For example, a mitigative feature for a certain flood is closure of a valve upstream of the break point. Table 3-5 gives additional examples. The references for this information are indicated in the sixth column. Finally, the last column of the table is dedicated to other remarks that need to be noted, but which do not belong to the other columns.

3.3 SCENARIO TABLES

Having listed all possible sources and their respective propagation and mitigation factors, the tables that document specific scenarios, called scenario tables, can be constructed. Table 3-6 gives an example of these tables. Appendix C contains the scenario tables put together for TMI-1 fire zones. In the first two columns, the source is reiterated as it is in the source table. The second column is simply a synopsis of the second column of the source table.

The scenario description is broken into three major parts to cover: (1) the source, (2) the paths of propagation, and (3) the mitigation factors. The source category (the third column) describes how the phenomenon is initiated and its severity level. The propagation to other locations, if any, is recorded in the next two columns. In column four, the type of propagation is given in detail; e.g., a certain door has to leak grossly to cause flooding of an adjacent room. In column five, the location to which the phenomenon propagates is indicated. The third part of the scenario describes mitigation factors that help characterize the resultant damage of the event.

In column seven, it is stated whether further analysis is performed for the scenario; that is, whether it is quantified and considered for inclusion in the plant model. This decision point is included to reduce the number of items that are produced in the quantification tasks. Judgment is used at this level. For example, the analysis is stopped for a very unlikely event if it would affect only the location of origin, and (based on the inventory tables) there are only a few important components in the area. The reasons are recorded in the last column (the 11th) under "Remarks."

The next two columns (i.e., the eighth and ninth) are related to the quantification process. A frequency of occurrence is estimated and recorded in the eighth column. This is discussed in detail in Section 3.5. In the ninth column, a summary of the impact is given. Impact tables are used at this point to record the impact on plant systems (see the following section).

3.4 IMPACT TABLES

For some scenarios, an impact table is put together for documenting the impact of the hazard-induced failures on systems and system trains. Table 3-6 gives an example. The impact tables are put together for only those scenarios whose impact on components and, subsequently, on systems is not easily identifiable.

The impact tables indicate how a hazard scenario impacts the components within the affected locations. They indicate the component failure modes and the status of the affected systems. For example, in Table 3-7, the fire would cause a hot short in the control cables of valve NR-V-5 and lead to the closure of the valve and failure of the associated system.

In these tables, the potential for failure recovery by manual actions is also given. For example, in Table 3-7, system failure because of the closure of NR-V-5 would be recovered by opening redundant valves. Direct recovery of NR-V-5 by deenergizing the motor and manually opening the valve is not possible because the fire is in the pathway of personnel to the valve location.

3.5 FREQUENCY ESTIMATION

Point estimate frequencies are assigned to all hazard scenarios that are chosen for quantification. We have used sources that have evaluated similar hazardous situations or judgment based on the results of other PRA studies. The scenario frequencies are generally the multiplication of several elemental frequencies. These elemental parts account for the severity of the hazard, location of the hazard within the room, failure of timely mitigation of the hazard, fragility of the components, and other relevant factors.

For fire frequencies, Reference 3-3 has been the main source of data. It gives fire frequencies for an auxiliary building in a nuclear plant (mean value of 0.048 per year), a control room (mean value of 0.0049 per year), a cable spreading room (mean value of 0.0067 per year), a diesel generator (mean value of 0.00074 per diesel engine start), a turbine building (mean value of 0.016 per year), and a reactor coolant pump (mean value of 0.0074 per year). These frequencies are adjusted to account for the specific areas within the plant. Generally, a fire frequency of 0.001 per year is used for a typical room or area in the plant. This room or area may contain several cable trays, a few pumps and valves, and may have a moderate level of personnel traffic. Its dimensions are in 10s of feet. For larger areas, or areas containing electrical cabinets, a larger frequency, 0.003 per year, is used. For areas that are small, do not contain any electrical cabinets or motors, or are not visited regularly by plant personnel, a lower frequency, 3×10^{-4} or 1×10^{-4} per year, is used. The sum of all fire scenario frequencies for the auxiliary building, intake structure, fuel handling building, intermediate building, and the control building (not including the control room and the cable spreading room) is 0.049 per reactor year. This sum is very close to the mean frequency given in Reference 3-3.

The other factors of a fire hazard scenario are dependent on the specifics of the scenario itself. For example, the geometric factor (fraction of the room area where a fire would lead to the same component damages of interest) depends on how the important components are arranged within the room. The severity factor depends on the distance between the origin of the fire and target components or on the protective devices in the area. These protective devices are fire suppression systems and fire-rated barriers around cables. The values for these factors are taken from like scenarios in other PRAs for which a detailed fire analysis had been completed. For example, for location AB-FZ-4, scenario 1 in which a fire damages the cables near the ceiling, the severity factor is 0.05 because only a very severe fire can heat the cables to their damage temperature. For such a severity level, the likelihood of suppression is small and the nonsuppression factor is judged to be 0.5. As another example, for a fire that fails cables protected by a fire barrier, a severity factor of 0.03 and a nonsuppression factor of 0.2 have been used.

For flood incidents, the main source is Reference 3-4. The flood frequency was adjusted like the fire frequency for specific locations, sources, and severities. For a severe flooding incident with a multitude of sources within a location, a frequency of 1.0×10^{-4} per reactor year is used. The turbine building is an exception here. Because it is one large, open building, the overall flood frequency is employed. If the flood source consists of a few pipe sections, a frequency of 8×10^{-6} per year per pipe section is used.

For events other than fire and flood, the frequencies are derived from judgment. For smoke propagation incidents, the fire frequencies are used. Several smoke scenarios were found to be important. Since, in all cases, electrical switchgears are involved, the evaluation of the conditional frequencies of damage by smoke are discussed separately below. For steam environment and pipe movement, the pipe failure frequencies are used. For explosions and falling objects, conservative frequencies are used. None of these scenarios was found to have sufficiently large contributions to systems unavailability to require detailed analysis. Missile frequencies are derived from the frequency of the source being present in the location and being mishandled. Only one missile scenario was found to be important enough to warrant a more detailed evaluation. This is scenario 11 for fire zone AB-FZ-6.

3.6 EVALUATION OF SMOKE AND STEAM PROPAGATION SCENARIOS

Several fire hazard scenarios involve smoke propagation to adjacent rooms where the redundant electrical switchgear or the bus bars are located. Several mechanisms can be envisioned for smoke damaging switchgear. The first mechanism scenario involves the failure by corrosion of the contacting surfaces or dielectric breakdown between the bus bars of the different phases. The corrosion is caused by hydrochloric acid (HCl) formed during fires that burn PE/PVC cables and was of concern in a number of large fires; e.g., Muhlenberg and Browns Ferry. However, the degradation of a large switchgear is expected to occur over a long time scale; in no cases have corrosive effects of smoke been recorded that lead to the loss of electrical equipment on a time scale relevant to that associated with safe shutdown. The corrosive smoke hazard is slow acting; therefore, we judge that the likelihood of this damage mechanism leading to core damage is dominated by other scenarios.

The other smoke failure mechanism postulated is the failure of the insulating gap between different phases due to the presence of ionized combustion products in the gap. Large amounts of smoke can be generated in the switchgear room by fires involving any insulating materials in cabinets and the cables above the cabinets. To cause damage, this smoke must infiltrate the switchgear cabinets or bus bars in sufficient density to cause dielectric breakdown.

The failure of electrical equipment under smoke conditions has been observed in some fire incidents (Reference 3-5). However, the influencing parameters, such as smoke density and smoke characteristics and the speed of damage progression, are not well understood yet. There are serious doubts that, because of their size and voltage levels, high voltage switchgears are readily susceptible (within a few hours) to smoke damage (Reference 3-6). Therefore, in this study those scenarios that involve smoke impact of switchgear or high voltage bus bars are judged to be insignificant contributors to risk. Steam propagation is considered for all steam release scenarios. The propagation occurs through doors and HVAC ducts. Typically, in a steam release, it is judged that most of the building would be affected. Therefore, propagation of steam through drain piping is not looked at in detail. For the majority of steam release scenarios, the impact on plant safety is found to be of little importance to plant risk.

3.7 IMPORTANT CONTRIBUTORS

All the hazard scenarios are of two types. The first type of scenarios includes those that impact more than one system and may initiate an event. The second type of scenarios impacts only one system. Table 3-8 summarizes the disposition of all of the hazard scenario tables included in Figures 3-1 through 3-8. These two types of scenarios are further broken down into four categories according to their disposition.

The hazard scenarios that belong to the second type are included in the separate systems analysis sections and are designated category B in Table 3-8. Their contribution to the overall system unavailability or failure frequency is evaluated in those sections.

For the scenarios that impact more than one system, several approaches are taken to establish their level of importance. The simplest case is the one in which it can be established from the impacted equipment that core damage may result from the hazard scenarios. For these scenarios, the core damage frequencies are the same as those for the frequency of the hazard scenario.

For scenarios that do not lead to core damage, three types of actions are taken. Those hazard scenarios that have an annual frequency of less than 3.0×10^{-6} were judged not to be important enough for any further analysis* (category D in Table 3-8). Hazard scenarios with a frequency greater than 3×10^{-6} that fail a large number of equipment (and, therefore, a large number of systems) are analyzed for their potential core damage frequency. The additional equipment failures that can cause core damage to occur are identified, and their conditional split fraction is estimated by using the system analysis report. It must be noted that, for all hazard scenarios in this category, it is assumed that the balance of plant would be affected. If the equipment failures do not lead to this situation, it is assumed that the operators would trip the plant.

Scenarios that fail only a few systems and have an annual frequency greater than 3×10^{-6} are compared with scenarios producing similar effects from the internal event analysis. Scenarios which require additional frequency less than 3×10^{-6} are shown in category C in Table 3-8.

The results for the hazard scenarios that were selected for further analysis (see Appendix C), are summarized in Figures 3-1 through 3-8. (Notations for these figures are given in Figure 3-9. Fold out Figure 3-9 while examining Figures 3-1 through 3-8.) A total of 128 scenarios are addressed in these figures. The level of importance of a scenario and whether it is further analyzed in another section of the PRA are given in these figures. These figures also summarize some of the information given in Appendix C. They reflect the system trains or equipment that are affected by the hazard scenario. This information is put together either directly from the impact tables or by combining the information in the location inventory tables, the susceptibility of the components to the hazard (fragility information), and the propagation pattern of the hazard (given in the scenario tables).

Figures 3-1 through 3-8 indicate whether core damage is possible. If it is not possible, they show, for some hazard scenarios, which additional failures lead to core damage and the frequency of core damage.

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*The 3.0×10^{-6} frequency is an estimate used for screening purposes only. More accurate estimates of core damage frequency, including all possible mitigating actions, were made for all scenarios with a frequency greater than 3.0×10^{-6}.

The final entry of these figures is an indication of whether the hazard scenario is important and how the hazard scenario is incorporated into the main body of the PRA analysis. Six scenarios are found to fail several systems and their estimated core damage frequencies are greater than 3 x 10^{-6} per year (category A in Table 3-8). These scenarios are scenario 1 of fire zone AB-FZ-6 (sheet 19 of Figure 3-1), scenario 1a of fire zone CB-FA-2b (sheet 4 of Figure 3-7), scenario 1 of fire zone CB-FA-2d (sheet 8 of Figure 3-7), scenario 2 of fire zone CB-FA-3a (sheet 14 of Figure 3-7), scenario 1 of fire zone CB-FA-3b (sheet 16 of Figure 3-7), and scenario 1 of fire zone CB-FA-3c (sheet 18 of Figure 3-7). These scenarios are included in the risk model. Special event tree computations are performed using their frequency of occurrence and systems impacted by them.

3.8 REFERENCES

3-1. NSAC, "Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3," cosponsored by the Nuclear Safety Analysis Center, Electric Power Research Institute, and Duke Power Company, NSAC 60-SY, June 1984. (Primary author of Oconee PRA is Nuclear Safety Analysis Center; Pickard, Lowe and Garrick, Inc., either authored or coauthored "Data Base Development," "Turbine Building Flooding," "Seismic," and "Fire.").

3-2. GPU Nuclear Corporation, "TMI-1 Fire Hazard Analysis Report and Appendix R, Section IIIG, Safe Shutdown Evaluation," November 1, 1985.

3-3. Kazarians, M., and G. Apostolakis, "Modeling Rare Events: The Frequencies of Fires in Nuclear Power Plants," presented at the Workshop on Low-Probability/High-Consequence Risk Analysis, Society for Risk Analysis, Arlington, Virginia, June 15-17, 1982.

3-4. Kazarians, M., and K. N. Fleming, "Internal Flood Hazard Model," ANS Transactions, Vol. 45, p. 385, 1983.

3-5. Lefetra, F. E., and E. I. McGowan, "Power, Control, and Instrumentation Cable for Nuclear Fueled Power Plants," IEEE Transactions on Nuclear Science, Vol. NS21 No. 1, February 1974.

3-6. Asselin, L., Electrical Engineering Department, Ecole Polytechnique, Montreal, Quebec, Canada, personal communications with M. Kazarians, Pickard, Lowe and Garrick, Inc., April 10, 1986.

TABLE 3-1. EXCERPT FROM OCONEE PRA REPORT
(Reference 3-1)

Sheet 1 of 2

Fires and their effects on plant safety have not received as much attention as other parts of risk assessment. Therefore, major assumptions had to be made to perform the analysis. The following remarks and those in Section 9.3.5 will place the results of this study into perspective:

1. The analysis was limited to areas where the analysts believed the most damage can be anticipated. Many more areas of the plant would have to be investigated in more detail for a complete fire risk analysis. The degree to which additional analysis is warranted must be balanced by the importance to the overall study results and an understanding of the limitations associated with the state of the art in the analysis of fire event sequences.

2. The frequencies of fires were derived from the experience of all U.S. nuclear power plants. The extent to which they reflect the conditions at Oconee Unit 3 is not entirely certain. For example, it is debatable whether fires like the Brown's Ferry incident should be included in the data base because modifications have been implemented as a result of that fire. Nevertheless, all fires were included in the data base.

3. Simple models were used to assess the propagation of fires in cable trays and the temperature rise in compartments due to the heat released by the fire.

4. The analysis of the fire-initiated sequences was not detailed. Such an analysis would explicitly include the timing of events, the possibility of restoring lost functions, the possibility of errors of commission, and a detailed analysis of local actions outside the control room.

5. Whenever a fire is postulated in an area where it can affect instrumentation, the question of completeness of the analysis becomes very important. It is very difficult to know what information would be presented to the operators and how they would respond. However, the impact of such events on the fire risk is judged to be included in the uncertainties assessed for the dominant sequences.

Limiting Factor	Comment
Probability of Specific Locations of Fires	Based on a review of data and an analysis of the specific areas in relationship to the entire auxiliary building. Considerable analyst judgment involved.
Locations of Critical Fires	Based on review of systems, areas, and locations of important equipment. The areas identified as important may not be the only ones that could result in fire risk.

TABLE 3-1 (continued)

Sheet 2 of 2

Limiting Factor	Comment
Cable Routings	Much uncertainty since detailed information was not available. A number of conservative assumptions had to be made concerning vital equipment.
Failure Modes	Hot-short calculations used to identify probability of spurious actuation are heavily influenced by analysts' judgment. Detailed data do not exist.
Fire Growth	Fire propagation is based on physical models, and there are large uncertainties about the results of these models. The analysis included consideration of, but not direct data from, tests on Oconee interlocked armor cable.
Fire Suppression	Fire suppression is based on industrywide data and is not necessarily directly representative of the actual characteristics of the fire areas of concern.
Operations Staff Effects	Errors of commission by the control room operators, as instigated by failures in the instrumentation circuits, were not analyzed explicitly. It was judged that the loss of function from fires in the critical areas envelops these potential human errors.
Smoke Propagation	The effects of smoke on the operations staff were not analyzed explicitly.
Flooding from Fire Suppression Activities	The effects of flooding from fire fighting activities were not analyzed explicitly.

TABLE 3-2. AN EXAMPLE FROM LOCATION INVENTORY CODIFICATION TABLES

Location Name: Control Building Patio Area
Designator: FH-F/-5
Building: Fuel Handling Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH						X		AH-E-88A		Fail open on loss of air; not significant.
AH						X		AH-E-88B		Fail open on loss of air; not significant.
ES	C				X					At Elevation 331' 4". We assume 1C-480V ESF valve control center.
AH	B				AH-E-1B				Color-Coded Drawings	It is assumed that power cables for the fans are in trays (Elevation 380').
						AH-E-1C			FHA	It is assumed that power cables for the fans are in trays (Elevation 380').
					AH-E-18A				FHA	It is assumed that power cables for the fans are in trays (Elevation 380').
					AH-E-18B				FHA	It is assumed that power cables for the fans are in trays (Elevation 380').
Instrument	A					X				
	B					X				
EP					480V AC ES-CC-1T					
MU	C				MU-P-1C	MU-P-3C			FHA	
	B					MU-P-2B			FHA	
					MU-V17				FHA	

TABLE 3-3. AN EXAMPLE FROM SOURCE AND MITIGATION TABLES

Location Name: Turbine-Driven Emergency Feedwater Pump Room
Designator: IB-FZ-2
Building: Intermediate Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigation Feature	Reference	
Fire and Smoke	Turbine Bearing Oil System		1-FHA-039	Ionizaton Fire Detector	Fire Hazard Report	
	Cabling		Fire Hazard Report	Location IB-FZ-5 (upstairs) Contains Portable CO^2 Extinquishers (two), Portable H_2O Extinquishers (two), and Hose Protection (two)		
[illegible]	Steam Piping for the EFW Pump	Any Break Upstream of Top Steam Admission Valves	Fire Hazard Report			
[illegible]od	Pipe Section EFW Piping	Any Break Upstream of Pump	2			
Missiles	EFW Turbine Pump			Walls and a Missile Shield Guarding Opening to IB-FZ-3	Plant Walkdown	
Pipe Whip	Steam Piping	Any Break Upstream of Top Steam Admission Valves				

TABLE 3-4. LIST OF HAZARD TYPES AND EXAMPLE SOURCES

Hazard Type	Example Sources
1. High Energy Line Break	Main Feedwater Piping
2. Flood	River Water Piping
3. Fires	Oil-Lubricated Large Pump, Transient Fuels
4. Missiles	Turbine-Driven Auxiliary Feedwater Pump; Pressurized Bottle of Gas
5. Steam	Auxiliary Steam Piping
6. Explosion	Propane Piping
7. Water Jets	Makeup System Piping
8. Water Sprays	Nuclear Services Piping
9. Falling Objects	Crane Equipment
10. Smoke	Electrical Cables

TABLE 3-5. EXAMPLES OF PROPAGATION/MITIGATION FACTORS

Environmental Hazard Type	Examples of Mitigative Factors
1. High Energy Line Break	Walls, Restrainers, Heavy Equipment
2. Flood	Drains, Doorways, Openings
3. Fire	Fire Detectors, Fire Suppression Equipment
4. Missiles	Walls, Doors
5. Steam Leak	Doors, Walls, Penetration Seals, Ducts
6. Explosion	Walls, Doors, Penetration Seals
7. Water Jets	Spray Equipment Construction, Walls, Doors
8. Water Spray	Waterproof Equipment, Walls, Doors
9. Falling Objects	Floor, Gratings
10. Smoke	Walls, Doors, Dampers

TABLE 3-6. AN EXAMPLE FROM SCENARIO TABLES

Location Name: General Area - Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	1. Cable burning due to an electrical short or transient fuel. Localized.				Yes.	3×10^{-3}	(comparison)	The loss of all vital components does not lead to any major events except for loss of several standby trains needed for LOCA mitigation. LOCA not possible from this zone.
		2. Near the boundary.	Open	FH-FZ-1		Yes.	3×10^{-3}	(comparison) MU-P-2C, MU-P-3C AH-E-1B; 480V-AC-ESV and CC1B; BS-P-1B.	
		3. Near the boundary.	Open	AB-FZ-4		Yes.	10^{-3}	No action; subset of AB-FZ-4, scenario 1.	
		4. Near the boundary.	Doors	AB-FZ-2B AB-FZ-2C		Yes.	3×10^{-3}	(comparison) 480V-AC-CC-1B; AH-E-1B; DH-P-1B; MU-P-2C, 3C; BS-P-1B.	
		5. Near the boundary.	Open	AB-FZ-1		No, because only MOV cables are affected, and MOVs are normally in operational position.			

TABLE 3-6 (continued)

Location Name: General Area - Elevation 281'-0"
Designator: AB-F7-5
Building: Auxiliary Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Flood	Pipe Section or Tank	6. Pipe break of closed loops or tanks.	Opening	AB-F7-1 AB-FA-1 AB-FA-2 AB-F7-4 AB-F7-3 FH-F7-1 AB-F7-2A AB-F7-2B AB-F7-2C		Yes.	10^{-2}	(systems) DHR and building spray pumps flooded.	
Steam	Steam Pipe	7. Steam pipe rupture (8-inch line, 6-psig steam pressure).	Openings	Most of Auxiliary Building and Fuel Handling		Yes.	10^{-5}	(comparison) 480V-AC-ESV-CC-1C, 1A, and 1B.	The equipment susceptible to this scenario are ESV CCs, which are very far from the source. The operator, who is on watch 24 hours a day on Elevation 305' 0" dimension, will notice the steam. Conservatively assumed as affected.

TABLE 3-7. AN EXAMPLE FOR IMPACT TABLE

LOCATION NAME: Penetration Area

DESIGNATOR: AB-FZ-4

BUILDING: Auxiliary Building

SCENARIO SUMMARY: Fire, Scenario 1, Fire on the Floor or in Cables; Affects Cables Near the Ceiling; Propagates to AB-FZ-5

Sheet 1 of 2

Systems Lost	Components Affected by the Hazard
NR All Trains	Hot short in the control cable of NR-V-5 (a normally open MOV). This valve is controlled from 480V-ESV-1A. Recovery of this valve not possible because fire is in operator's path. However, an alternate path can be used by opening two MOVs.
RCP Thermal Barrier Cooling	Hot short in the control cable of IC-V-2 (a normally open MOV).
RCP Motor Cooling Letdown Cooling	Affects motor cooling and letdown cooling.
MU All	Damage to control or power cables of MV-V-14A and MV-V-14B (normally closed MOVs).
BS All	Damage to control or power cables of BS-V-1A and BS-V-1B (normally closed MOVs).
IC All	IC-V-3 would fail closed if copper tubing of air line to air operator fails from the fire; hot short in control cables of IC-V-2.
AH-V-1B and AH-V-1C	Hot short in the control cables of these valves (MOVs, normally closed) may open the valve.
MU Trains A and C	Power cables to pumps MV-P-2A and MV-P-2C.
480V-ESV-1A and 480V-ESV-1B	Power feeds to these two electrical cabinets.
CF Trains A and B	Power cables to AH-E-1A and AH-E-1B in the fire zone.

TABLE 3-7 (continued)

Sheet 2 of 2

Systems Lost	Components Affected by the Hazard
HL-1	Valve DH-V3 power cable in the area can be recovered by manual operation of the valve after the fire is put out.
HL2	Valves DH-V7A and DH-V7B power cables in the area can be recovered by manual operation of the valves after the fire is put out.

TABLE 3-8. DISPOSITION OF HAZARD SCENARIOS FROM FIGURES 3-1 THROUGH 3-8

Sheet 1 of 3

Location	Scenario	Hazard	Hazard Frequency	Core Damage	Core Damage Frequency	Disposition
A. IMPORTANT DIRECT CORE DAMAGE SCENARIOS OR SCENARIOS TREATED IN A PLANT MODEL EVENT TREE						
CB-FA-3a	2	Fire	1.4-4	No	-	
CB-FA-3c	1	Fire	1.0-4	Yes	2.0-5	
AB-FZ-6	1	Fire	3.0-5	Yes	3.0-5	
CB-FA-2b	1a	Fire	2.0-5	Yes	2.0-5	
CB-FA-3b	1	Fire	1.0-5	Yes	1.0-5	
CB-FA-2d	1	Fire	5.0-6	Yes	5.0-6	
B. SCENARIOS INCLUDED IN THE SYSTEMS OR INITIATING EVENT FREQUENCY ANALYSIS						
TB-FA-1	16	Flood	1.0-2	No	-	Included in turbine trip IE frequency
TB-FA-1	9	Steam	2.0-3	No	-	Included in SLB IE frequency*
AB-FZ-6a	1	Fire	1.0-3	No	-	Included in SAR, Section 2
FH-FZ-1	4	Fire	1.0-3	No	<<1.0-4	Included in SAR, Section 2
FH-FZ-6	1	Fire	1.0-3	No	-	Included in SAR, Section 6
IB-FZ-1	1	Fire	1.0-3	No	-	Included in SAR, Section 16
CB-FA-2g	1	Fire	1.0-3	No	-	Included in SAR, Section 2
CB-FA-5b	1	Fire	1.0-3	No	-	Included in SAR, Section 6
RB-FZ-1a	1	Fire	1.0-3	No	-	Included in SAR, Section 16
RB-FZ-1c	1	Fire	1.0-3	No	-	Included in SAR, Section 16
DG-FA-1	1	Fire	7.4-4	No	-	Included in SAR, Section 2
DG-FA-1	1	Fire	7.4-4	No	-	Included in SAR, Section 2
FH-FZ-5	1	Fire	3.0-4	No	-	Included in SAR, Section 6
FH-FZ-5	2	Fire	3.0-4	No	-	Included in SAR, Section 6
FH-FZ-5	3	Fire	3.0-4	No	-	Included in SAR, Section 6
AB-FZ-7	3	Fire	1.0-4	No	-	Included in SAR, Section 13
AB-FZ-7	4	Fire	1.0-4	No	-	Included in SAR, Section 13
FH-FZ-5	9	Flood	1.0-4	No	-	Included in SAR, Section 6
IB-FZ-3	3	Flood	1.0-4	No	-	Included in SAR, Section 11
IB-FZ-6	3	Steam	1.0-4	No	-	Initiates loss of MFW at low frequency
IB-FZ-6	4	Steam	1.0-4	No	-	Included in SLB IE frequency*
CB Stairs	1	Flood	1.0-4	No	-	Included in SAR, Section 6
AB-FZ-1	7	Flood	3.0-5	No	-	Produces loss of NS IE at low frequency
AB-FZ-3	1	Fire	3.0-5	No	-	Included in SAR, Section 13
AB-FZ-7	1	Fire	3.0-5	No	-	Included in loss of NS IE frequency*
TB-FA-1	6	Flood	3.0-5	No	-	Included in SAR, Section 6
FH-FZ-2	1	Fire	3.0-5	No	-	Included in SAR, Section 4
CB-FA-5a	1	Fire	3.0-5	No	-	Included in SAR, Section 6
AB-FZ-6	5	Flood	2.0-5	No	-	Included in loss of NS IE frequency*
FH-FZ-6	4	Flood	2.0-5	No	-	Included in SAR, Section 6
IB-FZ-2	3	Flood	2.0-5	No	-	Included in SAR, Section 11
IB-FZ-2	4	Steam	2.0-5	No	-	Included in SLB IE frequency*
RB-FZ-2	11	Steam	2.0-5	No	-	Included in SLB IE frequency*
AB-FZ-4	3	Flood	1.0-5	No	-	Included in SAR, Section 14
AB-FZ-6	7	Flood	1.0-5	No	-	Included in SAR, Section 13
AB-FZ-6	11	Missile	1.0-5	No	-	Included in loss of NR IE frequency*
AB-FZ-7	2	Fire	1.0-5	No	-	Included in loss of NR IE frequency*

*Initiating event frequency calculations shown in Data Analysis Report, Section 3.

TABLE 3-8 (continued)

Sheet 2 of 3

Location	Scenario	Hazard	Hazard Frequency	Core Damage	Core Damage Frequency	Disposition
AB-FZ-7	5	Flood	1.0-5	No	-	Included in SAR, Section 13
TB-FA-1	7	Flood	1.0-5	No	-	Included in SAR, Section 6
FH-FZ-1	6	Steam	1.0-5	No	-	Included in SAR, Section 2
FH-FZ-2	4	Flood	1.0-5	No	-	Included in SAR, Section 6
FH-FZ-5	4	Fire	1.0-5	No	-	Included in SAR, Section 4
FH-FZ-5	5	Fire	1.0-5	No	<1.0-6	Included in SAR, Section 4
FH-FZ-5	6	Fire	1.0-5	No	<1.0-6	Included in SAR, Section 4
FH-FZ-5	7	Fire	1.0-5	No	<1.0-5	Included in SAR, Section 4
CB-FA-1	6	Flood	1.0-5	No	-	Included in SAR, Section 6
RB-FZ-1b	7	Flood	8.0-6	No	-	Included in SAR, Section 13
RB-FZ-1d	11	Steam	8.0-6	No	-	Included in Steamline break frequency*
RB-FZ-1e	11	Steam	8.0-6	No	-	Included in SLB I.E. frequency*
RB-FZ-1a	7	Flood	8.0-6	No	-	Initiates loss of MFW at low frequency
AB-FZ-2a	2	Flood	3.0-6	No	-	Included in SAE, Section 14
AB-FZ-2b	2	Flood	3.0-6	No	-	Included in SAE, Section 14
AB-FZ-2c	2	Flood	3.0-6	No	-	Included in SAE, Section 14
AB-FZ-3	5	Flood	3.0-6	No	-	Included in SAE, Section 14
AB-FA-1	4	Flood	2.0-6	No	-	Included in SAR, Section 14
AB-FA-2	4	Flood	2.0-6	No	-	Included in SAR, Section 14
C. SCENARIOS REQUIRING OTHER FAILURES TO PRODUCE CORE DAMAGE AND RESULTING IN A SCREENING CORE DAMAGE FREQUENCY LESS THAN OR EQUAL TO $3x10^{-6}$						
TB-FA-1	2	Fire	1.0-2	No	4.8-7	
RB-FZ-1d	1	Fire	1.0-2	No	4.6-7	
IB-FZ-2	1	Fire	1.0-3	No	5.2-7	
IB-FZ-3	1	Fire	1.0-3	No	3.5-9	
DB-FA-2a	1a	Fire	1.0-3	No	3.0-6	
RB-FZ-1b	1	Fire	1.0-3	No	-	
RB-FZ-1e	1	Fire	1.0-3	No	-	
AB-FZ-5	1	Fire	3.0-4	No	9.0-7	
AB-FZ-5	2	Fire	3.0-4	No	9.0-7	
AB-FA-1	1	Fire	3.0-4	No	<<8.0-7	
AB-FA-2	1	Fire	3.0-4	No	<<8.0-7	
AB-FZ-5	4	Fire	1.4-4	No	1.4-6	
AB-FZ-5	6	Flood	1.0-4	No	<2.0-7	
TB-FA-1	12	Explosion	1.0-4	No	5.6-7	
ISPH-FZ-1	4	Flood	1.0-4	No	5.0-7	
ISPH-FZ-2	4	Flood	1.0-4	No	5.0-7	
FH-FZ-1	7	Flood	1.0-4	No	<2.0-7	
CB-FA-2a	4	Fire	1.0-4	No	1.0-7	
CB-FA-2a	3	Fire	6.0-5	No	4-7	
TB-FA-1	1	Fire	5.0-5	No	2.6-6	
TB-FA-1	3	Fire	3.0-5	No	1.7-7	
FH-FZ-2	2	Fire	3.0-5	No	3.0-6	
IB-FZ-2	6	Steam	2.0-5	No	6-7	
CB-FA-2e	1	Fire	2.0-5	No	2.0-6	

*Initiating event frequency calculations shown in Data Analysis Report, Section 3.

TABLE 3-8 (continued)

Sheet 3 of 3

Location	Scenario	Hazard	Hazard Frequency	Core Damage	Core Damage Frequency	Disposition
RB-FZ-1c	8	Flood	2.0-5	No	6-7	
CB-FA-2c	1a	Fire	1.5-5	No	2-6	
AB-FZ-4	7	Flood	1.0-5	No	1.4-6	
AB-FZ-7	7	Missile	1.0-5	Yes	6-7	
AB-FZ-4	10	Steam	1.0-5	Yes	3.0-7	
AB-FZ-5	7	Steam	1.0-5	No	3-6	
AB-FZ-5	12	Missile	1.0-5	No	3-6	
TB-FA-1	15	Steam	1.0-5	No	1.0-6	
ISPH-FZ-1	5	Spray	1.0-5	No	5.0-7	
ISPH-FZ-2	5	Spray	1.0-5	No	5.0-7	
FH-FZ-1	1	Fire	1.0-5	No	1.0-6	
FH-FZ-1	2	Fire	1.0-5	No	5-7	
CB-FA-2c	6	Fire	1.0-5	No	1.0-7	
RB-FZ-1d	8	Flood	8.0-6	No	–	Partial loss of FW at low frequency
RB-FZ-1e	8	Flood	8.0-6	No	–	Partial loss of FW at low frequency
CB-FA-2f	1	Fire	6.0-6	No	1.1-6	Train B of some safety systems, Train A of batteries
TB-FA-1	8	Spray	5.0-6	No	2.8-8	Initiates LOSP at low frequency
TB-FA-1	11	Missile	5.0-6	No	2.28-8	Initiates LOSP at low frequency
AB-FZ-7	8	Missile	3.0-6	No	3.0-7	
D. SCREENING HAZARD FREQUENCY LESS THAN OR EQUAL TO 3×10^{-6}						
AB-FZ-6a	2	Fire	3.0-6	Yes	3.0-6	
ISPH-FZ-1	2	Fire	3.0-6	Yes	3.0-6	
ISPH-FZ-2	4	Fire	3.0-6	Yes	3.0-6	
FH-FZ-1	3	Fire	3.0-6	Yes	3.0-6	
FH-FZ-1	5	Fire	3.0-6	Yes	3.0-6	
CB-FA-2b	3	Fire	3.0-6	Yes	3.0-6	
CB-FA-3a	3	Fire	3.0-6	Yes	3.0-6	
CB-FA-4b	1	Fire	3.0-6	Yes	3.0-6	
CB-FA-1	1	Fire	3.0-6	Yes	3.0-6	
AB-FZ-4	1	Fire	2.0-6	Yes	2.0-6	
CB-FA-2d	3	Fire	2.0-6	Yes	2.0-6	
CB-FA-2d	4	Fire	2.0-6	Yes	2.0-6	
CB-FA-3d	1	Fire	2.0-6	Yes	<2.0-6	
AB-FZ-1	4	Flood	1.0-6	Yes	1.0-6	
AB-FZ-4	9	Flood	1.0-6	Yes	<<1.0-6	
AB-FZ-6	2	Fire	1.0-6	Yes	1.0-6	
AB-FZ-6	4	Fire	1.0-6	Yes	1.0-6	
AB-FZ-7	6	Flood	1.0-6	No	1.0-7	
ISPH-FZ-1	1	Fire	1.0-6	Yes	1.0-6	
ISPH-FZ-2	1	Fire	1.0-6	Yes	1.0-6	
FH-FZ-1	8	Missile	1.0-6	Yes	1.0-6	
FH-FZ-6	2	Fire	9.0-7	No	–	Initiates loss of CBV at low frequency
CB-FA-3b	3	Fire	5.0-7	Yes	5.0-7	Results in SBO at low frequency

*Initiating event frequency calculations shown in Data Analysis Report, Section 3.

LOCATION DESIGNATOR:* AB-FZ-1

HAZARD SCENARIO: 4

HAZARD: Flood

REMARKS: Pipe break in nuclear river.

ANNUAL FREQUENCY OF HAZARD: 1.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All, NS/All, NR/All

CORE DAMAGE: Yes

PLANT IMPACT:

Compares with [LNS (1.3-3) + LRW(1.0-3)]*[HPA-1(2.7-3) + HIA-1(3.0-4)] = 6.9-6 per Year

CORE DAMAGE FREQUENCY: 1.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-1. HAZARD SCENARIO SHEETS FOR THE AUXILIARY BUILDING (Sheet 1 of 38)

LOCATION DESIGNATOR:* AB-FZ-1

HAZARD SCENARIO: 7

HAZARD: Flood

REMARKS: Break in nuclear service pipe or heat exchanger.

ANNUAL FREQUENCY OF HAZARD: 3.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: NS/All

CORE DAMAGE:

PLANT IMPACT: Compares with NS-1 (1.1-3).

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 2 of 38)

LOCATION DESIGNATOR:* AB-FZ-2a

HAZARD SCENARIO: 2

HAZARD: Flood

REMARKS: Pipe break in BWST-related piping.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All, BWST

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of BWST failure analysis; see Section 14 of the System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 3 of 38)

LOCATION DESIGNATOR:* AB-FZ-2b

HAZARD SCENARIO: 2

HAZARD: Flood

REMARKS: Pipe break in BWST-related piping.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All, BWST

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of BWST failure; see Section 14 of the System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 4 of 38)

LOCATION DESIGNATOR:* AB-FZ-2c

HAZARD SCENARIO: 2

HAZARD: Flood

REMARKS: Pipe break in BWST-related piping.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All, BWST

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of BWST failure; see Section 14 of the System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 5 of 38)

LOCATION DESIGNATOR:* AB-FZ-3

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables to MU valves V4, V5, and V32 affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All

CORE DAMAGE: No

PLANT IMPACT: Compares with HPA-1 (2.7-3) + HIA-1 (3.0-11) = 2.7-3

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of HP injection failure; see Section 13 of the System Analysis report.

FIGURE 3-1 (continued)
(Sheet 6 of 38)

LOCATION DESIGNATOR:* AB-FZ-3

HAZARD SCENARIO: 5

HAZARD: Flood

REMARKS: Pipe break in BWST-related piping.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All, BWST

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of BWST failure; see Section 14 of the System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 7 of 38)

LOCATION DESIGNATOR:* AB-FZ-4

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 2.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

ESV/A, ESV/B, NR/All, IC/All, BS/All, RCP Seal Failure, MU/All, 48-inch Purge Line, CF/A, CF/B

CORE DAMAGE: Yes

PLANT IMPACT: Leads to damage state 3A, HL-1, HL-2, and fan coolers lost.

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 8 of 38)

LOCATION DESIGNATOR:* AB-FZ-4

HAZARD SCENARIO: 3

HAZARD: Flood

REMARKS: Pipe break in BWST-related piping.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All, BSWT

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of BWST failure; see Section 14 of the System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 9 of 38)

LOCATION DESIGNATOR:* AB-FZ-4

HAZARD SCENARIO: 7

HAZARD: Flood

REMARKS: Water jets affecting cables in the area.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All, DH/All, BS/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY: 1.0-5 x EF-1 (1.4-1) = 1.4-6 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 10 of 38)

LOCATION DESIGNATOR:* AB-FZ-4

HAZARD SCENARIO: 9

HAZARD: Flood

REMARKS: DHR pipe break during hot leg recirculation.

ANNUAL FREQUENCY OF HAZARD: 1.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

V-Scenario, NR/B, ESV/A, ESV/B, PORV, MU/All, BS/All, DH/All

CORE DAMAGE: Yes

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Not important because is has to occur when a LOCA mitigation is in progress.

FIGURE 3-1 (continued)
(Sheet 11 of 38)

LOCATION DESIGNATOR:* AB-FZ-4

HAZARD SCENARIO: 10

HAZARD: Steam

REMARKS:

MUPS letdown pipe break during normal operation. The break will be isolated by the fail-closed type isolation valves automatically or by the operators after the leak source is identified. The leak rate would be very small, therefore allowing a long time for recovery.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

V-Scenario, NR/B, ESV/A, ESV/B, PORV, MU/All, BS/All

CORE DAMAGE: Yes

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

1.0-5 x [failure to isolate 0.03 (estimate)] = 3.0-7.

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 12 of 38)

LOCATION DESIGNATOR:* AB-FZ-5

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD:

ESV/B, NS/A, AH1/B, DH/A, DH/B, DC/A, MU/C, BS/B, BWST Makeup, DH-VGA, DH-VGB, NR/All, IC/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

3.0-4 x [(HPA-1)(HPB)(3.0-3)] = 9.0-7 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 13 of 38)

LOCATION DESIGNATOR:* AB-FZ-5

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD:

ESV/B, AH1/B, MU/C, BS/B, BWST Makeup, DH-V/6A, DH-V/6B, NR/All, IC/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

3.0-4 x [HPA-1(HPB)(3.0-3)] = 9.0-7 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 14 of 38)

LOCATION DESIGNATOR:* AB-FZ-5

HAZARD SCENARIO: 4

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.4-4

SYSTEM/TRAIN AFFECTED BY HAZARD:

ESV/B, AH1/B, DH/B, MU/C, BS/B, BWST Makeup

CORE DAMAGE: No

PLANT IMPACT: Train B valves, assume reactor trip occurs.

CORE DAMAGE FREQUENCY:

1.4-4 x [EF(GE)(0.1) x CV(GB)(0.1)] = 1.4-6 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 15 of 38)

LOCATION DESIGNATOR:* AB-FZ-5

HAZARD SCENARIO: 6

HAZARD: Flood

REMARKS: Flooding from any one of the sources in the area.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: DH/All, BS/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

Compares with simultaneous failure of both systems [DHR(3-3)] x [CS(1.6-3)] = 4.0-6; the equivalent unavailability of the flooding event is 1.0-4/365 = 2.0-7.

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 16 of 38)

LOCATION DESIGNATOR:* AB-FZ-5

HAZARD SCENARIO: 7

HAZARD: Steam

REMARKS: Auxiliary steam pipe break.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: ESV/All

CORE DAMAGE: No

PLANT IMPACT: Vital MOV power and control lost.

CORE DAMAGE FREQUENCY: 1.0-5 x [manual valve operation (0.3)] = 3-6

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 17 of 38)

LOCATION DESIGNATOR:* AB-FZ-5

HAZARD SCENARIO: 12

HAZARD: Missile

REMARKS: Missile impacting cables and electrical cabinets.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

FREQUENCY EVALUATION:

SYSTEM/TRAIN AFFECTED BY HAZARD:

ESV/All, NS/A, AH1/B, DH/All, DC/A, MU/C. BS/All

CORE DAMAGE: No

PLANT IMPACT:

Vital MOV power and control lost; one train of several systems, DH/All, BS/All

CORE DAMAGE FREQUENCY: 1.0-5 x [manual valve operation (0.3)] = 3-6

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 18 of 38)

LOCATION DESIGNATOR:* AB-FZ-6

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

ESV/A, RCP Motor Cooling, NR/All, DC/B, IC/B, NS/B, NS/C, HPI/All, CPI/A, RCP Seal Injection, Thermal Barrier of at Least One Pump, BS/A

CORE DAMAGE: Yes

PLANT IMPACT:

CORE DAMAGE FREQUENCY: 3.0-5

FURTHER ACTIONS:

Importance to be determined; impact is recoverable.

FIGURE 3-1 (continued)
(Sheet 19 of 38)

LOCATION DESIGNATOR:* AB-FZ-6

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables affected by fire ESV/B smoke damage.

ANNUAL FREQUENCY OF HAZARD: 1.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

ESV/A, ESV/B, NS/B, NS/C, DC/B, IC/B, DH/All, BS/All, MU/All

CORE DAMAGE: Yes

PLANT IMPACT:

CORE DAMAGE FREQUENCY: 1.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 20 of 38)

LOCATION DESIGNATOR:* AB-FZ-6

HAZARD SCENARIO: 4

HAZARD: Fire

REMARKS: Cables affected by fire in two zones.

ANNUAL FREQUENCY OF HAZARD: 1.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

ESV/A, NS/All, AH1/A, DH/A, DC/A, DC/B, MU/A

CORE DAMAGE: Yes

PLANT IMPACT:

CORE DAMAGE FREQUENCY: 1.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 21 of 38)

LOCATION DESIGNATOR:* AB-FZ-6

HAZARD SCENARIO: 5

HAZARD: Flood

REMARKS: Pipe break in nuclear services closed.

ANNUAL FREQUENCY OF HAZARD: 2.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: NS/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of NS initiating event analysis; see Data Analysis Report, Section 3.

FIGURE 3-1 (continued)
(Sheet 22 of 38)

LOCATION DESIGNATOR:* AB-FZ-6

HAZARD SCENARIO: 7

HAZARD: Flood

REMARKS: MUPS pipe break.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

Seal Injection, MUPS to Loop B and Loop D RCPs

CORE DAMAGE:

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event SE analysis; see Section 13 of the System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 23 of 38)

LOCATION DESIGNATOR:* AB-FZ-6

HAZARD SCENARIO: 11

HAZARD: Missile

REMARKS: Missiles (transient sources) in northern part of zone.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: ESV/A, NR/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Part of NR Analysis.

FIGURE 3-1 (continued)
(Sheet 24 of 38)

LOCATION DESIGNATOR:* AB-FZ-6a

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Electrical cabinet affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD: ESV/B

CORE DAMAGE: No

PLANT IMPACT: Train B of all systems.

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event GB system analysis; see Section 2 of the System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 25 of 38)

LOCATION DESIGNATOR:* AB-FZ-6a

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: ESV/B heat damage; ESV/A smoke damage.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: ESV/A, ESV/B, MU/All, DH/All, BS/All

CORE DAMAGE: Yes

PLANT IMPACT:

CORE DAMAGE FREQUENCY: 3.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 26 of 38)

LOCATION DESIGNATOR:* AB-FZ-7

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire on floor.

ANNUAL FREQUENCY OF HAZARD: 3.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: AH15/A, AH15/B

CORE DAMAGE:

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of NS initiating event analysis; see Data Analysis Report, Section 3.

FIGURE 3-1 (continued)
(Sheet 27 of 38)

LOCATION DESIGNATOR:* AB-FZ-7

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables affected by fire on top of slab.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: AH15/A, AH15/B

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of NS initiating event analysis; see Data Analysis Report, Section 3.

FIGURE 3-1 (continued)
(Sheet 28 of 38)

LOCATION DESIGNATOR:* AB-FZ-7

HAZARD SCENARIO: 3

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: IC/A, IC/B

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event SE analysis; see Section 13 of the System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 29 of 38)

LOCATION DESIGNATOR:* AB-FZ-7

HAZARD SCENARIO: 4

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: NS/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of NS analysis, Section 4 of Systems Analysis Report.

FIGURE 3-1 (continued)
(Sheet 30 of 38)

LOCATION DESIGNATOR:* AB-FZ-7

HAZARD SCENARIO: 5

HAZARD: Flood

REMARKS: Pipe break.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: IC/A, IC/B, DC/A

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event SE analysis; see Section 13 of System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 31 of 38)

LOCATION DESIGNATOR:* AB-FZ-7

HAZARD SCENARIO: 6

HAZARD: Flood

REMARKS: IC pipe leak spray on IC and DC pumps.

ANNUAL FREQUENCY OF HAZARD: 1.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: IC/A, IC/B, DC/A, DC/B

CORE DAMAGE: No

PLANT IMPACT: DC and IC lost, and assume RT.

CORE DAMAGE FREQUENCY: 1.0-6 x [mitigation (<0.1)] = 1.0-7

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 32 of 38)

LOCATION DESIGNATOR:* AB-FZ-7

HAZARD SCENARIO: 7

HAZARD: Missile

REMARKS: Missile impacting cables and pumps.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: NS/All, DC/All

CORE DAMAGE: Yes

PLANT IMPACT: NS and DC lost.

CORE DAMAGE FREQUENCY: 1.0-5 x [INA 0.06] = 6-7 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 33 of 38)

LOCATION DESIGNATOR:* AB-FZ-7

HAZARD SCENARIO: 8

HAZARD: Missile

REMARKS: Missile impacting cables and pumps.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: IC/All, DC/All

CORE DAMAGE: No

PLANT IMPACT: DC and IC lost, plus assume RT.

CORE DAMAGE FREQUENCY: 3.0-6 x [mitigation (0.1)] = 3.0-7

FURTHER ACTIONS: Not important.

FIGURE 3-1 (continued)
(Sheet 34 of 38)

LOCATION DESIGNATOR:* AB-FA-1

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: DH/A, BS-V-3A

CORE DAMAGE: No

PLANT IMPACT: Fails train A of DHR and spray B.

CORE DAMAGE FREQUENCY: $<< 8 \times 10^{-7}$

FURTHER ACTIONS:

Not important because $3.0 \times 10^{-4}/365 = 8 \times 10^{-7}$ and this is much less than $[CS - 1(\overline{GA}/\overline{GB})(3.4 \times 10^{-2})] \times [DH - 1(\overline{GA}/\overline{GB})(10^{-2})]$ $\approx 3 \times$ 10-4.

FIGURE 3-1 (continued)
(Sheet 35 of 38)

LOCATION DESIGNATOR:* AB-FA-1

HAZARD SCENARIO: 4

HAZARD: Flood

REMARKS: Pipe break.

ANNUAL FREQUENCY OF HAZARD: 2.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: BWST Empty, BS/All, DH/All

CORE DAMAGE: No

PLANT IMPACT: Loss of BWST.

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of BWST analysis; see Section 14 of System Analysis Report.

FIGURE 3-1 (continued)
(Sheet 36 of 38)

LOCATION DESIGNATOR:* AB-FA-2

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: DH/B, BS-V-3B

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY: $<< 8 \times 10^{-7}$

FURTHER ACTIONS:

Not important because $3.0 \times 10^{-4}/365 = 8 \times 10^{-7}$ and this is much smaller than $[CS - 1(\overline{GA}/\overline{GB})(3.4 \times 10^{-2})] \times [DH - 1(\overline{GA}/\overline{GB})(10^{-2})] \approx 3 \times 10^{-4}$.

FIGURE 3-1 (continued)
(Sheet 37 of 38)

LOCATION DESIGNATOR:* AB-FA-2

HAZARD SCENARIO: 4

HAZARD: Flood

REMARKS: DHR or RBS pipe break.

ANNUAL FREQUENCY OF HAZARD: 2.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: BS/All, DH/All, BSWT

CORE DAMAGE: No

PLANT IMPACT: Loss of BWST

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of BWST failure; see Section 14 of the Systems Analysis Report.

FIGURE 3-1 (continued)
(Sheet 38 of 38)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 5.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: LOSP, NR/B, ESV/C, MU/B, AH1/C

CORE DAMAGE: No

PLANT IMPACT: One train of NR and LOSP.

CORE DAMAGE FREQUENCY: 5.0-5 x [NSC(5.2-2)] = 2.6-6

FURTHER ACTIONS: Not important.

FIGURE 3-2. HAZARD SCENARIO SHEETS FOR THE TURBINE BUILDING
(Sheet 1 of 11)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-2

SYSTEM/TRAIN AFFECTED BY HAZARD: TT, Loss of Main Feedwater, TBV/A, TBV/B

CORE DAMAGE: No

PLANT IMPACT: MF-1 = 1.0

CORE DAMAGE FREQUENCY:

1.0-2 x [HPI + PSV + PORV] (estimate 0.1) x [EF-1(4.6-4) + SD-1(1.5-5)] = 4.8-7 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-2 (continued)
(Sheet 2 of 11)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 3

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: LOOP, NR/B, ESV/C, MU/B, AH1/C

CORE DAMAGE: No

PLANT IMPACT: Failures other than LOOP not important.

CORE DAMAGE FREQUENCY:

3.0-5 x [GA/OP(0.07) x GB/OP,GA(0.08)] = 1.7 x 10-7

FURTHER ACTIONS: Not important.

FIGURE 3-2 (continued)
(Sheet 3 of 11)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 6

HAZARD: Flood

REMARKS: Very large flood and rollup door failure.

ANNUAL FREQUENCY OF HAZARD: 3.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: Control Building Chiller Pumps

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Part of CB-HVAC, Section 6 of System Analysis Report.

FIGURE 3-2 (continued)
(Sheet 4 of 11)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 7

HAZARD: Flood

REMARKS: Large flood and rollup door failure.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: Control Building Chiller Pumps

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Part of CB-HVAC, Section 6 of System Analysis Report.

FIGURE 3-2 (continued)
(Sheet 5 of 11)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 8

HAZARD: Spray

REMARKS: Spray from fire protection sources.

ANNUAL FREQUENCY OF HAZARD: 5.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: LOSP, TT

CORE DAMAGE: No

PLANT IMPACT: LOSP (unrecoverable).

CORE DAMAGE FREQUENCY:

5.0-6 x [GA/OP(0.07) x GB/OP,GA(0.08)] = 2.8 x 10-8

FURTHER ACTIONS: Not important.

FIGURE 3-2 (continued)
(Sheet 6 of 11)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 9

HAZARD: Steam

REMARKS: Steam from main steam line break.

ANNUAL FREQUENCY OF HAZARD: 2.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD: MS, LOSP, TT, MF

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of steam line break in turbine building initiating event.

FIGURE 3-2 (continued)
(Sheet 7 of 11)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 11

HAZARD: Missile

REMARKS: Missiles from transient or in situ sources.

ANNUAL FREQUENCY OF HAZARD: 5.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: LOOP (unrecoverable)

CORE DAMAGE: No

PLANT IMPACT:

5.0-6 x [GA/OP(0.07) x GA/OP,GA(0.08)] = 2.28 x 10-8

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Not important.

FIGURE 3-2 (continued)
(Sheet 8 of 11)

<u>LOCATION DESIGNATOR</u>:* TB-FA-1

<u>HAZARD SCENARIO</u>: 12

<u>HAZARD</u>: Explosion

<u>REMARKS</u>: Primarily hydrogen explosion.

<u>ANNUAL FREQUENCY OF HAZARD</u>: 1.0-4

<u>SYSTEM/TRAIN AFFECTED BY HAZARD</u>: LOOP (unrecoverable), TT

<u>CORE DAMAGE</u>: No

<u>PLANT IMPACT</u>: LOOP (unrecoverable).

<u>CORE DAMAGE FREQUENCY</u>:

1.0-9 x [GA/OP(0.07) x GB/OP and GA(0.08)] = 5.6-7

<u>FURTHER ACTIONS</u>: Not important.

FIGURE 3-2 (continued)
(Sheet 9 of 11)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 15

HAZARD: Steam

REMARKS: Steam and flood from main feedwater pipe.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: MF, NR/B, MU/B, AH1/C, LOOP

CORE DAMAGE: No

PLANT IMPACT: LOOP recoverable.

CORE DAMAGE FREQUENCY: 1.0-5 x [mitigation estimate (0.1)] = 1.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-2 (continued)
(Sheet 10 of 11)

LOCATION DESIGNATOR:* TB-FA-1

HAZARD SCENARIO: 16

HAZARD: Flood

REMARKS: Flood confined to turbine building.

ANNUAL FREQUENCY OF HAZARD: 1.0-2

SYSTEM/TRAIN AFFECTED BY HAZARD: Turbine Trip

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Part of turbine trip initiating event.

FIGURE 3-2 (continued)
(Sheet 11 of 11)

LOCATION DESIGNATOR:* ISPH-FZ-1

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: 1R-SWGR, 1T-SWGR

CORE DAMAGE: Yes

PLANT IMPACT:

Loss of all screen house switchgears; core damage may result.

CORE DAMAGE FREQUENCY: 1.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-3. HAZARD SCENARIO SHEETS FOR THE INTAKE SCREEN AND PUMP HOUSE (Sheet 1 of 8)

LOCATION DESIGNATOR:* ISPH-FZ-1

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables affected by missile.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: 1T-SWGR, 1R-SWGR

CORE DAMAGE: Yes

PLANT IMPACT:

Loss of all screen house switchgears; core damage may result.

CORE DAMAGE FREQUENCY: 3.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-3 (continued)
(Sheet 2 of 8)

LOCATION DESIGNATOR:* ISPH-FZ-1

HAZARD SCENARIO: 4

HAZARD: Flood

REMARKS: Spill rate large, and flood in ISPH-FZ-2 not severe.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: RR/B, DR/B, NR/B

CORE DAMAGE: No

PLANT IMPACT: Train B of river water system.

CORE DAMAGE FREQUENCY:

1.0-4 x [NSS(0.05)] x [EF(0.1 estimate)] = 5.0-7

FURTHER ACTIONS: Not important.

FIGURE 3-3 (continued)
(Sheet 3 of 8)

LOCATION DESIGNATOR:* ISPH-FZ-1

HAZARD SCENARIO: 5

HAZARD: Spray

REMARKS: Spill rate large, and flood in ISPH-FZ-2 severe.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: RR/B, DR/B, NR/B

CORE DAMAGE: No

PLANT IMPACT: Train B of river water systems.

CORE DAMAGE FREQUENCY: 1.0-5 x [NSC(0.05)] = 5.0-7 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-3 (continued)
(Sheet 4 of 8)

LOCATION DESIGNATOR:* ISPH-FZ-2

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: 1R-SWGR, 1T-SWGR

CORE DAMAGE: Yes

PLANT IMPACT:

Loss of all screen house switchgears; core damage may result.

CORE DAMAGE FREQUENCY: 1.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-3 (continued)
(Sheet 5 of 8)

LOCATION DESIGNATOR:* ISPH-FZ-2

HAZARD SCENARIO: 4

HAZARD: Fire

REMARKS: Cables affected by missile.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: 1T-SWGR, 1R-SWGR

CORE DAMAGE: Yes

PLANT IMPACT:

Loss of all screen house switchgears; core damage may result.

CORE DAMAGE FREQUENCY: 3.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-3 (continued)
(Sheet 6 of 8)

LOCATION DESIGNATOR:* ISPH-FZ-2

HAZARD SCENARIO: 4

HAZARD: Flood

REMARKS: Spill rate large, and flood in ISPH-FZ-1 not severe.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: RR/A, DR/A, NR/A, NR/C

CORE DAMAGE: No

PLANT IMPACT: Train A of river water system.

CORE DAMAGE FREQUENCY:

1.0-4 x [NSS(0.05)] x [EF(0.1 estimate)] = 5.0-7

FURTHER ACTIONS: Not important.

FIGURE 3-3 (continued)
(Sheet 7 of 8)

LOCATION DESIGNATOR:* ISPH-FZ-2

HAZARD SCENARIO: 5

HAZARD: Spray

REMARKS: Spill rate large, and flood in ISPH-FZ-1 severe.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: RR/A, DR/A, NR/A, NR/C

CORE DAMAGE: No

PLANT IMPACT: Train A of river water systems.

CORE DAMAGE FREQUENCY: 1.0-5 x [NSC(0.05)] = 5.0-7

FURTHER ACTIONS: Not important.

FIGURE 3-3 (continued)
(Sheet 8 of 8)

LOCATION DESIGNATOR:* FH-FZ-1

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/A, MU/B, NS/A, NS/B, NR/A, ESV/A, AH1/A, DH/A, DC/A, BS/A

CORE DAMAGE: No

PLANT IMPACT: Train A of several components and train B of MU and NS.

CORE DAMAGE FREQUENCY: 1.0-5 x [other train, estimate, 0.1] = 1.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-4. HAZARD SCENARIO SHEETS FOR THE FUEL HANDLING BUILDING (Sheet 1 of 22)

LOCATION DESIGNATOR:* FH-FL-1

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All, NS/A, ESV/A, AH1/A, DH/All, DC/A

CORE DAMAGE: No

PLANT IMPACT: MU/All, DH/All plus partial loss of several systems.

CORE DAMAGE FREQUENCY: 1.0-5 x [NSB, NSC) (0.15)] = 5-7

FURTHER ACTIONS: Not important.

FIGURE 3-4 (continued)
(Sheet 2 of 22)

LOCATION DESIGNATOR:* FH-FZ-1

HAZARD SCENARIO: 3

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: 1T-SWGR, 1R-SWGR, BS/All

CORE DAMAGE: Yes

PLANT IMPACT:

Loss of all screen house switchgears, spray B; core damage may result.

CORE DAMAGE FREQUENCY: 3.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-4 (continued)
(Sheet 3 of 22)

LOCATION DESIGNATOR:* FH-FZ-1

HAZARD SCENARIO: 4

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD:

ESV/C, NS/B, AH/C, MU/B, Instrumentation Channels

CORE DAMAGE: No

PLANT IMPACT:

C-valves lost, train B of NS + MU.

CORE DAMAGE FREQUENCY: 1.0-3 x [CVC (0.1)] = 1.0-4

FURTHER ACTIONS:

Part of Top Event 1C analysis; see Section 2 of System Analysis Report.

FIGURE 3-4 (continued)
(Sheet 4 of 22)

LOCATION DESIGNATOR:* FH-FZ-1

HAZARD SCENARIO: 5

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

1T-SWGR, 1R-SWGR, ESV/A, ESV/B, MU/All, DC/A, BS/All, AH1/A, AH1/B, NS/A, NS/B

CORE DAMAGE: Yes

PLANT IMPACT:

Loss of all screen house switchgears, HPI and spray B; core damage may result.

CORE DAMAGE FREQUENCY: 3.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-4 (continued)
(Sheet 5 of 22)

LOCATION DESIGNATOR:* FH-FZ-1

HAZARD SCENARIO: 6

HAZARD: Steam

REMARKS: Auxiliary steam pipe break.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: ESV/C

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event 1C analysis; see Section 2 of System Analysis Report.

FIGURE 3-4 (continued)
(Sheet 6 of 22)

LOCATION DESIGNATOR:* FH-FZ-1

HAZARD SCENARIO: 7

HAZARD: Flood

REMARKS: Flood from fire protection and seal injection piping.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: DH/All, BS/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

Compares with simultaneous failure of both systems [DH(3.0-3)] x [CS(1.6-3)] = 4.0-6; the equivalent unavailability of the flooding event is 1.0-4/365 = 2.0-7.

FURTHER ACTIONS: Not important.

FIGURE 3-4 (continued)
(Sheet 7 of 22)

LOCATION DESIGNATOR:* FH-FZ-1

HAZARD SCENARIO: 8

HAZARD: Missile

REMARKS: Cables affected by missile.

ANNUAL FREQUENCY OF HAZARD: 1.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

1T-SWGR, 1R-SWGR, ESV/A, ESV/B, MU/All, DC/A, BS/All, AH1/A, AH1/B, NS/A, NS/B

CORE DAMAGE: Yes

PLANT IMPACT:

Loss of all screen house switchgears, spray D; core damage may result.

CORE DAMAGE FREQUENCY: 1.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-4 (continued)
(Sheet 8 of 22)

LOCATION DESIGNATOR:* FH-FZ-2

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/C, NS/C, NR/C, ESV/B, 1T-SWGR, AH1/B, AH18/A, AH18/B, Train B of DH, DR, DC, IC, and RR

CORE DAMAGE: No

PLANT IMPACT: Trains B and C of several systems.

CORE DAMAGE FREQUENCY:

3.0-5 per year x [one failure in train A equipment, 0.1 estimate)] = 3.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-4 (continued)
(Sheet 9 of 22)

LOCATION DESIGNATOR:* FH-FZ-2

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/C, NS/C, NR/C

CORE DAMAGE:

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of NS/NR analysis; not important because two additional pumps must fail and their combined unavailability is less than 0.01; thus, 0.01 x 10-5 = 1.0-7 per year.

FIGURE 3-4 (continued)
(Sheet 10 of 22)

LOCATION DESIGNATOR:* FH-FZ-2

HAZARD SCENARIO: 4

HAZARD: Flood

REMARKS: Fire protection pipe break.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: Control Building Chiller Pumps

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC, Section 6 of System Analysis Report.

FIGURE 3-4 (continued)
(Sheet 11 of 22)

LOCATION DESIGNATOR:* FH-FZ-5

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire on Elevation 380'0".

ANNUAL FREQUENCY OF HAZARD: 3.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: AH18A, AH18B

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC; not important becase normal redundant fans AH-E-17A and AH-E-17B remain unaffected.

FIGURE 3-4 (continued)
(Sheet 12 of 22)

LOCATION DESIGNATOR:* FH-FZ-5

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables affected by fire on Elevation 355'0".

ANNUAL FREQUENCY OF HAZARD: 3.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: AH18A, AH18B

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC; not important because normal redundant fans AH-E-17A and AH-E-17B remain unaffected by this scenario.

FIGURE 3-4 (continued)
(Sheet 13 of 22)

LOCATION DESIGNATOR:* FH-FZ-5

HAZARD SCENARIO: 3

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: AH18/A, AH18/B, Instrumentation Channels

CORE DAMAGE: No

PLANT IMPACT:

Impact on instrumentation not important to accident sequences.

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC analysis; not important because redundant normal duty supply fans AH-E-17A and AH-E-17B are not affected by this scenario and one of four is needed for success.

FIGURE 3-4 (continued)
(Sheet 14 of 22)

LOCATION DESIGNATOR:* FH-FZ-5

HAZARD SCENARIO: 4

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

AH18/A, AH18/B, NR/B, NR/C, Instrumentation Channels

CORE DAMAGE: No

PLANT IMPACT:

Part of CB-HVAC lost, and NR partial loss; instrumentation not important.

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of NR; CB-HVAC failure unlikely because redundant normal supply fans AH-E-17A and AH-E-17B remain unaffected.

FIGURE 3-4 (continued)
(Sheet 15 of 22)

LOCATION DESIGNATOR:* FH-FZ-5

HAZARD SCENARIO: 5

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

NR/B, NR/C, RCP Rack A, RCP Rack B, Transfer Switch for 1C Valves

CORE DAMAGE: No

PLANT IMPACT:

NR system affected; RCP rack failure not important to our study.

CORE DAMAGE FREQUENCY: < 0.1 x 1.0-5 = 1.0-6

FURTHER ACTIONS:

Part of NR study: not important because the unaffected pump train has an unavailability less than 0.1.

FIGURE 3-4 (continued)
(Sheet 16 of 22)

LOCATION DESIGNATOR:* FH-FZ-5

HAZARD SCENARIO: 6

HAZARD: Fire

REMARKS: Cables affected by fire; cabinets affected by smoke.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

NR/B, NR/C, CRD Cables, Cabinets XCL, XCC, XCR, XPL, and XPCR, PS-1

CORE DAMAGE: No

PLANT IMPACT: NR system affected; other failures not important.

CORE DAMAGE FREQUENCY: < 0.1 x 1.0-5 = 1.0-6

FURTHER ACTIONS:

Part of NR study; not important because it requires an additional failure which has an unavailability less than 0.1.

FIGURE 3-4 (continued)
(Sheet 17 of 22)

LOCATION DESIGNATOR:* FH-FZ-5

HAZARD SCENARIO: 7

HAZARD: Fire

REMARKS: Cabinets affected by smoke.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

NR/B, NR/C, CRD Cables, Cabinets XCL, XCC, XCR, XPL, and XPCR, PS-1

CORE DAMAGE: No

PLANT IMPACT: NR systems affected; other failures not important.

CORE DAMAGE FREQUENCY: < 0.1 x 1.0-5 = 1.0-6

FURTHER ACTIONS:

Part of NR study; not important because an additional failure must occur that has unavailability less than 0.10.

FIGURE 3-4 (continued)
(Sheet 18 of 22)

LOCATION DESIGNATOR:* FH-FZ-5

HAZARD SCENARIO: 9

HAZARD: Flood

REMARKS: Fire protection pipe break.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: Control Building Chiller Pumps

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC; Section 6 of System Analysis Report.

FIGURE 3-4 (continued)
(Sheet 19 of 22)

LOCATION DESIGNATOR:* FH-FZ-6

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Fire affecting chillers or chiller pumps.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD: Control Building Chillers

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC; Section 6 of System Analysis Report.

FIGURE 3-4 (continued)
(Sheet 20 of 22)

LOCATION DESIGNATOR:* FH-FZ-6

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables affected by fire, control building HVAC affected.

ANNUAL FREQUENCY OF HAZARD: 9.0-7

SYSTEM/TRAIN AFFECTED BY HAZARD:

Control Building HVAC, NS/A, NS/B, MU/A, MU/B, DH/A, NR/A, BS/A, EF/Valves, IC/All, DC/A, DR/A, RR/A

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Not important; CB-HVAC failure may be recovered by portable ventilation units or use of outside air with normal fans.

FIGURE 3-4 (continued)
(Sheet 21 of 22)

LOCATION DESIGNATOR:* FH-FZ-6

HAZARD SCENARIO: 4

HAZARD: Flood

REMARKS: Nuclear services or fire protection pipe break.

ANNUAL FREQUENCY OF HAZARD: 2.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: Control Building Chiller Pumps

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC; Section 6 of System Analysis Report.

FIGURE 3-4 (continued)
(Sheet 22 of 22)

LOCATION DESIGNATOR:* IB-FZ-1

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD:

RR-V-4A, RR-V-4B, RR-V-4C, RR-V-4D, RR-V/5, NS-V/52A, NS-V/52B, NS-V-52C, NS-V-53A, NS-V-53B, NS-V-53C

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event CF analysis; see Section 16 of System Analysis Report.

FIGURE 3-5. HAZARD SCENARIO SHEETS FOR THE
INTERMEDIATE BUILDING
(Sheet 1 of 9)

LOCATION DESIGNATOR:* IB-FZ-2

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD:

EF/TD, RR-V/4A, RR-V/4B, RR-V/4C, RR-V/4D, NS-V/52A, NS-V/52B, NS-V/52C, NS-V/53A, NS-V/53B, NS-V/53C

CORE DAMAGE: No

PLANT IMPACT:

Fan coolers lost; EF/turbine-driven; assume TT.

CORE DAMAGE FREQUENCY:

1.0-3 x [EFA/EFF(5.2-4/.1)]*[LOCA mitigating (estimate, 0.1)] = 5.2-7

FURTHER ACTIONS: Not important.

FIGURE 3-5 (continued)
(Sheet 2 of 9)

LOCATION DESIGNATOR:* IB-FZ-2

HAZARD SCENARIO: 3

HAZARD: Flood

REMARKS: Flood from EFW piping.

ANNUAL FREQUENCY OF HAZARD: 2.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: One CST

CORE DAMAGE:

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event EF analysis; see Section 11 of System Analysis Report.

FIGURE 3-5 (continued)
(Sheet 3 of 9)

LOCATION DESIGNATOR:* IB-FZ-2

HAZARD SCENARIO: 4

HAZARD: Steam

REMARKS: Break in main steam line to EFW pump; no pipe movement.

ANNUAL FREQUENCY OF HAZARD: 2.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: Main Steam, Partial

CORE DAMAGE: No

PLANT IMPACT: MS break only.

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of steam line break inside intermediate building initiating event.

FIGURE 3-5 (continued)
(Sheet 4 of 9)

LOCATION DESIGNATOR:* IB-FZ-2

HAZARD SCENARIO: 6

HAZARD: Steam

REMARKS: Break in main steam line.

ANNUAL FREQUENCY OF HAZARD: 2.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: Main Steam, RR/All

CORE DAMAGE: No

PLANT IMPACT: MS break plus fan coolers are unavailable.

CORE DAMAGE FREQUENCY: 2-5 x (MS mitigation, estimate 0.03) = 6-7.

FURTHER ACTIONS: Not important.

FIGURE 3-5 (continued)
(Sheet 5 of 9)

LOCATION DESIGNATOR:* IB-FZ-3

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD:

EF/2A, EF/2B, EF-V1A, EF-2A, EF-2B, EF-30A, EF-30B, RR-V/4A, RR-V/4B, RR-V/4C, RR-V/4D, NS-V/52A, NS-V/52B, NS-V/52C, NS-V/53A, NS-V/53B, NS-V/53C

CORE DAMAGE: No

PLANT IMPACT: Motor-driven pumps of EF lost; fan coolers lost.

CORE DAMAGE FREQUENCY.

1.0-3 x [EFF(0.1)] x HPA(2.7-3)] x [HPB(1.3-2)] = 3.5-9

FURTHER ACTIONS: Not important.

FIGURE 3-5 (continued)
(Sheet 6 of 9)

LOCATION DESIGNATOR:* IB-FZ-3

HAZARD SCENARIO: 3

HAZARD: Flood

REMARKS: Break in NS piping.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: EF/2A, EF/2B

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event EF analysis; see Section 11 of System Analysis Report.

FIGURE 3-5 (continued)
(Sheet 7 of 9)

LOCATION DESIGNATOR:* IB-FZ-6

HAZARD SCENARIO: 3

HAZARD: Steam

EMARKS: Break in main feedwater pipes.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: Main Feedwater, EF/TD

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Compares with loss of MFW (0.23 per year) x EFTD pump unavailability (0.1) = 0.023 per year; therefore, 1.0-4 per year for the same event is not important.

FIGURE 3-5 (continued)
(Sheet 8 of 9)

LOCATION DESIGNATOR:* IB-FZ-6

HAZARD SCENARIO: 4

HAZARD: Steam

REMARKS: Break in main steam pipes.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD:

Main Steam Line Break, Loss of Air Compressors

CORE DAMAGE: No

PLANT IMPACT: Loss of air compressor not important.

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of steam line break inside intermediate building initiating event.

FIGURE 3-5 (continued)
(Sheet 9 of 9)

LOCATION DESIGNATOR:* DG-FA-1

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Diesel generator fire.

ANNUAL FREQUENCY OF HAZARD: 7.4-4 per demand [(2.0-2/(2 x 12)]

SYSTEM/TRAIN AFFECTED BY HAZARD: Diesel Generator Train A

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Part of diesel generator analysis, not important.

FIGURE 3-6. HAZARD SCENARIO SHEETS FOR THE DIESEL GENERATOR BUILDING (Sheet 1 of 2)

LOCATION DESIGNATOR:* DG-FA-2

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Diesel generator fire.

ANNUAL FREQUENCY OF HAZARD: 7.4-4 per demand [(2.0-2/(2 x 12)]

SYSTEM/TRAIN AFFECTED BY HAZARD: Diesel Generator Train B

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Part of diesel generator analysis, not important.

FIGURE 3-6 (continued)
(Sheet 2 of 2)

LOCATION DESIGNATOR:* CB-FA-2a

HAZARD SCENARIO: 1a

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/A, MU/B, EF/2A, DC/A, IC/A, NS/All, RR/A, DH/A, DR/A, NR/All, 1D-SWGR, 1P-SWGR, 1R-SWGR, AH1/A, AH18/A

CORE DAMAGE: No

PLANT IMPACT: NS failure plus train A.

CORE DAMAGE FREQUENCY:

1.0-3 x [(HPA-1)(HPB)(3.0-3)] = 3.0-6 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-7. HAZARD SCENARIO SHEETS FOR THE
CONTROL BUILDING
(Sheet 1 of 25)

LOCATION DESIGNATOR:* CB-FA-2a

HAZARD SCENARIO: 3

HAZARD: Fire

REMARKS:

Cables affected by fire; IC/B and NS/C recoverable via remote shutdown system.

ANNUAL FREQUENCY OF HAZARD: 9.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/A, MU/B, EF/2A, DC/A, IC/All, NS/All, RR/A, DH/A, DR/A, NR/A, 1D-SWGR, 1P-SWGR, 1R-SWGR, 1A-SWGR, AH1/A, AH18/A, BS/All, DC/A

CORE DAMAGE: No

PLANT IMPACT:

Train A of all systems lost. Core damage may occur if MU/C fails and IC/B is not recovered.

CORE DAMAGE FREQUENCY

9.0-5 [HPC (0.013) x operator error in using alternate S/D (0.3 estimate)] = 4-7

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 2 of 25)

LOCATION DESIGNATOR:* CB-FA-2a

HAZARD SCENARIO: 4

HAZARD: Fire

REMARKS: Cables affected by fire; cabinets affected by smoke.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED [illegible] HAZARD:

DC/A, IC/All [illegible]/All, RR/A, DH/A, DR/A, NR/All, 1D, 1P-SWGR, 1R-SWGR,
AH1/A [illegible] [illegible]U/B, EF/2A, AH18/A

CORE DAMAGE: No

PLANT IMPACT: Impact the same as CB-FA-2a scenario 1a.

CORE DAMAGE FREQUENCY: 1 x 10^{-7} (see CB-FA-2a scenario 1a).

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 3 of 25)

LOCATION DESIGNATOR:* CB-FA-2b

HAZARD SCENARIO: 1a

HAZARD: Fire

REMARKS: Cables affected by fire, cabinet fire.

ANNUAL FREQUENCY OF HAZARD: 2.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/B, MU/C, ESV/B, 1E-SWGR, 1T-SWGR, 1S-SWGR, 125V/Q, VBB, VBD, EF/2B, DC/B, IC/All, NS/All, RR/All, DH/All, DR/All, CB/VAC, NR/All

CORE DAMAGE: Yes

PLANT IMPACT:

CORE DAMAGE FREQUENCY: 2.0-5

FURTHER ACTIONS: Importance to be determined.

FIGURE 3-7 (continued)
(Sheet 4 of 25)

LOCATION DESIGNATOR:* CB-FA-2b

HAZARD SCENARIO: 3

HAZARD: Fire

REMARKS: Cables affected by fire; cabinets affected by smoke.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/C, SFCC/B, ESV/B, 1E-SWGR, 1T-SWGR, 1S-SWGR, 1P-SWGR, 125V/Q, VBB, VBD, EF/2B, DC/B, NS/B, NS/C, RR/B, DH/B, DR/B

CORE DAMAGE: Yes

PLANT IMPACT: Total loss of vital power.

CORE DAMAGE FREQUENCY: 3.0-6 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 5 of 25)

LOCATION DESIGNATOR:* CB-FA-2c

HAZARD SCENARIO: 1a

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.5 x 10^{-5}

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/All, 1T-SWGR, RR/B, DH/B, DR/B, NR/C, RCP Monitor Racks, DC/1M, AH1/B, 1C Transfer Switch, BS/All, IC/All

CORE DAMAGE: No

PLANT IMPACT:

RCP seal failure because IC + MU failure; train B of several systems.

CORE DAMAGE FREQUENCY:

1.5 x 10-5 x [HPB (7-3) + other train A (0.1 estimate)] = 2-6

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 6 of 25)

LOCATION DESIGNATOR:* CB-FA-2c

HAZARD SCENARIO: 6

HAZARD: Fire

REMARKS: Cables affected by fire; cabinets affected by smoke.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/All, 1T-SWGR, RR/B, DH/B, DR/B, NR/C, RCP Racks, DC/1M, AH1/B, 1C Transfer Switch, BS/B, IC/All, INV/B, INV/D

CORE DAMAGE: No

PLANT IMPACT: RCP seal failure because of IC and MU; train B of systems.

CORE DAMAGE FREQUENCY:

1.0-5 x [EFF(0.1)] [LOC + mitigation estimate (0.1) = 1.0-7

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 7 of 25)

LOCATION DESIGNATOR:* CB-FA-2d

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 5.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

NS/All, NR/A, NR/C, DC/All, MU/All, IC/All, INV/A, INV/C, INV/E, VBA, VBC, CHG/A, CHG/C, CHG/E, DC Pan/A, DG Pan/P, AH1/A, AH1/B, AH18/All

CORE DAMAGE: Yes

PLANT IMPACT:

DC train A, CB-HVAC, RCP seal failure, NS/All, no LOCA mitigation (recoverable).

CORE DAMAGE FREQUENCY: 5.0-6

FURTHER ACTIONS: Importance to be determined.

FIGURE 3-7 (continued)
(Sheet 8 of 25)

LOCATION DESIGNATOR:* CB-FA-2d

HAZARD SCENARIO: 3

HAZARD: Fire

REMARKS: Cables affected by fire; cabinets affected by smoke.

ANNUAL FREQUENCY OF HAZARD: 2.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

NS/All, NR/A, NR/C, DC/A, DC/B, MU/All, IC/A, IC/B, INV/A, INV/B, INV/C, INV/D, INV/E, VBA, VBC, CHG/A, CHG/C, CHG/E, DC Pan/A, DG Pan/P, AH1/A, AH1/B

CORE DAMAGE: Yes

PLANT IMPACT:

All DC, CB-HVAC, RCP seal failure, LOCA mitigation (recoverable).

CORE DAMAGE FREQUENCY: 2.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 9 of 25)

LOCATION DESIGNATOR:* CB-FA-2d

HAZARD SCENARIO: 4

HAZARD: Fire

REMARKS: Cables affected by fire; cabinets affected by smoke.

ANNUAL FREQUENCY OF HAZARD: 2.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

1P-SWGR, NS/All, NR/A, NR/C, DC/A, DC/B, MU/All, IC/All, INV/A, INV/C, INV/E, VBA, VBC, CHG/A, CHG/C, CHG/E, DC Pan/A, DG Pan P, AH1, AH18

CORE DAMAGE: Yes

PLANT IMPACT:

DC train A, 4.11 kV train A, CB-HVAC, RCP seal, LOCA mitigation (recoverable).

CORE DAMAGE FREQUENCY: 2.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 10 of 25)

LOCATION DESIGNATOR:* CB-FA-2e

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 2.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

INV/B, INV/D, MU/A, MU/B, DC Pan/B, DG Pan/Q, CHG/C, CHG/D, CHG/F, VBB, VBD, NR/All, EP/B, ESV/C

CORE DAMAGE: No

PLANT IMPACT: CB-HVAC and NR are lost; train B of all systems.

CORE DAMAGE FREQUENCY:

2.0-5 x [train A failure (0.3 estimate) and operator fails to recover NR (0.3 estimate)] = 2.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 11 of 25)

LOCATION DESIGNATOR:* C3-FA-2f

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 6.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

DC/B, IC/All, Bat Rack/A, Bat Rack/C, 125V/B, AH18/B, AH1/B, MU/A, MU/C, DH/B, BS/B, NS/C, NR/C, DR/B, ESV/B, ESSH/B

CORE DAMAGE: No

PLANT IMPACT: Train B of system and train A of batteries.

CORE DAMAGE FREQUENCY: 6.0-6 x [SEC(0.18)] = 1.1-6

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 12 of 25)

LOCATION DESIGNATOR:* CB-FA-2g

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD:

Bat Rack/B, Bat Rack/D, 125V/B, Some Instrument Cables, EFV-30B, EFV-30D

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of DC bus B analysis (DB); see Section 2 of System Analysis Report.

FIGURE 3-7 (continued)
(Sheet 13 of 25)

LOCATION DESIGNATOR:* CB-FA-3a

HAZARD SCENARIO: 2

HAZARD: Fire

REMARKS: Cables and bus bars affected by large fire.

ANNUAL FREQUENCY OF HAZARD: 1.5-4

SYSTEM/TRAIN AFFECTED BY HAZARD:

1D-SWGR, 1P-SWGR, 1R-SWGR, ESV/A, OP Bus to 1D-SWGR, MU/A, EF/2A, RR/A, DR/A, DH/A

CORE DAMAGE: No

PLANT IMPACT: LOOP plus train A.

CORE DAMAGE FREQUENCY: 1.5-4 x [GB(.08)] = 2.4-5

FURTHER ACTIONS:

Important; recovery of offsite power to one source is not possible.

FIGURE 3-7 (continued)
(Sheet 14 of 25)

LOCATION DESIGNATOR:* CB-FA-3a

HAZARD SCENARIO: 3

HAZARD: Fire

REMARKS: Cables affected by fire; cabinets affected by smoke.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

1D-SWGR, 1P-SWGR, 1R-SWGR, ESV/A, OP Bus to 1D-SWGR, 1E-SWGR, MU/A, MU/B, EF/2A, RR/A, DR/A, DH/A

CORE DAMAGE: Yes

PLANT IMPACT:

LOOP and both 4.16-kV AC buses.

CORE DAMAGE FREQUENCY: 3.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 15 of 25)

LOCATION DESIGNATOR:* CB-FA-3b

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

1E-SWGR, 1S-SWGR, 1T-SWGR, ESV/B, MU/All, EF/2B, NS/C, RR/B, DH/B, IC/All, DC/B, DR/B

CORE DAMAGE: Yes

PLANT IMPACT:

RCP seal failure because MU and IC are lost.

CORE DAMAGE FREQUENCY: 1.0-5

FURTHER ACTIONS: Importance to be determined.

FIGURE 3-7 (continued)
(Sheet 16 of 25)

LOCATION DESIGNATOR:* CB-FA-3b

HAZARD SCENARIO: 3

HAZARD: Fire

REMARKS: Cables affected by fire; cabinets affected by smoke.

ANNUAL FREQUENCY OF HAZARD: 5.0-7

SYSTEM/TRAIN AFFECTED BY HAZARD:

1E-SWGR, 1D-SWGR, 1S-SWGR, 1T-SWGR, ESV/B, MU/All, EF/2A, EF/3B, NS/All, RR/B, DH/A, DH/B

CORE DAMAGE: Yes

PLANT IMPACT: Total loss of vital power.

CORE DAMAGE FREQUENCY: 5.0-7 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 17 of 25)

LOCATION DESIGNATOR:* CB-FA-3c

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS:

Cables affected by fire; impact can be mitigated if alternate shutdown system is used.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/All, EP/All, Instrumentation, ESAS, AH/All, DH/All, BS/All, IC/All, DC/All, RR/All, Condenser Steam Dump

CORE DAMAGE: Yes

PLANT IMPACT:

All LOCA mitigation systems are lost; fan coolers + BS lost; RCP seal failure because of IC and MU; all electric power lost (recoverable).

CORE DAMAGE FREQUENCY:

1.0-4 x [operator error in using alternate shutdown system (0.2)]
= 2.0-5

FURTHER ACTIONS:

Importance to be determined. Detailed operator action is needed.

FIGURE 3-7 (continued)
(Sheet 18 of 25)

LOCATION DESIGNATOR:* CB-FA-3d (cable spreading room)

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS:

Fire at center of rom affects several control cables. Operators can use alternate shutdown system for recovery.

ANNUAL FREQUENCY OF HAZARD: 2.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

Large number of systems, including reactor building functions.

CORE DAMAGE: Yes

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 19 of 25)

LOCATION DESIGNATOR:* CB-FA-4b (control room)

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS:

Fire in control panels CC and CR. Operators use alternate shutdown system.

ANNUAL FREQUENCY OF HAZARD: 3.0-6 (includes human error)

SYSTEM/TRAIN AFFECTED BY HAZARD:

The control circuits of large number of vital systems affected.

CORE DAMAGE: Yes

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 20 of 25)

LOCATION DESIGNATOR:* CB-FA-5a

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Fire affecting HVAC fans, damper motors, and cables.

ANNUAL FREQUENCY OF HAZARD: 3.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD:

AH-E-19A, AH-E-19B, AH-E-18A, AH-E-17A, several dampers

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC, Section 6 of System Analysis Report.

FIGURE 3-7 (continued)
(Sheet 21 of 25)

LOCATION DESIGNATOR:* CB-FA-5b

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Fire affecting HVAC fans, damper motors, and cables.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD: AH-E-18B, AH-E-17B, AH-D-41B, AH-D-32B

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC; not important when compared with CB-FA-5a, scenario 1.

FIGURE 3-7 (continued)
(Sheet 22 of 25)

LOCATION DESIGNATOR:* CB-FA-1

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 3.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU/All, EF/2A, EF/2B, IC/A, IC/B, NS/All, RR/A, DR/A, NR/A, 1D-SWGR, 1E-SWGR

CORE DAMAGE: Yes

PLANT IMPACT: Large number of components.

CORE DAMAGE FREQUENCY: 3.0-6

FURTHER ACTIONS: Not important.

FIGURE 3-7 (continued)
(Sheet 23 of 25)

LOCATION DESIGNATOR:* CB-FA-1

HAZARD SCENARIO: 6

HAZARD: Flood

REMARKS: Large flood from lab and housekeeping activities.

ANNUAL FREQUENCY OF HAZARD: 1.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: Control Building Chiller Pumps

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC, Section 6 of System Analysis Report.

FIGURE 3-7 (continued)
(Sheet 24 of 25)

LOCATION DESIGNATOR:* CB Stairs

HAZARD SCENARIO: 1

HAZARD: Flood

REMARKS: Flood from fire protection and HVAC pipes.

ANNUAL FREQUENCY OF HAZARD: 1.0-4

SYSTEM/TRAIN AFFECTED BY HAZARD: Control Building Chiller Pumps

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of CB-HVAC, Section 6 of the Systems Analysis Report.

FIGURE 3-7 (continued)
(Sheet 25 of 25)

LOCATION DESIGNATOR:* RB-FZ-1a

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD: AH1/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event CF analysis; see Section 16 of System Analysis Report.

FIGURE 3-8. HAZARD SCENARIO SHEETS FOR THE
REACTOR BUILDING
(Sheet 1 of 13)

LOCATION DESIGNATOR:* RB-FZ-1a

HAZARD SCENARIO: 7

HAZARD: Flood

REMARKS: Break in main feedwater pipes.

ANNUAL FREQUENCY OF HAZARD: 8.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: AH1/All, Main Feedwater

CORE DAMAGE: No

PLANT IMPACT: Fan cooler plus MF initiating event.

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Not important; compares with loss of MF (0.23/year) x [CFI(0.0066)] = 1.2-3/year.

FIGURE 3-8 (continued)
(Sheet 2 of 13)

LOCATION DESIGNATOR:* RB-FZ-1c

HAZARD SCENARIO: 8

HAZARD: Flood

REMARKS: Break in MUPS piping.

ANNUAL FREQUENCY OF HAZARD: 2.0-5

SYSTEM/TRAIN AFFECTED BY HAZARD: MU/All, RB/A

CORE DAMAGE: No

PLANT IMPACT: MUPs plus RB train A.

CORE DAMAGE FREQUENCY: 2-5 x (mitigation, estimate, 0.03) = 6-7

FURTHER ACTIONS: Not important.

FIGURE 3-8 (continued)
(Sheet 3 of 13)

LOCATION DESIGNATOR:* RB-FZ-1c

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD: Part of Instrumentation, AH1/A11

CORE DAMAGE: No

PLANT IMPACT:

Fan coolers lost; assume reactor trip; operators can recover.

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event CF analysis; see Section 16 of System Analysis Report.

FIGURE 3-8 (continued)
(Sheet 4 of 13)

LOCATION DESIGNATOR:* RB-FZ-1b

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD:

Part of Instrumentation

CORE DAMAGE: No

PLANT IMPACT: Partial loss of instrumentation.

CORE DAMAGE FREQUENCY:

The operators can recover from this event because (1) some instrumentation will be left unaffected, (2) all vital and balance of plant equipment will remain unaffected by the fire, and (3) loss of instrumentation channels by themselves cannot lead to adverse situations.

FURTHER ACTIONS: Not important.

FIGURE 3-8 (continued)
(Sheet 5 of 13)

LOCATION DESIGNATOR:* RB-FZ-1e

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-3

SYSTEM/TRAIN AFFECTED BY HAZARD:

Part of Instrumentation, RCP Oil Cooling

CORE DAMAGE: No

PLANT IMPACT: Partial loss of instrumentation.

CORE DAMAGE FREQUENCY:

The operators can recover from this event because (1) some instrumentation will be left unaffected, (2) all vital and balance of plant equipment will remain unaffected by the fire, and (3) loss of the instrumentation channels by themselves cannot lead to adverse situations.

FURTHER ACTIONS: Not important.

FIGURE 3-8 (continued)
(Sheet 6 of 13)

LOCATION DESIGNATOR:* RB-FZ-1b

HAZARD SCENARIO: 7

HAZARD: Flood

REMARKS: Break in MUPS piping.

ANNUAL FREQUENCY OF HAZARD: 8.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD: MUPS Pipe Break

CORE DAMAGE:

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of Top Event HPA/HPB analysis; see Section 13 of System Analysis Report.

FIGURE 3-8 (continued)
(Sheet 7 of 13)

LOCATION DESIGNATOR:* RB-FZ-1d

HAZARD SCENARIO: 1

HAZARD: Fire

REMARKS: Cables affected by fire.

ANNUAL FREQUENCY OF HAZARD: 1.0-2

SYSTEM/TRAIN AFFECTED BY HAZARD:

Part of Instrumentation, PORV, Block Valves, Pressurizer Spray, RCP Oil Cooling

CORE DAMAGE: No

PLANT IMPACT:

Impact on instrumentation not important; PO-1 = 1.0, CD-1 = 1.0

CORE DAMAGE FREQUENCY:

1.0-2 x [MFA(0.016) + MFG(0.081)] x [EF-1(4.6-4)
+ SD-1(1.5-5)] = 4.6-7 per Year

FURTHER ACTIONS: Not important.

FIGURE 3-8 (continued)
(Sheet 8 of 13)

LOCATION DESIGNATOR:* RB-FZ-1d

HAZARD SCENARIO: 8

HAZARD: Flood

REMARKS: Break in main feedwater pipes.

ANNUAL FREQUENCY OF HAZARD: 8.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

MU to two RCPS, EF to one OTSG, Main Feed to Loop A, PORV-Related Cables.

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Not important, other OTSG remains unaffected, and reactor pressure can be controlled using heat removal capability from that OTSG, or through the safety valves.

FIGURE 3-8 (continued)
(Sheet 9 of 13)

LOCATION DESIGNATOR:* RB-FZ-1d

HAZARD SCENARIO: 11

HAZARD: Steam

REMARKS: Break in main steam line.

ANNUAL FREQUENCY OF HAZARD: 8.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

Main Steam Break, partial loss of MU, and EF, Main Feed to Loop A, and RCP Seal Piping

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of steam line pipe break inside intermediate building initiating event.

FIGURE 3-8 (continued)
(Sheet 10 of 13)

LOCATION DESIGNATOR:* RB-FZ-1e

HAZARD SCENARIO: 8

HAZARD: Flood

REMARKS: Break in main feedwater pipes.

ANNUAL FREQUENCY OF HAZARD: 8.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

Half of MU, Half of EF, and Main Feed to Loop B

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Not important, the other OTSG remains unaffected and can remove heat. Aso, PORVs remain unaffected. Many cooldown paths remain unaffected.

FIGURE 3-8 (continued)
(Sheet 11 of 13)

LOCATION DESIGNATOR:* RB-FZ-1e

HAZARD SCENARIO: 11

HAZARD: Steam

REMARKS: Break in main steam line.

ANNUAL FREQUENCY OF HAZARD: 8.0-6

SYSTEM/TRAIN AFFECTED BY HAZARD:

Main Steam Break, Partial Loss of MU and EF, Main Feed to Loop B, and RCP Seal Piping

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of steam line pipe break inside intermediate building initiating event.

FIGURE 3-8 (continued)
(Sheet 12 of 13)

LOCATION DESIGNATOR:* RB-FZ-2

HAZARD SCENARIO: 11

HAZARD: Steam

REMARKS: Break in main steam line.

ANNUAL FREQUENCY OF HAZARD: 2.0-5

FREQUENCY EVALUATION:

SYSTEM/TRAIN AFFECTED BY HAZARD:

Main Steam Line Break, Partial FW and EF, CF/A, and Instrumentation, IC/All

CORE DAMAGE: No

PLANT IMPACT:

CORE DAMAGE FREQUENCY:

FURTHER ACTIONS:

Part of steam line break inside intermediate building initiating event.

FIGURE 3-8 (continued)
(Sheet 13 of 13)

4. EXTERNAL FLOODING ANALYSIS

Potential accident sequences initiated by external floods at Three Mile Island Unit 1 are investigated in this section, and the frequency of the resulting core damage scenarios are calculated.

4.1 FLOODING FREQUENCY

The TMI-1 plant is located on the northern part of the Three Mile Island River, approximately 2.5 miles south of Middletown, Pennsylvania. The drainage area of the river at Harrisburg gauging station, located approximately 11 river miles upstream from the site, is 24,100 square miles (Reference 4-1). The island is elongated parallel to the flow of the river and is about 11,000 feet in length and 1,700 feet in width. The width of the river north of the island is approximately 1.5 miles (Figure 4-1). There are no large dams or reservoirs immediately upstream from the site. The Susquehanna River is the prinicipal source of flood in the Harrisburg area. The flood history dating back to 1786 indicates that the highest flood of record prior to 1972 occurred on March 19, 1936, and is believed to have been the highest known flood since 1784 and, probably, since 1740 (Reference 4-1). The 1936 flood discharge rate at Harrisburg was 740,000 cfs and resulted from a large-scale snow melt over the entire area of Pennsylvania. It is believed that the most likely source of large floods of magnitude equal to or greater than the 1936 flood would be the result of one or a combination of such conditions as large-scale snow melt and antecedent rainfall and cyclones. However, in June 1972, rainfall due to the tropical storm Agnes caused a flood that reached the 300-foot elevation at the south end of the site. The 1972 flood therefore is the highest flood of record to date.

According to flood analysis reported in Reference 4-1, the discharge rate corresponding to the so-called probable maximum flood is 1,625,000 cfs. This, at the site, corresponds to a calculated surface elevation of 310 feet at the tip of Three Mile Island.

Figure 4-2 presents the annual frequency of exceeding various flood levels at the TMI-1 site. The data points on the curve represent the following major floods:

Flood Year	Elevation (feet)	Annual Frequency
1964	292	4.5×10^{-2}
1936	298	5.0×10^{-3}
1972	300	2.5×10^{-3}

The frequency of the 1936 flood was calculated based on the observation period, 1786 through 1986. In the case of the 1972 Agnes flood, a return period of 400 years was assumed, based on the assessment of a study performed by the Corps of Engineers, as reported in Reference 4-2. According to that assessment, the recurrence interval of the 1972 flood is between 400 and 500 years.

To obtain the frequency of exceedance for probable maximum flood, the curve in Figure 4-2 was extrapolated linearly, resulting in an annual frequency of 1.0×10^{-5}. It can be argued that linear extrapolation is conservative since higher level floods require extreme conditions that are increasingly unlikely to be met. The early portions of the graph also indicate a nonlinear behavior. Nevertheless, in light of significant modeling and data uncertainties, a conservative assessment is prudent. Therefore, for the purpose of this analysis, 1.0×10^{-5} is taken as the mean value of the uncertainty distribution of PMF frequency. Using a lognormal model with a range factor of 5 puts the lower bound (5th percentile) estimate of the frequency at about 1.2×10^{-6}, which is about the same as the prediction of Reference 4-2 for the site. The upper bound (95th percentile) in this case is 3.1×10^{-5} per year.

Elevation 310' is particularly important because, as discussed in the following sections, flood protection is provided for the plant against flood levels up to 310 feet.

As will be discussed in the following sections, there are several core damage scenarios that can be initiated by floods between Elevation 305' and Elevation 310'. The mean annual frequency of such floods, from Figure 4-2, is 1.5×10^{-4}. Using a lognormal model with a range factor of 5 to represent uncertainties, we obtain

95th percentile = 4.7×10^{-4} per year

5th percentile = 1.9×10^{-5} per year

4.2 FLOOD DESIGN CONSIDERATIONS

TMI-1 plant grade is Elevation 304' to protect the plant from floods up to the design flood (a discharge rate of 1,100,000 cfs and water Elevation 304' at the tip of the island and Elevation 303' at the southern side of the plant). A protective dike has been constructed at Elevation 310' at the tip of the island and descends uniformly along both sides of the island to Elevation 305'. The dike does not protect the site from PMF. However, flood gates are provided for protecting the safety-related structures and components with minimum top Elevation 311' except for the air intake tower for which the lowest opening is at an Elevation 310'. The following is a summary of protective measures for various critical buildings and equipment according to TMI-1 Flood Emergency Procedure 1202-32.

- Intake Screen and Pump House
    - Stop logs.
    - Seals where pump shafts penetrate the floor slab.

- Manholes in slab at Elevation 308' sealed.
- Electrical equipment located in elevation above PMF level.

- Fuel Storage Building

  - Stop logs.

  - Watertight door to tendon gallery.

  - Inflatable rubber seal around railroad door.

  - The 3-inch gap between this and the reactor building made watertight.

- Control Building. Stop logs.

- Auxiliary Building

  - Watertight door to tendon gallery.
  - Inflatable rubber seal around truck unloading door.
  - Pipe and conduit penetrations made watertight.

- Intermediate Building

  - Stop logs.
  - Drain system designed for flood condition.

- Diesel Generator Building

  - Stop logs.
  - Tornado panels made watertight.

- Air Ventilation Inlet. Located at an elevation above flood level.

- Diesel Fuel Oil Storage Tank

  - Top of the right wall located at Elevation 312' and designed to withstand hydraulic forces.

  - Tank foundation designed for full uplift (tank empty).

Also, all openings for ducts, pipes, conduits, cable trays, etc., are sealed. The effectiveness of these measures for preventing core damage is discussed in the following sections.

4.3 FLOOD-INITIATED SCENARIOS

Flood scenarios are analyzed in three categories. These categories, determined by the flood elevation, are: (1) floods more than Elevation 310' in which the critical structures are flooded even if all the protective measures summarized in Section 4.2 are taken, (2) floods between Elevation 305' (dike overflow) and Elevation 310', and (3) floods less than Elevation 305' in which the site will not be impacted unless the dike is failed. A key element in developing flood scenarios is the plant flood response procedure. The key actions called for by the flood protection procedures are (Reference 4-1):

1. Initiation of the Flood Protection Procedure. Described in Emergency Procedure 1202-32, with a 36-hour forecast of 350,000 cfs or more discharge rate.

2. Initiation of Flood Alert. Initiated by the operations and maintenance director, with a 36-hour forecast of 640,000 cfs or more discharge.

3. Emergency Closure. Called by the operations and maintenance director, with a 36-hour forecast of 940,000 cfs or more discharge rate.

4. Shutdown Alert. Ordered by the operations and maintenance director if the water elevation at the Unit 1 river water intake structure reaches 301 feet (950,000 cfs).

5. Shutdown. Ordered by the operations and maintenance director if the river stage reaches 302 feet (1,000,000 cfs).

In response to an "emergency closure" order, the operators perform the steps indicated in Table 4-1. The impact of omitting any one of these steps is also identified in Table 4-1. Of course, there is no impact if the flood level does not exceed the plant elevation of 305 feet.

Table 4-2 identifies the potential flood impacts on the plant equipment needed to maintain cold shutdown, the equipment needed to achieve a slow cooldown to cold shutdown without offsite power, and the equipment needed to perform a normal cooldown to cold shutdown conditions, with offsite power available.

Successful implementation of the flood procedure steps depends on the availability of sufficient time. The initiation of flood procedure and the subsequent actions are mainly determined by the observed flood level. Hydrographs of Figure 4-3 show the available time in various historical floods and the probable maximum flood.

According to the PMF hydrograph in Figure 4-3, the time from shutdown order until water reaches Elevation 310' at TMI (corresponding to a flood discharge rate of 1,625,000 cfs) is about 27 hours. In a fast-developing flood, such as flood caused by hurricane, the length of time may be shorter. For example, extrapolation of the hydrograph for the Agnes flood from Elevation 302' to Elevation 310' gives a time interval of about 10 hours. This, however, is unrealistic since a hurricane is unlikely to produce the PMF at the site. The various flood-initiated core damage scenarios are described and quantified in the following.

4.3.1 FLOODS WITH AN ELEVATION GREATER THAN 310 FEET

As discussed earlier, the critical plant equipment required for cold shutdown is only protected up to Elevation 310'. In the event of a flood that exceeds 310 feet, it is expected that the operators would take actions not currently covered by procedure to protect the plant. These actions would likely be initiated as soon as plant personnel realize that the flood level might exceed the design basis flood level. In particular, it is expected that additional actions would be initiated no later than when the flood levels exceed 305 feet and the operators realize that their previous actions have been instrumental in protecting the plant. By this time, the plant staff will be highly motivated to take further actions to protect the plant and themselves. Even in the event of a hurricane, at least 5 additional hours should be available before flood levels exceed Elevation 310'. From the hydrographs, as may as 25 hours may be available if the flood is not caused by a hurricane.

It is anticipated that the operators could protect the plant to even higher elevations by stacking sandbags and installing additional metal covers on the openings that would otherwise be exposed. Piles of sand are specifically available onsite for this purpose, and there is a substantial amount of spare metal available that could be used. However, the number of openings that must be covered up to, say, 312 feet is substantial; i.e., more than 10. For example, there are at least four such openings for the river water pumphouse. Consequently, the likelihood of success is judged to not be very great. A uniform distribution is assumed for the error rate for this nonproceduralized action, with a mean value of 0.5. The mean annual frequency of core damage due to floods of more than 310 feet is then calculated as the frequency of flood (ϕ_{F1}) and the conditional probability of core damage (p_{CD}):

$$\phi_{CD1} = \phi_{F1} * P_{CD}$$

$$= (1.0 \times 10^{-5})(0.5) = 5.0 \times 10^{-6} \text{ per year}$$

4.3.2 FLOODS WITH ELEVATIONS BETWEEN 305 FEET AND 310 FEET

At Elevation 305', or more, the dike will overflow and the critical equipment at the plant will be exposed to flood if the protective measures described earlier are not taken by the plant personnel. An event tree, Figure 4-4, was developed to describe and quantify the various core damage scenarios initiated by such a flood. The top events of this tree and the associated split fractions are described in the following:

- Top Event EW. This top event represents success of early warning (EW). This top event is particularly important in view of the fact that successful implementation of flood procedure depends on the availability of sufficient time to perform the necessary actions. The first operator action in the model for external flooding questions whether the operators have recognized that a flood watch is required and have therefore initiated monitoring of the river level. This monitoring is important because the procedural guidance for ordering a plant shutdown is keyed to when the elevation of the river reaches 302 feet and no automatic alarms are available in the control room. The error rate for this action is computed using the methods for dynamic human actions, nonresponse errors, described in Section 2 of the Human Actions Analysis Report. Two scenarios are evaluated: heavy rainfall and floods due to hurricanes, which are assumed to also cause failure of offsite power. The time from the first indications of severe flooding until Elevation 302' is reached depends on the cause of the flood. For our purposes, two different causes are modeled (i.e., heavy rainfall versus hurricanes), corresponding to the hydrographs in Figure 4-3. The human action analysis input for the early warning human actions is provided in Table 4-3 along with the resulting error rates. In the quantification of the event tree of Figure 4-4, the higher of the two numbers (i.e., HEWIC) is used. The mean value, based on a lognormal distribution fit to the lower and upper bounds as the 5th and 95th percentiles is 3.84×10^{-4}.

- Top Event OP. This top event represents the possibility of losing offsite power (OP) in a flood between Elevation 305' and Elevation 310'. Offsite power is lost if the water level exceeds 307 feet due to flooding of transformers and other equipment in the switchyard. From Figure 4-2, it can be seen that the frequency of floods higher than Elevation 307' is about one-third of the frequency of floods in the range between Elevation 305' and Elevation 310'. Also, of the three major floods at the site, only one has been due to hurricane in which there is a high chance of offsite power being lost. Therefore, in this analysis, the likelihood of loss of offiste power, given a flood in the range between Elevation 305' and Elevation 310', is assessed to be 0.33.

- Top Event EP. This top event represents the availability of onsite emergency power (EP) for a period of 24 hours if offsite power becomes unavailable. Probability of failure of both emergency power trains from causes other than flooding was calculated from the emergency power system model (Systems Analysis Report), assuming a 24-hour mission time. The mean value was calculated to be 1.55×10^{-2}. (This number was produced using the equations in the Systems Analysis Report, Section 2, for a 24-hour mission time instead of the 6 hours used in GAC and GBD.)

- Top Event CS. This top event represents the successful and timely response of operators to initiate shutdown and bring the plant to cold shutdown (CS) condition. Note that the success (no core damage) paths require success in both Top Events CS and SL described below.

  Two conditional split fractions for Top Event CS are estimated, depending on whether offsite power is available or not. Table 4-4 represents the input to the human actions analysis process for these two cases. Note that, in the second case, it is assumed that the cause of flooding is a hurricane in which the available time before the flood peaks is less than for other flooding conditions (heavy rainfall or large-scale snow melt). The resulting number, therefore, is a conservative representation of various possible floods. The mean frequency for the two cases are:

  HCD6A (offsite power available) = 9.32×10^{-3}

  HCD6C (offsite power failed, hurricane condition) = 0.857

- Top Event SL. This top event refers to the successful implementation of a series of steps listed in the emergency procedure under "emergency closure" category (see Tables 4-1 and 4-2).

  For floods in the range of interest, eight of the steps identified in Table 4-1, if omitted, would cause failure of the equipment needed to maintain cold shutdown; i.e., steps 3.5.1-D-4, D-1, D-3, D-2, B-1, B-2, A-2, and A-3. If the plant is not already at cold shutdown before the flood peaks, omission of step 3.5.1-C-1 would be likely to prevent attaining DHR entry conditions.

  Sufficient time is assumed available for the performance of all the 3.5.1 step so that the actions required in the implementation of Emergency Procedure 1202-32 can be considered routine rather than dynamic. The frequency for errors of omission when a procedure is used and a long list of involved instructional items are presented in Table 2-2 of the Human Actions Analysis Report. The basic human error rate HE01B (i.e., see also Table 1-3 from the Human Actions Analysis Report) is per item, so it is therefore multiplied by 8 if the plant is already in cold shutdown at the time the flood peaks (i.e., HSL1) or by 9 if the plant is still on its way to cold shutdown; i.e., HSL2. The results for these two actions are indicated in the following table.

ERROR RATE FOR FAILING TO IMPLEMENT ALL STEPS
REQUIRED IN AN EMERGENCY CLOSURE

Split Fraction	Mean	5th Percentile	50th Percentile	95th Percentile
HSL1	5.62-2	2.63-3	2.34-2	.20
HSL2	6.33-2	2.96-3	7.63-2	.224

NOTE: Exponential notation is indicated in abbreviated form; i.e., 5.62-2 = 5.62 x 10^{-2}.

  If a step in the procedure is omitted, there is still a good chance that other plant operations personnel will discover the omission in time to perform corrective action. Initially, it is expected that plant personnel will be stationed at various key locations throughout the plant, specifically to look for leaks. Also, once the flood levels have reached 305 feet, some flooding is expected. Building sump level alarms would actuate, causing the control room crew to investigate. In light of the potential for discovering such omissions in time to recover, a factor to account for such actions is applied to both actions discussed previously. Medium dependence on the original action is assumed; i.e., the error rates are multiplied by HEMD (see Human Action Analysis Report), which has a mean error rate of about 0.19 to account for the potential recovery.

Using the top event split fractions estimated above, and the frequency, ϕ_{F2}, of floods in the range between Elevation 305' and Elevation 310', the various core damage sequences identified in the event tree of Figure 4-4 can be quantified. In particular, we have

$$\phi_{2A} = \phi_{F2} * (1-\overline{OP}) * HSL1 * HEMD$$

$$= (1.5 \times 10^{-4})(1-0.33)(5.62 \times 10^{-2})(0.19) = 1.07 \times 10^{-6}$$

$$\phi_{2B} = \phi_{F2} * (1-\overline{OP}) * HCD6A * HSL2 * HEMD$$

$$= (1.5 \times 10^{-4})(1-0.33)(9.32 \times 10^{-3})(6.33 \times 10^{-2})(0.19)$$

$$= 1.12 \times 10^{-8}$$

$$\phi_{2C} = \phi_{F2} * \overline{OP} * (1-HCD6C) * HSL2 * HEMD$$

$$= (1.5 \times 10^{-4})(0.33)(1-0.857)(6.33 \times 10^{-2})(0.19)$$

$$= 8.51 \times 10^{-8}$$

$$\phi_{2D} = \phi_{F2} * \overline{OP} * HCD6C * HSL2 * HEMD$$

$$= (1.5 \times 10^{-4})(0.33)(0.857)(6.33 \times 10^{-2})(0.19)$$

$$= 5.10 \times 10^{-7}$$

$$\phi_{2E} = \phi_{F2} * \overline{OP} * \overline{EP}$$

$$= (1.5 \times 10^{-4})(0.33)(1.55 \times 10^{-2}) = 7.67 \times 10^{-7}$$

$$\phi_{2F} = \phi_{F2} * HEWIC$$

$$= (1.5 \times 10^{-4})(3.84 \times 10^{-4}) = 5.76 \times 10^{-8}$$

The total mean core damage frequency for scenarios initiated by floods in the range between Elevation 305' and Elevation 310' is the sum of the above frequencies; i.e.,

$$\phi_{CD2} = 2.50 \times 10^{-6}/\text{year}$$

The uncertainty distribution of ϕ_{CD2} is developed from the distribution of its components, using the Monte Carlo technique.

4.3.3 FLOODS WITH AN ELEVATION LESS THAN 305 FEET

Floods with elevations less than 300 feet have no impact on the site. Flooding of the site in floods in the range between Elevation 300' and Elevation 305' is prevented by the protective dike. However, the site would be flooded in such floods if the dike fails. The annual frequency of floods in the range between Elevation 300' and Elevation 305' is about 2.5×10^{-3} per year (Figure 4-2), and the probability of dike failure is less than 10^{-3}. This number is based on the probability of failure of earth dams (Reference 4-3). Consequently, the frequency of flooding of the site is about 2.5×10^{-6} per year.

The consequences of a flood in this category are less severe than the category described in the previous section because the openings to most critical buildings are at about Elevation 305'. Therefore, the conditional probability of core melt, given this type of flood, should be smaller than the conditional probabilities assessed in the previous section.

Consequently, the annual frequency of core damage scenarios due to this category of floods would be more than two orders of magnitude smaller than the frequencies in the previous category. The total contribution of scenarios in this category is therefore conservatively estimated to be

$$\phi_{CD3} = (2.5 \times 10^{-6})(10^{-2}) = 2.5 \times 10^{-8} \text{ per year}$$

The above value was used as the mean of a lognormal distribution, with a range factor of 10, to represent modeling and data uncertanties.

4.3.4 TOTAL CORE DAMAGE FREQUENCY

The annual frequency of core damage due to external flooding is calculated from

$$\phi_{CD} = \phi_{CD1} + \phi_{CD2} + \phi_{CD3}$$

The distribution of the various terms in the right-hand side of the above equation were used to generate the uncertainty distribution of ϕ_{CD}. The main characteristics of the resulting distribution are

5th Percentile = 7.6×10^{-7}

50th Percentile = 4.0×10^{-6}

95th Percentile = 1.8×10^{-5}

Mean = 7.5×10^{-6}

4.4 REFERENCES

4-1 GPU Nuclear Corporation, TMI-1 FSAR, Update 4, July 1985.

4-2 GPU Nuclear Corporation, TMI-2 FSAR, Amendment 14, 1974.

4-3 Goubet, A., "Risques Associes aux Barrages," La Houille Blanche/N° 8-1979.

TABLE 4-1. IMPACT OF OMITTING STEP IN FLOOD PROCEDURE 1202-32

Sheet 1 of 4

Step	Procedure	Location	Number	Impact if Omitted
3.5.1	Install flood panels.			
D-4		Diesel Generator Building Air Intake Openings	2	Lose both diesel generators if flood > 305 feet.
D-1		North Entrance to Diesel Generator Building	1	Lose both diesel generators if flood > 305 feet.
D-3		East Entrance to Diesel Generator Building	1	Lose both diesel generators if flood > 305 feet.
D-2		West Entrance to Diesel Generator Building	2	Lose both diesel generators if flood > 305 feet.
E-1		South Entrance to Intake, Screen, and Pumphouse Switchgear and Pumproom	1	Lose all river water systems if flood > 311 feet. (Pump motors are above the 308-foot floor level.)
E-2		Doorways between Screen Rooms and Pumphouse Rooms in Intake, Screen, and Pumphouse	3	Lose all river water systems if flood > 311 feet. (Pump motors are above the 308-foot floor level.)
E-3		Doorway to Diesel-Driven Fire Pumproom Adjacent to the Intake, Screen, and Pumphouse	1	Lose all river water systems if flood > 311 feet. (18-inch connecting pipe to pumphouse).
E-4		Nine foot Wide Doorway between Screen and Pump Rooms in Intake, Screen, and Pumphouse	1	Lose all river water systems if flood > 311 feet.
C-1		East Entrance to Intermediate Building	1	Lose emergency feedwater and instrument air if flood > 305.5 feet.

TABLE 4-1 (continued)

Sheet 2 of 4

Step	Procedure	Location	Number	Impact if Omitted
3.5.1	(continued)			
B-1		Entrance to Fuel Handling Building	1	If flood > 305 feet, it floods auxiliary building and basement of control building. Therefore, it fails: • Makeup pumps. • 480V MCC IC-ESV. • Decay heat closed cooling water pumps. • Nuclear services closed cooling water pumps. • 480V switchgear. • Intermediate closed cooling water pumps. • Control building ventilation. Subsequent flooding of the control building intake tunnel causes loss of temporary emergency ventilation to control building and loss of all AC power.
B-2		Entrance to Control Building Doorway to BWST Tunnel	1	If flood > 305 feet, it floods auxiliary building and basement of control building. Therefore, it fails: • Makeup pumps. • 480V MCC IC-ESV. • Decay heat closed cooling water pumps. • Nuclear services closed cooling water pumps.

TABLE 4-1 (continued)

Sheet 3 of 4

Step	Procedure	Location	Number	Impact if [illegible]tted
3.5.1	(continued)			• 480V switchgear. • Intermediate closed cooling water pumps. • Control building ventilation. Subsequent flooding of the control building intake tunnel causes loss of temporarly emergency ventilation to control building and loss of all AC power.
[illegible]	[illegible]late door seals.			
[illegible]		Fuel Handling Building (rail entrance)	1	If flood > 305 feet, it fails control building ventilation; subsequent flooding of the control building intake tunnel causes loss of temporarly emergency ventilation to control building and loss of all AC power.
A-3		Auxiliary Building (loading dock)	1	If flood > 305 feet, it fails control building ventilation; subsequent flooding of the control building intake tunnel causes loss of temporarly emergency ventilation to control building and loss of all AC power.
3.5.3	Secure the chlorine cylinders to their concrete support in the circulating water chlorinator house for Unit 1 and the river water chlorinator house in Unit 1.			No impact.
3.5.4	Check and fill f[illegible]il storage tanks.			No impact.
3.5.5	Procure an additional source of diesel fuel oil and make arrangements to airlift the fuel oil to the site.			No impact; plenty of fuel onsite already.

TABLE 4-1 (continued)

Sheet 4 of 4

Step	Procedure	Location	Number	Impact if Omitted
3.5.6	Ver[illegible] closure of watertight doo[illegible].			No impact.
A-4		Reactor Building Canal West Door	1	No impact.
A-5		Reactor Building Canal East Door	1	No impact.
		Reactor Building Access to Tendon Gallery (in alligator pit)	2	No impact.
3.5.7	Increase makeup water to all storage tanks as much as possible, and fill all outdoor tanks to at least Elevation 312' to help prevent flotation in case of site flooding; i.e., all tanks > 7-foot level (305-foot grade)			Tanks kept full anyway, not expected to float until water level is > 311 feet.

TABLE 4-2. POTENTIAL FLOOD IMPACTS ON KEY SYSTEMS

Sheet 1 of 2

Equipment/Systems Needed To Maintain Cold Shutdown	Potential Impacts
Decay Heat Removal	None, pumps located in vaults.
Decay Heat Closed Cooling Water	Floods at 311 feet cause system failure, or at 305 feet if flood panel B-1 or B-2 not installed.
Decay Heat River Water	All river water pumps flooded for floods > 311 feet.
Offsite Power	Offsite power, if not indirectly failed by the storm, would be flooded at the site if flood is greater than about 306 feet.
Diesel Generators	Diesel generators would flood at 311 feet, or, if any one of the following procedural steps are omitted, for floods above 305 feet: Section 3.4.1; D-4, D-1, D-3 or D-2.
Control Building Ventilation	Floods at 310 feet via air intake tunnel or at 305 feet if one of the following procedural steps is omitted: Section 3.4.1, B-1, B-2 or Section 3.4.2, A-2, or A-3.
125V DC Control Power	None; located high in control tower.
120V Vital Instrumentation	None; located high in control tower.
Vital AC Switchgear	Floods at 305 feet if B2 is omitted from procedural step 3.4.1.

TABLE 4-2 (continued)

Sheet 2 of 2

Additional Equipment/Systems Needed To Get to Cold Shutdown Without Offsite Power*	Potential Impacts
Emergency Feedwater and Instrument Air	Floods at 311 feet or above 305 feet, if procedure step 3.4.1, C-1 is omitted.
Makeup or HPI	Floods at 311 feet or above 305 feet if procedure step 3.4.1, B-1 or B-2, is omitted.
Nuclear Services Closed Cooling Water	Floods at 311 feet or above 305 feet if procedure step 3.4.1, B-1 or B-2, is omitted.
Nuclear Services River Water	Floods if > 311 feet.
Intermediate Closed Cooling Water	Floods at 311 feet or at 305 feet if procedure step 3.4.1, B-1 or B-2, is omitted.
Reactor Coolant Pumps	Failure limited by availability of offsite power, which floods at about 306 feet.
Turbine Bypass Valves	None.
Main Feedwater and Associated Systems, Such As: • Condensate • Circulating Water • Secondary Closed Cooling Water • Secondary Cooling River Water	Failure limited by availability of offsite power, which floods at about 306 feet.

*Approximately 25 hours is required before DHR entry conditions are achieved.

TABLE 4-3. HUMAN ACTION ANALYSIS INPUT FOR TOP EVENT EW
(Sheet 1 of 2)

```
HEW1B- EARLY WARNING FOR FLOOD EVENT-RAINFALL W/OUT OP

  INPUT ECHO:

    TYPE OF COGNITIVE PROCESSING IS =                         RULE
    EXPERIENCE LEVEL OF OPERATING CREW IS =                   AVERAGE
    STRESS LEVEL IN CONTROL ROOM IS =                         OPTIMAL CONDITIONS
    QUALITY OF PLANT INTERFACE WITH OPERATORS IS =            POOR
    TYPE OF HUMAN ACTION TASK IS =                            PLANNED MANUAL ACTION
    ADDITIONAL CREW AVAILABLE FOR DIAGNOSIS IS =              FULL SUPPORT
    ADDITIONAL PLANT FEEDBACK TO ALERT OPERATOR =             YES
    TYPE OF DEPENDENCY BETWEEN TASKS IS =                     ZERO
    STATUS OF TASK WHICH THIS ACTION DEPENDS ON IS =          FAILED
    THE MEDIAN ESTIMATE OF THE TIME TO DIAGNOSE IS =            0.500  HOURS
    ESTIMATES OF TIME AVAILABLE ARE =                         POINT ESTIMATE
    (UNITS FOR TIME ARE THE SAME AS FOR THE MEDIAN TIME)        HOURS
    BEST ESTIMATE OF THE TIME AVAILABLE FOR DIAGNOSIS IS = 18.000  HOURS
    (UNITS FOR TIME ARE THE SAME AS FOR THE MEDIAN TIME)

  RESULTS:

         FAILURE FREQUENCY RANGE
            LOWER BOUND= 1.43E-05
            BEST ESTIMATE= 1.43E-04
            UPPER BOUND= 1.43E-03
         BEST ESTIMATE TIME DEPENDENT=  NEGLIGIBLE
         BEST ESTIMATE TIME INDEPENDENT = 1.43E-04
         TOTAL BEFORE ACCOUNTING FOR DEPENDENCY BETWEEN TASKS=1.43E-04
```

TABLE 4-3 (Sheet 2 of 2)

```
HEW1C- EARLY WARNING FOR FLOOD EVENT-HURRICANE W/G OP

  INPUT ECHO:

    TYPE OF COGNITIVE PROCESSING IS =                         RULE
    EXPERIENCE LEVEL OF OPERATING CREW IS =                   AVERAGE
    STRESS LEVEL IN CONTROL ROOM IS =                         POTENTIAL EMERGENCY
    QUALITY OF PLANT INTERFACE WITH OPERATORS IS =            POOR
    TYPE OF HUMAN ACTION TASK IS =                            PLANNED MANUAL ACTION
    ADDITIONAL CREW AVAILABLE FOR DIAGNOSIS IS =              FULL SUPPORT
    ADDITIONAL PLANT FEEDBACK TO ALERT OPERATOR =              YES
    TYPE OF DEPENDENCY BETWEEN TASKS IS =                     ZERO
    STATUS OF TASK WHICH THIS ACTION DEPENDS ON IS =          FAILED
    THE MEDIAN ESTIMATE OF THE TIME TO DIAGNOSE IS =            0.500  HOURS
    ESTIMATES OF TIME AVAILABLE ARE =                         POINT ESTIMATE
    (UNITS FOR TIME ARE THE SAME AS FOR THE MEDIAN TIME)        HOURS
    BEST ESTIMATE OF THE TIME AVAILABLE FOR DIAGNOSIS IS =  12.000  HOURS
    (UNITS FOR TIME ARE THE SAME AS FOR THE MEDIAN TIME)

  RESULTS:

        FAILURE FREQUENCY RANGE
           LOWER BOUND= 1.47E-05
           BEST ESTIMATE= 1.47E-04
           UPPER BOUND= 1.47E-03
        BEST ESTIMATE TIME DEPENDENT= 3.78E-06
        BEST ESTIMATE TIME INDEPENDENT = 1.43E-04
        TOTAL BEFORE ACCOUNTING FOR DEPENDENCY BETWEEN TASKS=1.47E-04
```

TABLE 4-4. HUMAN ACTION ANALYSIS FOR TOP EVENT CS
(Sheet 1 of 2)

```
HCD6A- COOLDOWN AND DEPRESSURIZE GIVEN FLOOD,RAINFALL W/OP SUCCESS

 INPUT ECHO:

   TYPE OF COGNITIVE PROCESSING IS =                            RULE
   EXPERIENCE LEVEL OF OPERATING CREW IS =                      AVERAGE
   STRESS LEVEL IN CONTROL ROOM IS =                            OPTIMAL CONDITIONS
   QUALITY OF PLANT INTERFACE WITH OPERATORS IS =               POOR
   TYPE OF HUMAN ACTION TASK IS =                               PLANNED MANUAL ACTION
   ADDITIONAL CREW AVAILABLE FOR DIAGNOSIS IS =                 FULL SUPPORT
   ADDITIONAL PLANT FEEDBACK TO ALERT OPERATOR =                YES
   TYPE OF DEPENDENCY BETWEEN TASKS IS =                        MEDIUM
   TITLE OF TASK WHICH THIS ACTION DEPENDS ON IS =              HEW1
   STATUS OF TASK WHICH THIS ACTION DEPENDS ON IS =             SUCCEEDED
   THE MEDIAN ESTIMATE OF THE TIME TO DIAGNOSE IS =               0.100  HOURS
   ESTIMATES OF TIME AVAILABLE ARE =                            POINT ESTIMATE
   (UNITS FOR TIME ARE THE SAME AS FOR THE MEDIAN TIME)           HOURS
   BEST ESTIMATE OF THE TIME AVAILABLE FOR DIAGNOSIS IS =         1.000  HOURS
   (UNITS FOR TIME ARE THE SAME AS FOR THE MEDIAN TIME)

 RESULTS:

        FAILURE FREQUENCY RANGE
           LOWER BOUND= 3.57E-04
           BEST ESTIMATE= 3.57E-03
           UPPER BOUND= 3.57E-02
        BEST ESTIMATE TIME DEPENDENT= 1.17E-03
        BEST ESTIMATE TIME INDEPENDENT = 3.00E-03
        TOTAL BEFORE ACCOUNTING FOR DEPENDENCY BETWEEN TASKS=4.17E-03
```

TABLE 4-4 (Sheet 2 of 2)

```
HCD6C- COOLDOWN AND DEPRESSURIZE GIVEN FLOOD,HURRICANE WITH OP FAILED

  INPUT ECHO:

    TYPE OF COGNITIVE PROCESSING IS =                           RULE
    EXPERIENCE LEVEL OF OPERATING CREW IS =                     AVERAGE
    STRESS LEVEL IN CONTROL ROOM IS =                           POTENTIAL EMERGENCY
    QUALITY OF PLANT INTERFACE WITH OPERATORS IS =              POOR
    TYPE OF HUMAN ACTION TASK IS =                              PLANNED MANUAL ACTION
    ADDITIONAL CREW AVAILABLE FOR DIAGNOSIS IS =                FULL SUPPORT
    ADDITIONAL PLANT FEEDBACK TO ALERT OPERATOR =               YES
    TYPE OF DEPENDENCY BETWEEN TASKS IS =                       MEDIUM
    TITLE OF TASK WHICH THIS ACTION DEPENDS ON IS =             HEW1
    STATUS OF TASK WHICH THIS ACTION DEPENDS ON IS =            SUCCEEDED
    THE MEDIAN ESTIMATE OF THE TIME TO DIAGNOSE IS =              0.100  HOURS
    ESTIMATES OF TIME AVAILABLE ARE =                           POINT ESTIMATE
    (UNITS FOR TIME ARE THE SAME AS FOR THE MEDIAN TIME)          HOURS
    BEST ESTIMATE OF THE TIME AVAILABLE FOR DIAGNOSIS IS =        0.000  HOURS
    (UNITS FOR TIME ARE THE SAME AS FOR THE MEDIAN TIME)

  RESULTS:

          FAILURE FREQUENCY RANGE
             LOWER BOUND= 1.71E-01
             BEST ESTIMATE= 8.57E-01
             UPPER BOUND= 1.00E+00
          BEST ESTIMATE TIME DEPENDENT= 1.00E+00
          BEST ESTIMATE TIME INDEPENDENT = 3.00E-03
          TOTAL BEFORE ACCOUNTING FOR DEPENDENCY BETWEEN TASKS=1.00E+00
```

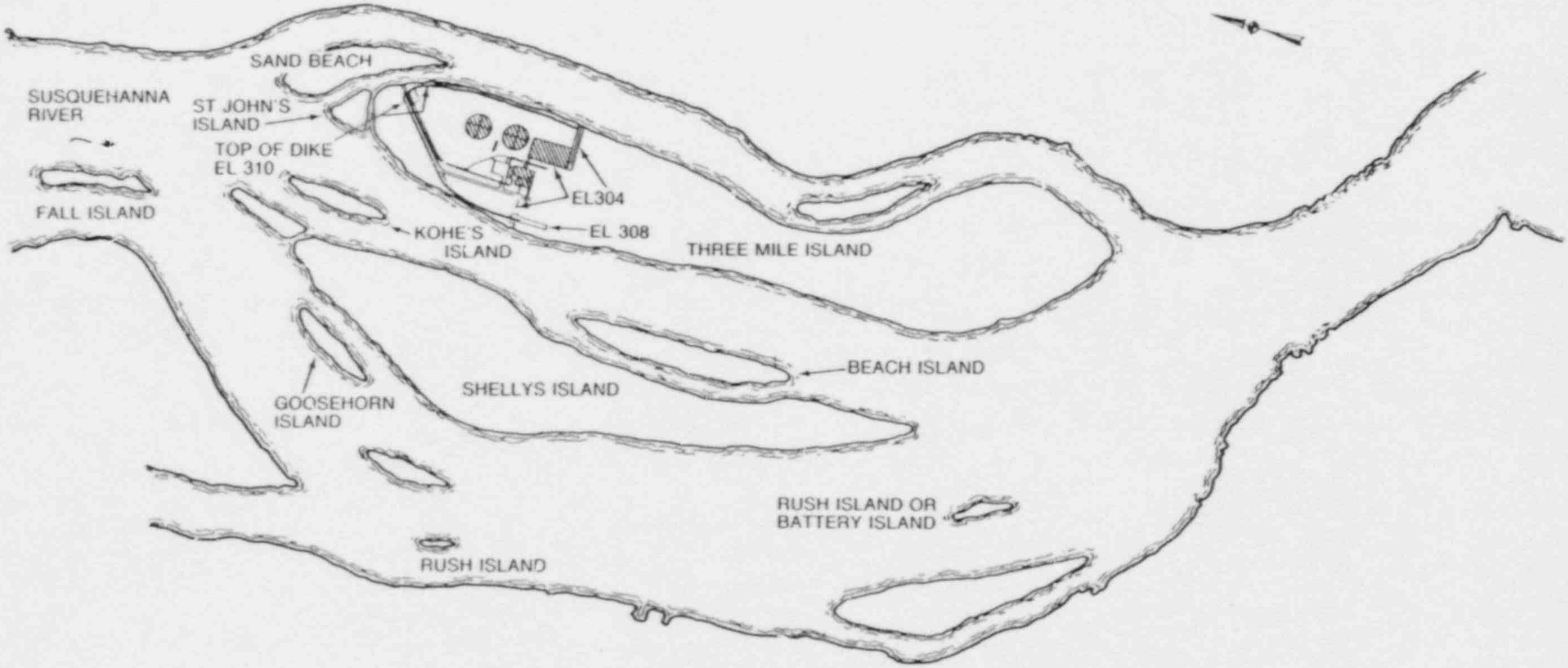


FIGURE 4-1. TMI SITE

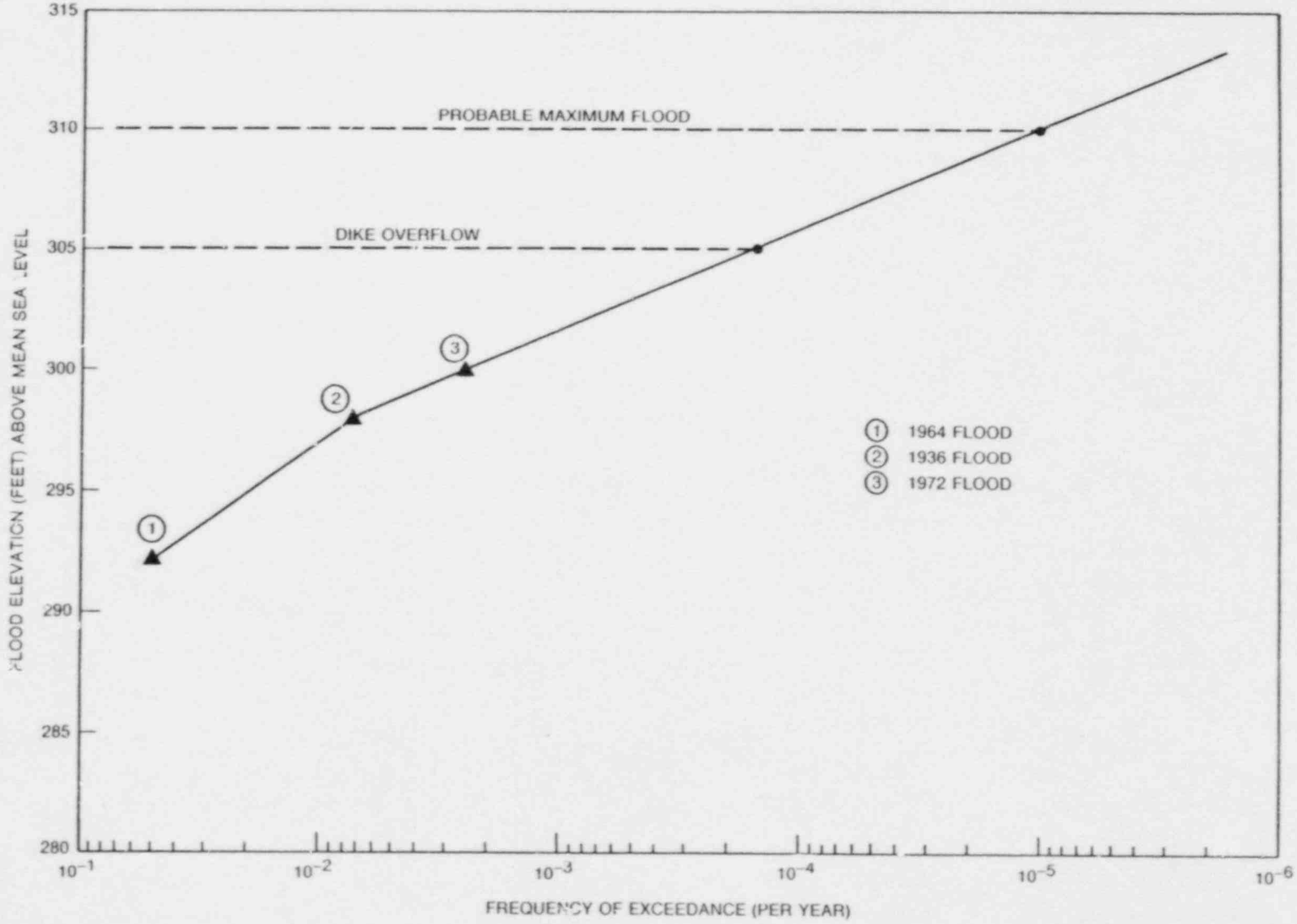


FIGURE 4-2. FLOOD FREQUENCY-MAGNITUDE CURVE

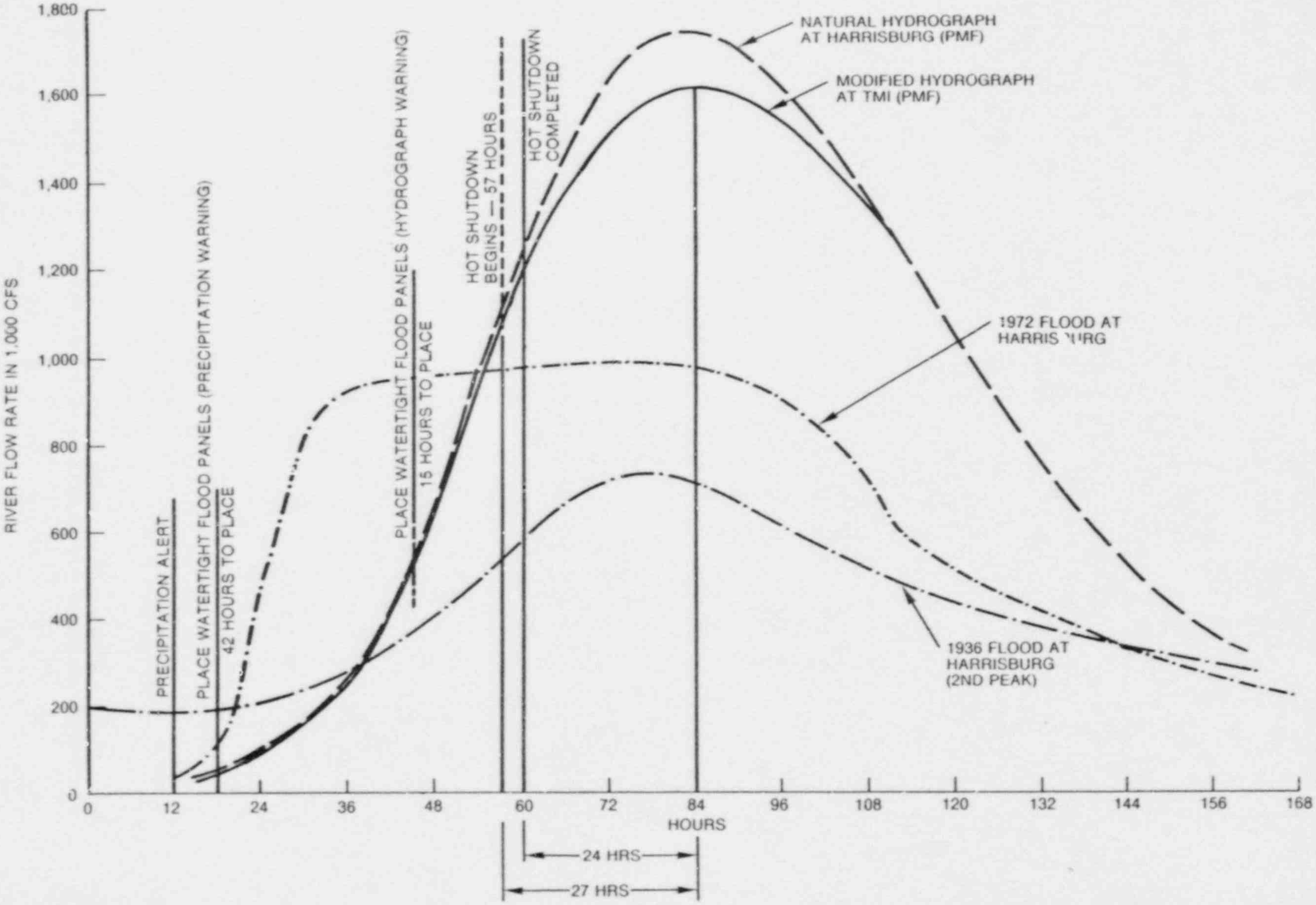
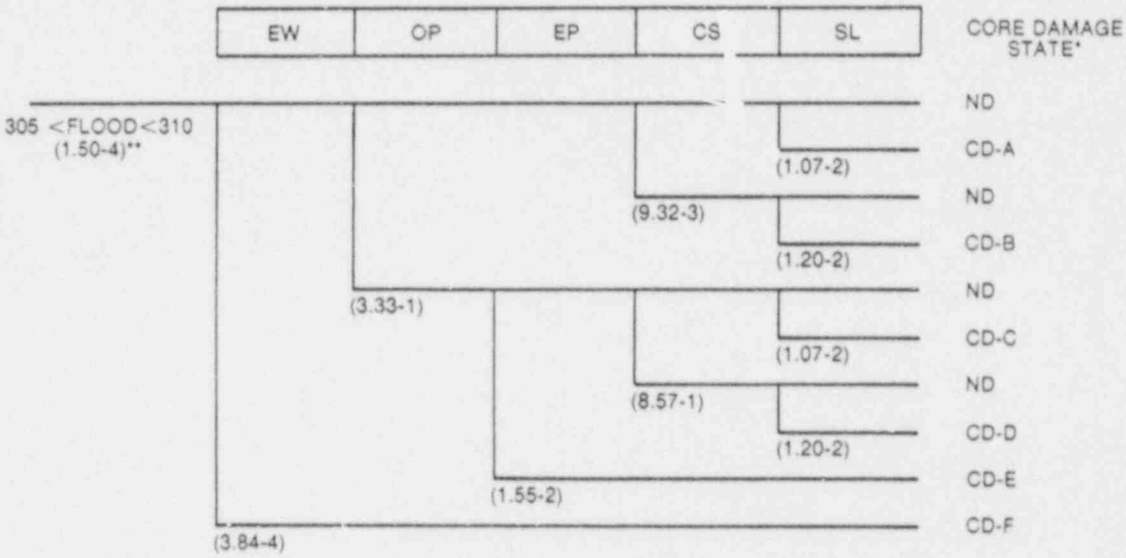


FIGURE 4-3. HYDROGRAPHS FOR VARIOUS FLOODS AT OR NEAR THE SITE



*ND ≡ NO CORE DAMAGE
CD ≡ CORE DAMAGE

**NOTE: EXPONENTIAL NOTATION IS SHOWN IN ABBREVIATED FORM;
i.e., 1.50-4 = 1.50×10^{-4}.

FIGURE 4-4. EVENT TREE FOR FLOOD WITH WATER ELEVATION BETWEEN 305 AND 310 FEET

5. TORNADO WIND AND MISSILE HAZARD

5.1 INTRODUCTION AND SUMMARY

Winds can affect critical structures at the plant site in at least two ways. If wind forces exceed the load capacity of a building or other external facility, the incident walls or framing might collapse or the structure overturn from the excessive loading. If the wind is strong enough, such as in a tornado, it might be capable of lifting materials and thrusting them as missiles against some of these critical facilities. Critical components or other contents of facilities not designed to resist missile penetration might be damaged and lose their function. This section presents an analysis of the risk to the TMI-1 plant from tornado wind and missile. It is concluded that neither a tornado wind load nor a potential missile generated in a tornado event leads to scenarios that would contribute significantly to the total core damage frequency.

5.2 TORNADO WIND HAZARD AND FREQUENCY

To estimate ϕ_t, the frequency of a tornado striking the plant, we use the following algorithm (Reference 5-1):

$$\phi_t = n \cdot \frac{w}{A} \tag{5.1}$$

where w is the mean path area of a tornado in square miles, A is the area of interest within which it is assumed the tornado could strike the site, and n is the mean number of tornado occurrences per year in this area. According to tornado data for the period 1917 through 1969 (Reference 5-2), there were 22 tornadoes within 25 miles of the site. This leads to an annual occurrence rate of 0.43 in the area. Reference 5-1 gives a mean path area of 2.82 square miles for tornadoes. Therefore, the result is

$$\phi_t = (0.43)\ \frac{2.82}{\pi(25)^2} = 6.18 \times 10^{-4} \text{ strike/mile}^2 \text{ per year} \tag{5.2}$$

which is the annual frequency of all tornadoes regardless of their intensity. This frequency should be modified to obtain the frequency of those tornadoes that are strong enough to damage various critical structures of the plant. The annual frequency, ϕ, of excessive tornado wind load on the structures can be found by

$$\phi = \phi_t \cdot \phi_{v|t} \tag{5.3}$$

where $\phi_{v|t}$ is the fraction of tornadoes with peak windspeed greater than v.

Tornado wind exceedance probability, $\phi_{v|t}$, is a more difficult quantity to estimate due to the inaccuracy of indirect measuring techniques and the lack of a good analytical model for tornado behavior. An analysis of 4,582 tornados whose intensities were classified according to the Fujita F-scale is presented in Reference 5-3. Table 5-1 shows the histogram of frequencies of tornado windspeeds based on a Johnson S_B distribution fit to the data for NRC tornado Region 1, which is applicable to the TMI-1 site. This distribution will be used later to obtain the frequency of tornadoes with windspeeds exceeding wind fragility of the critical structures of the plant.

5.3 TORNADO WIND FRAGILITY OF STRUCTURES

The design basis tornado windspeed for the seismic category I structures of the TMI-1 plant which include all critical structures except metal tanks located outdoors, is 360 mph, which is composed of a translational component of 70 mph and a rotational component of 290 mph (Reference 5-2). Seismic category I structures include the containment building, intermediate and auxiliary buildings, control tower, diesel generator room, and river water pump house. From Table 5-1 we can see that the frequency of wind-speed exceedance in Region 1 for tornado intensity, $F \geq F6$, is obviously an upper bound for the frequency of windspeeds exceeding 360 mph. Therefore, 0.0005 was conservatively chosen as the value of $\phi_{v|t}$.

Although no upper bound for the windspeed is indicated in this histogram, Reference 5-3 proposes a value of 300 mph as the maximum windspeed in Region 1. Other experts indicate that a tornado windspeed higher than 400 mph is not possible due to atmospheric friction. In this analysis, we assumed that 400 mph is the maximum windspeed for Region 1 tornadoes.

By combining the conservatively high values of ϕ_t and $\phi_{v|t}$, the annual frequency, ϕ, of tornado windspeeds in excess of 360 mph is found.

$$\phi = (6.18 \times 10^{-4})(5 \times 10^{-4}) = 3.09 \times 10^{-7} \text{ per square mile, per year}$$

Tornado wind load on seismic category I structures can be calculated by obtaining the maximum windspeed pressure, q_{max}, from the following formula (Reference 5-2)

$$q_{max}(V) = 0.00256V^2 \tag{5.4}$$

where V is the total tornado windspeed. Therefore, for V = 360 mph, we (360) = 332 psf.

or V = 400 mph, which was used as the maximum possible tornado windspeed, we obtain $q_{max}(400) = 410$ psf, which is higher than the design pressure calculated for a 360-mph windspeed by a factor of 1.23. The conservative factor of safety applied to material yield stress to obtain design allowable stresses was judged to be well within the margin of safety for category I structures. Therefore, the lower end of tornado wind fragility curve for such structures can be assumed to be in the vicinity of 400 mph.

We conservatively assume a step function fragility curve for wind load on the safety-related concrete structures at 400 mph. In other words, we assume that these structures do not fail under 400-mph wind load and that failure is certain above that value.

There is some critical equipment outdoors that can be damaged at windspeeds less than 360 mph. For instance, power lines, transformers, and related equipment would be lost in weaker but more frequent tornadoes. It is assumed in this analysis that in a tornado event the offsite power is lost.

The critical exterior metal vessels, such as the borated water storage tank and the condensate storage tank, may also be subject to failure from negative or positive pressures generated by winds at tornado levels. However, these tanks are normally about two-thirds to three-fourths full when in service, with resultant uniform internal pressures ranging to over 2,000 psf at the bottom walls. As long as they carry such a capacity, large external wind pressures cannot develop sufficiently to cause asymmetrical loads that would threaten buckling of the tanks although the tank top might be blown out from negative pressures. This, however, would not create buckling effects on the tank walls. Therefore, loss of contents from these metal vessels due to tornado wind load is highly unlikely.

A bounding analysis as mentioned in Reference 5-4 indicates that the damage due to negative or positive pressure to the tanks may occur at windspeeds greater than 150 mph. However, according to Reference 5-5, the analysis referred to in Reference 5-4 is based on overly conservative assumptions, which do not, in any way, apply to the construction and wind load capacity of typical BWSTs and CSTs. Reference 5-5 also indicated that some recent and more realistic analyses show a much greater wind load capacity of 350 mph or more for these tanks. With this wind load capacity, using the windspeed distribution of Table 5-1 and the site-specific tornado hit frequency calculated in Section 1, the annual frequency of tornado wind damage to the BWST or CST is also 3.09×10^{-7} per year. This value conservatively assumes that a tornado striking the site also strikes the tanks even though the tanks constitute only a fraction of the total area of the plant.

5.4 TORNADO WIND-INITIATED SCENARIOS

Based on the discussion of the previous section, seismic class I structures of the plant are not expected to be damaged due to the tornado wind load. Even if windspeed is assumed to exceed 400 mph, an event that has a mean annual frequency less than 3.09×10^{-7}, the total annual frequency of core damage scenarios initiated by the failure of seismic class I buildings under wind load would be several orders of magnitude less than the core damage frequency due to other initiators.

As stated before, it is assumed that, given a tornado strike, offsite power is lost with a probability of 1. Scenarios in which tornado damage is limited to loss of offsite source of power are included in the loss of power scenarios. However, as described in the previous section, the CST and BWST may also fail in a tornado event. The joint occurrence of loss of offsite power and failure of CST results in a loss of offsite power scenario with no feedwater capability (the emergency feedwater system requires the availability of CST). Without BWST, which is also assumed to fail due to the same tornado load, the scenario would lead to core damage. The mean annual frequency of this scenario is less than 3.09×10^{-7}, which is a very small contribution to the total core damage frequency from all other scenarios. A similar scenario is discussed in Section 5.7.

5.5 TORNADO MISSILE HAZARD AND FREQUENCY

Tornado missile analysis involves information about the likelihood of a spectrum of available missiles in the plant vicinity, representation of the wind field in the tornado, and aerodynamic behavior relative to "liftoff" and flight of the potential missile. The analysis leads to a spectrum of missiles and missile impact velocities with their respective probabilities. A detailed analysis that integrated all these effects for typical plant layouts has previously been performed (Reference 5-6). The results of that work are considered to be reasonable gross estimates for the hazard of tornado missiles at Three Mile Island.

In Reference 5-6, calculations were made using tornado histories of each tornado region defined by the NRC. The analysis used a typical two-unit plant layout to establish the target envelope and a 26-missile spectrum, which includes the six missiles defined in the NRC Standard Review Plan, Section 3.5.1.4 (wood plank, steel pipe, steel rod, utility pole, and automobile). In general, the 26-missile spectrum of Reference 5-6 is more conservative than the Standard Review Plan spectrum with respect to damage potential. Calculations were made for several cases including a two-unit plant. Assuming 1,000 available missiles during the operating phase, the study obtains the following upper and lower bounds (at 95% confidence level) for the annual impact and damage frequency for all structures of a two-unit plant in NRC Region 1:

Upper Bound = 8.63×10^{-7}

Lower Bound = 6.64×10^{-9}

The thickness of the targets considered in this calculation ranges from 12 to 18 inches for targets such as the diesel generator building and service water intake structure, and from 24 to 36 inches for the containment. Storage tanks such as the BWST and CST are also considered to be enclosed by 12-inch thick concrete walls.

The above impact/damage frequency bounds were calculated on the basis of a tornado strike frequency of 2.3×10^{-3} per year, per square mile, while the strike frequency at the Three Mile Island site is 6.18×10^{-5} per year, per square mile. Adjusting for this factor results in the following bounds for a two-unit plant

Upper Bound = 2.32×10^{-7} per year, per square mile

Lower Bound = 1.78×10^{-9} per year, per square mile

In this analysis, the above values are used as the 5th and 95th percentiles of a lognormal distribution (a somewhat conservative use of the confidence bounds) resulting in a mean value of 6.08×10^{-8}.

To use the above results for TMI-1, we define a missile strike/damage density as the ratio of the annual frequency of impact/damage to any plant structure divided by the total exposed surface area of all TMI-1 structures. By multiplying this strike/damage density by the surface area of any target, we can then calculate the annual strike/damage frequency for that specific target.

$$\phi_i = \phi_s \frac{A_i}{A_t} \qquad (5.5)$$

where

ϕ_i = annual frequency of a tornado strike/damage for the i-th target.

ϕ_s = annual frequency of hitting any structure (safety-related and turbine building).

A_i = exposed area of the i-th target.

A_t = total exposed surface area of structures.

The surface area of the two-unit plant studied in Reference 5-6 is about a factor of 2 higher than the total exposed surface area of TMI-1. Note that using the TMI-1 total area in Equation (5.5) results in an overestimation of the missile strike/damage density by about a factor of 2.

Table 5-2 provides the ratio A_i/A_t for different safety-related structures at TMI-1. Also given in the table are mean strike/damage frequencies for each target, which, except for BWST and CST, were calculated on the basis of the above algorithm. The value listed for BWST and CST is the tornado missile hit frequency obtained in Reference 5-6 for the Unit 2 tank enclosure. It is assumed, as will be discussed in the following section, that a missile hit results in failure of those tanks, since, unlike the example plant of Reference 5-5, the TMI-1 BWST and CST are not protected by concrete wells.

5.6 TORNADO MISSILE FRAGILITY OF STRUCTURES

The values obtained in Table 5-2 are the annual frequency of inside wall scabbing for the concrete safety-related structures of TMI-1. All damages are believed to be localized; therefore, it is extremely conservative to assume scabbing causes damage to all the contents. It is also assumed that a hit by tornado-generated missiles would cause failure of such critical equipment located outdoors as the BWST, the CST, and the transformers.

5.7 TORNADO MISSILE-INITIATED SCENARIOS

The low frequency of a tornado missile hitting various critical structures (Table 5-2), combined with the fact that damage to the Class 1 structures would certainly be localized and not enough to impact several vital components, leads to extremely low frequencies of all tornado missile-initiated accident scenarios that can be hypothesized. Such scenarios can be easily shown to be dominated by others by several orders of magnitude.

In line with the discussion in Section 5.4, it is assumed that a tornado would cause offsite power to be lost to the plant. One could also postulate failure of the critical outdoor tanks; i.e., CST and BWST.

The frequency of losing offsite power and failing the CST due to tornado missiles is less than 4.27×10^{-7} per year, which is, according to Table 5-2, the frequency of tornado missile hitting either the CST or the BWST. In this case, the scenarios of interest, as in the case of tornado wind, would be loss of offsite power, unavailability of the emergency feedwater system as a result of failure of the CST, and eventual core damage. However, at a frequency of 4.27×10^{-7}, this scenario is a negligible contributor to the total core damage frequency even when the contribution from failure of the CST and BWST due to tornado wind load discussed in Section 5-4 is added.

5.8 REFERENCES

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5-2. GPU Nuclear Corporation, "TMI-1 Final Safety Analysis Report," Update 1, July 1982.

5-3. Twisdale, L. A., "Tornado Data Characterization and Windspeed Risk," Journal of the Structural Division, Proceedings of ASCE, Vol. 104, No. ST10, October 1978.

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5-5. Phone conversation with Dr. John Reed of Jack R. Benjamin and Associates, July 13, 1987.

5-6. Twisdale, L. A., W. L. Dunn, and J. Cho, "Tornado Missile Risk Analysis," Electric Power Research Institute, EPRI NP-768, May 1978.

TABLE 5-1. TORNADO WINDSPEED FRACTIONS

F-Scale	Windspeed Range (mph)	Frequency (NRC Region 1)
0	40 - 72	0.2440
1	72 - 112	0.4241
2	112 - 157	0.2375
3	157 - 206	0.0735
4	206 - 260	0.0172
5	260 - 318	0.0032
6	318 - 380	0.0005
> 6	> 380	

TABLE 5-2. RESULTS OF TORNADO MISSILE HIT FREQUENCY BY TARGET

Structure	Surface Ratio (A_i/A_t)	Mean Hit Frequency per Year
Containment Building	0.2054	1.25-8*
Control Tower	0.1159	7.04-9
Intermediate Building	0.1526	9.27-9
Auxiliary Building	0.0870	5.28-9
Diesel Generator Building	0.0505	3.07-9
Intake Screens and Pump House	0.0612	3.72-9
Outdoor Tanks (BWST, CST)	0.0134	4.27-7**

*Exponential notation is indicated in abbreviated form; i.e., 1.25-8 = 1.25 x 10^{-8}.

**This value is the mean annual tornado missile hit frequency for the tank enclosure of Unit 2 of the example plant in Reference 5-6. For TMI-1, damage is assumed to occur with probability 1 if the tanks are hit by a missile.

6. TURBINE MISSILES HAZARD

6.1 INTRODUCTION

Missiles generated in the event of turbine failure can potentially damage safety-related systems. Although highly unlikely, serious damage to a series of critical equipment in combination with a turbine failure may lead to undesirable consequences. In this section, the likelihood of generating turbine missiles is estimated, and the most probable consequences are analyzed.

The fundamental equation used to find the annual frequency, f, of serious damage to a specific system is

$$f = f_1 \cdot f_2 \cdot f_3 \qquad (6.1)$$

where f_1 is the annual frequency of missile generation due to turbine failure; f_2 is the conditional probability of a missile striking a barrier to an essential system, given that a turbine missile has been generated; and f_3 is the conditional probability of penetrating the barrier, striking system components and causing unacceptable damage to the system, given that a missile strikes the barrier.

6.2 FREQUENCY OF TURBINE MISSILE GENERATION, f_1

The TMI-1 plant uses General Electric Company turbine generators. A number of studies have been performed on the frequency of turbine failures that lead to generation of high energy missiles. Table 6-1 gives several estimates that are judged to be relevant to our study.

The two failure modes for release of external missiles are: (1) failure at or near operating speed and (2) overspeed failure.

The estimates provided by Bush and Heasler (Reference 6-1) are based on analysis of statistical records of failures relevant to turbine generators of the type used in nuclear power plants.

Table 6-1 also gives two different types of estimates provided by a General Electric Company report (Reference 6-2). One is based on historical records of GE turbines and is argued in the report as being inapplicable to modern GE nuclear turbines primarily because the failure incidents used in the statistical analysis involved turbine units different from the modern GE nuclear turbines. The second estimate recommended by Reference 6-2 for the two failure modes (Table 6-1) is based on analysis of causes and conditions for turbine generator failure. The frequency of release of external missiles, in this case, is calculated as a product of the conditional frequencies in a sequence of primary events leading to missile ejection.

The frequency for the first mode (operating speed) is based almost entirely on brittle fracture of wheel material that, according to

Reference 6-2, is the dominant cause of wheel burst at speeds up to 130% of the rated speed. In the second mode event, a perfect wheel is assumed to fail so that a postulated control system failure is the sole contributing cause. Note that both the statistical data and the vendor recommendation show a relatively small contribution by overspeed failure. Hence, the risk of turbine missiles is not very sensitive to possible improvements in control system reliability.

As can be seen from Table 6-1, these estimates are several orders of magnitude smaller than other estimates that are based on analysis of statistical records. There is an increasing amount of evidence that would justify higher frequencies than those suggested by Reference 6-2. For example, the reference does not consider the possibility of stress corrosion. A document by the General Electric Company (Reference 6-3) more recent than Reference 6-2 suggests that some cases of stress corrosion cracking have been observed in GE turbine generators.

To express our uncertainty about the frequency of release of external missiles due to turbine failure, we use the estimate of Reference 6-1 for each failure as an upper bound due to the fact that it does not directly represent modern GE turbine generators. The estimate from Reference 6-2 for the corresponding failure mode will be used as the lower bound.

We use the lower and upper bounds as the 5th and 95th percentiles of a lognormal distribution. The resulting mean values of distributions for the two failure modes are listed in Table 6-1. Other characteristics are given below (all numbers are events per turbine year):

- Missile Generation Frequency at Operating Speed (f_1')
    - 95th Percentile: 1.1×10^{-4}
    - 50th Percentile: 9.90×10^{-7}
    - 5th Percentile: 8.70×10^{-9}
- Missile Generation Frequency at Overspeed (f_1'')
    - 95th Percentile: 4.3×10^{-5}
    - 50th Percentile: 4.6×10^{-7}
    - 5th Percentile: 5.0×10^{-9}

6.3 CONDITIONAL PROBABILITY OF MISSILE IMPACT, f_2

To obtain f_2, the conditional probability of a missile striking a barrier to an essential system, given turbine failure, one must analyze the behavior of potential missiles ejected from the turbine, taking into account the kinetic energy and possible trajectories of the missiles as well as the location of potential barriers.

Detailed analysis of the impact frequencies was beyond the scope of this screening analysis. Instead, the simple method of Reference 6-4 was used in conjunction with conservative assumptions to achieve a bounding analysis.

Potential missiles are assumed to fall into the categories of high trajectory and low trajectory. Reference 6-4 provides the following simple approximations for the impact frequency of missiles in the two categories:

$$\text{High Trajectory} = f_2^H \approx \frac{220}{\theta_2^\circ - \theta_1^\circ} \times \frac{A_{roof}}{v^4} \tag{6.2}$$

$$\text{Low Trajectory} = f_2^L \approx \frac{9.1}{\theta_2^\circ - \theta_1^\circ} \times \frac{A_{wall}}{d^2} \tag{6.3}$$

where

v = missile velocity at ejection (m/s).

d = distance of the target from turbine axis (m).

A_{roof} = roof area of the target (m^2).

A_{wall} = wall area of the target (m^2).

θ = horizontal angular deviation of the missile (degrees). $\theta_2°$ and $\theta_1°$ are two bounds beyond which missile distribution is assumed to be zero. Uniform missile distribution is assumed for angles between $\theta_2°$ and $\theta_1°$.

Based on review of the plant layout and turbine orientation, and assuming slightly conservative values of θ_2 and θ_1 ($\theta_2 = -\theta_1 = 30°$), the following targets of barriers to essential systems and direct system component targets for high trajectory missiles and for low trajectory missiles are:

HTM Targets	LTM Targets
Containment Building	Containment Building
Control Tower	
Intermediate Building	
Auxiliary Building	
Fuel Handling Building	
Diesel Generator Room	
Turbine Building	
Intake Screens and Pump House	
Outdoor Tanks (BWST and CST)	

To calculate f_2 using Equations (6.2) and (6.3), values of A_{roof} and A_{wall} were calculated for each of the above targets. For low trajectory missiles, only that portion of the containment building that falls within the ejection cone, defined by the horizontal angular deviation of the missiles ($\theta_2° = -\theta_1° = 30°$), was included.

For high trajectory missiles, the following ejection velocities were considered, corresponding to failures at operating speed, as well as to overspeed failures.

Failure Mode	Ejection Velocity (m/s)		
	Low	High	Average
Operating Speed	80	120	100
Overspeed	110	160	135

These values were based on the information provided in Reference 6-5. The low and the high values are the average of the low and high values based on different assumptions regarding missile shape and energy, turbine model, casing penetration model, etc. The average values are based on assuming uniform distribution between low and high values.

Table 6-2 summarizes the result of f_2 calculations for the high trajectory missiles. The containment building is the only likely barrier for the low trajectory missiles. The corresponding f_2 for the low trajectory missile was calculated to be 3.46×10^{-2}.

6.4 TURBINE MISSILE FRAGILITY OF STRUCTURES, f_3

In general, such thick, reinforced concrete walls and roofs as those in place in nuclear power plants provide a powerful barrier against turbine-generated missiles.

The likelihood of perforation or back scabbing for a missile depends on such missile characteristics as weight, ejection speed, shape, angle of ejection, and angle of impact and on such target characteristics as concrete thickness, degree of reinforcement, etc.

Some full-scale concrete impact tests indicate that, for typical turbine missiles, 4 to 5-foot thick concrete walls show no perforation or back scabbing (Reference 6-6). Scale model test (1/11) of 4.5-foot thick heavily reinforced wall (Reference 6-7) also indicates that such walls can contain missiles at an impact velocity of up to 650 feet per second (198 meters per second).

The containment building at TMI-1 is a reinforced concrete structure with 3-1/2-foot thick walls. The test results from Reference 6-7 show that such walls contained a 3,250-pound turbine missile at an impact velocity as high as 650 feet per second (198 meters per second). When the missile weight was increased to 4,000 pounds, it was contained for an impact velocity of 520 feet per second (159 meters per second), but perforation occurred when the velocity was increased to 650 feet per second. Approximately doubling missile weight (8,300 pounds) resulted in perforation at 520 feet per second and 650 feet per second. However, the missile was contained for an impact velocity of 420 feet per second (128 meters per second).

According to Reference 6-8, based on a study of major missiles that might escape the TMI-1 turbine casing, the last stage wheel of the TMI-1 turbine is considered to have the worst combination of weight, size, and energy. The predicted properties and depth of penetration of the last stage wheel containment building are summarized in Table 6-3. By comparing the missile characteristics of this table with test results from References 6-6 and 6-7, it can be seen that perforation of the reactor building by turbine missiles is highly unlikely. In this analysis, we make a conservative assessment of the likelihood of missile penetration of the containment building by assuming the following (lognormal) distribution of f_3.

5th percentile = 0.01

50th percentile = 0.05

95th percentile = 0.25

Mean = 0.08

For other concrete structures identified as targets and listed in Section 6.3, we will use a much higher likelihood of perforation because of thinner wall and roof thicknesses. For those structures, the following distribution (truncated lognormal) will be used for f_3:

5th percentile = 0.50

95th percentile = 0.95

Mean = 0.70

It will also be assumed that $f_3 = 1$ for the exterior metal tanks (CST and BWST).

6.5 TURBINE MISSILE SCENARIOS

The mean annual frequency of damage to different structures due to turbine missile are listed in Table 6-4. Each frequency is calculated for each category from Equation (6.1) by using the appropriate numbers for f_1, f_2, and f_3 presented in previous sections for low and high trajectory missiles and adding the contributions of each category to get a total damage frequency for each structure.

The most critical single location that can be hit by a turbine missile with relatively high frequency and serious consequences is the containment building. All other structures have such a low damage frequency that, even by having assumed a conditional frequency of core damage equal to 1, given missile penetration, the contribution to the core damage frequency is negligible. In the case of the containment building, if the missile were to penetrate, that missile and secondary missiles are not expected to damage multiple systems inside the containment. Among possible scenarios, one that seems to be bounding due to the spatial arrangement of systems inside the containment is to assume that the missiles would damage one or two steam generators leading, at the most, to a large LOCA. This event, in addition, results in a loss of containment isolation and containment spray system failure due to missile hit. When these effects are combined with independent unavailability of one high pressure injection or low pressure injection train, the frequencies of the core melt scenarios once again become very small (at least one or two orders of magnitude smaller than the containment penetration scenario, 2.3 x 10^{-7} per year) and are bounded by other scenarios leading to similar plant damage states.

6.6 REFERENCES

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6-2. General Electric Company, "Hypothetical Turbine Missiles - Probability of Occurrence," Memo Report, March 1973.

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6-7. C. M. Romander, "Scale Model Tests of Turbine Missile Containment by Reinforced Concrete," presented at Electric Power Research Institute Seminar on Turbine Missile Effects in Nuclear Power Plants, Palo Alto, California, October 25-26, 1982.

6-8. GPU Nuclear Corporation, TMI-1 FSAR, updated August 1, 1982.

TABLE 6-1. ESTIMATES OF THE MEAN ANNUAL FREQUENCY OF TURBINE MISSILE GENERATION

Source	Failure Mode		Total
	Operating Speed (f_1)	Overspeed (f_1')	
Bush and Heasler*	1.1×10^{-4}	4.3×10^{-5}	1.6×10^{-4}
GE (statistics)**	-	-	1.4×10^{-4}
GE (analysis)**	8.7×10^{-9}	5.0×10^{-9}	1.4×10^{-8}
This Report (mean value)	6.3×10^{-5}	2.0×10^{-5}	8.3×10^{-5}

*Reference 6-1.
**Reference 6-2.

TABLE 6-2. CONDITIONAL FREQUENCY OF IMPACT FOR HIGH TRAJECTORY MISSILES

Target	Roof Area (m^2)	Impact Frequency	
		Operating Speed	Overspeed
Containment Building	1.47+3	5.39-5	1.62-5
Control Tower	8.85+2	3.24-5	9.77-6
Intermediate Building	1.18+3	4.32-5	1.30-5
Auxiliary Building	1.17+3	4.29-5	1.29-5
Fuel Handling Building	8.71+2	3.20-5	9.62-6
Diesel Generator Room	8.83+2	3.24-5	9.75-6
Intake Screens and Pump House	1.00+3	3.66-5	1.10-5
Outdoor Tanks (BWST, CST)	6.57+1	2.42-6	7.26-7

NOTE: Exponential notation is indicated in abbreviated form; i.e., 1.47+3 = 1.47 x 10^3; 5.39-5 = 5.39 x 10^{-5}.

TABLE 6-3. CHARACTERISTICS OF THE LAST STAGE WHEEL MISSILES
(Reference 6-8)

Fragment Angle	Weight (pounds)	Impact Area (ft.^{2})		Final Energy (ft.-lb.)	Final Velocity	Depth of Penetration (inches)	
		Side On	End On			Side On	End On
90°	4,458	6.83	3.17	15.0×10^6	464.0	5.45	11.8
120°	5,944	8.37	3.66	20.5×10^6	447.3	5.6	12.8
180°	8,916	9.66	4.83	17.2×10^6	351.0	5.04	10.1

TABLE 6-4. ANNUAL FREQUENCY OF TURBINE MISSILE PENETRATION

Target	High Trajectory		Low Trajectory		Total
	Operating Speed	Overspeed	Operating Speed	Overspeed	
Containment Building	2.71-10	2.59-11	1.74-7	5.54-8	2.30-7
Control Tower	1.43-9	1.37-10	-	-	1.57-9
Intermediate Building	1.91-9	1.82-10	-	-	2.09-9
Auxiliary Building	1.89-9	1.81-10	-	-	2.07-9
Fuel Handling Building	1.41-9	1.35-10	-	-	1.54-9
Diesel Generator Building	1.43-9	1.37-10	-	-	1.57-9
Intake Screens and Pump House	1.61-9	1.54-10	-	-	1.76-9
Outdoor Tanks (BWST, CST)	1.07-10	1.02-11	-	-	1.17-10

NOTE: Exponential notation is indicated in abbreviated form; i.e., 2.71-10 = 2.71 x 10^{-10}.

7. AIRCRAFT CRASH ANALYSIS

7.1 INTRODUCTION AND SUMMARY

This section analyzes accident scenarios initiated by the crash of aircraft into the TMI-1 plant. This is done by estimating the frequency of crashes into various plant structures by different types of aircraft and evaluating the consequences of such crashes by developing core damage scenarios.

Typically, such analysis considers all aircraft activities that could pose a hazard to the plant, including aircraft flights to and from airports in the vicinity of the site and flights along the air routes that pass near the plant.

However, the TMI-1 site is located close to a major airport and the aircraft crash risk is dominated by operations at that airport. Therefore, the following analysis focuses on the hazard from the aircraft operations from the airports in the vicinity of the site.

There are two airports within 10 miles of the site, Harrisburg International Airport and Capital City Airport (formerly Harrisburg-York Airport). Harrisburg International is primarily used by commercial aircraft, while the Capital City traffic is mainly general aviation. There is some military aircraft activity at both airports.

Harrisburg International Airport is located on the north bank of the river northwest of the site and has only one runway (13/31). The TMI plant is approximately at a radius of 2.7 miles and 34° off the center line from the southwest end of the runway (Figure 7-1). The landing strip is called Runway 31 when used in the northwest direction and Runway 13 when used in the southeast direction. The threat to the TMI site is from operations at the south end of this strip; that is, from landings taking place in the northwest direction (Runway 31) and takeoffs in the southeast direction (Runway 13).

Capital City Airport is located about 8 miles west-northwest of the site and has two runways: Runway 12/30, which is approximately 4,000 feet long, and Runway 8/26, which is approximately 5,000 feet long. Only some of the landing and departure patterns bring the aircraft near the site. These are landings on Runway 30 and departures on Runway 12. An instrument landing approach to Runway 30 would bring the aircraft to about 0.5 miles of the site at about Elevation 2,300'. Departing aircraft on Runway 12 normally turn right approximately 1 to 3 miles from the end of the runway. Aircraft operations in the other directions are out of the site area.

The analysis is done for three different categories of aircraft:

- Heavy; i.e., large civilian and military aircraft with maximum takeoff or landing weight equal to or greater than 200,000 pounds.

- Moderate; i.e., large civilian and military aircraft that weigh less than 200,000 pounds.

- Small; i.e., single or multiple-engine small aircraft usually categorized as general aviation, air taxi, etc., with average weight of about 10,000 to 20,000 pounds.

The reason for the above classification is that each has different effects on the critical structures at TMI-1. In particular, the critical structures of the plant are designed to withstand crashes of aircraft having gross weights of up to 200,000 pounds during landing and takeoff operations (Reference 7-1).

Section 7.2 briefly describes the analytical model used for calculating the annual frequency of aircraft crashes into the plant. The analysis for the heavy aircraft category is reported in Section 7.3, followed by similar analyses for the moderate and small aircraft categories in Sections 7.4 and 7.5, respectively. Section 7.6 discusses the integrity of the critical structures. Core damage scenarios and the associated annual frequencies are described in Section 7.7. It is concluded that the total frequency of core damage initiated by aircraft crash is about 9.8×10^{-8} per year which is a negligible contributor to the total core damage frequency due to other scenarios.

7.2 ANALYTICAL MODEL

The frequency of aircraft crashes into the plant is estimated using the following model

$$f_j = \sum_{i=j}^{M} N_{ij} A_j C_{ij} S_{ij} \qquad (7.1)$$

where

f_j = annual frequency of impact for aircraft type j.

N_{ij} = annual number of operations of aircraft of type j to or from airport i or along airway i.

A_j = effective impact area of the plant for aircraft of type j (square miles).

C_{ij} = frequency of crash per operation of aircraft of type j operating along airway or from airport i.

S_{ij} = fraction of crashes in 1 square-mile area at the plant site.

The level of modeling and quantitative detail for the parameters defined above varies from one aircraft category to another; e.g., general aviation, heavy commercial, etc. More importantly, the consequences of crashes vary between categories such that combining frequencies of core damage for each category is more appropriate than combining frequencies of crashes from the different aircraft.

Core melt frequency, ϕ_{cm}, is, therefore, developed using the following equation

$$\phi_{cm} = \sum_{j=1}^{N} f_j p_j \qquad (7.2)$$

where p_j is the probability of core damage as the result of a crash of aircraft type j, and total core damage frequency, ϕ_{cm}, is the sum of the frequencies from individual aircraft categories..

The next section develops the crash frequency of heavy aircraft based on the approach and data of Reference 7-2.

7.3 HEAVY AIRCRAFT CRASH FREQUENCY

As discussed in Section 7.1, of the two airports in the vicinity of the site, only the Harrisburg International Airport involves operation of commercial, or heavy aircraft. Of particular interest in this analysis are aircraft with a maximum takeoff or landing weight of 200,000 pounds or more. This is due to the aircraft impact design criteria for certain critical structures of the TMI-1 plant (Reference 7-1). In this section, the annual frequency of crashes into the site by aircraft heavier than 200,000 pounds will be calculated. The analytical model used for this category of aircraft will be more detailed because of the higher consequences of crashes from these aircraft.

The heavy aircraft hit frequency into TMI-1 is calculated from

$$f_H = f_{SL} + f_{ST} + f_{NL} + f_{NT} \qquad (7.3)$$

where f_H is the annual frequency of aircraft crashes into TMI-1 by heavy aircraft using HIA, and f_{SL}, f_{ST}, f_{NL}, and f_{NT} are contributors to that frequency from scheduled landings, scheduled takeoffs, nonscheduled landings, and nonscheduled takeoffs. These frequencies are calculated based on Equation (7.1) as follows

$$f_{SL} = N_{SL}\, C_{SL}\, S_L\,(r,\theta)\, A_L \qquad (7.4)$$

$$f_{ST} = N_{ST}\, C_{ST}\, S_T\,(r,\theta)\, A_T \qquad (7.5)$$

$$f_{NL} = N_{NL}\, C_{NL}\, S_L\,(r,\theta)\, A_L \qquad (7.6)$$

$$f_{NT} = N_{NT}\, C_{NT}\, S_T\,(r,\theta)\, A_T \qquad (7.7)$$

where

N_{ST} and N_{NT} = the annual number of large scheduled and nonscheduled aircraft, respectively, taking off on TMI-1 end of the runway; i.e., using HIA runway 13.

N_{SL} and N_{NL}	=	the annual number of large scheduled and nonscheduled aircraft, respectively, landing on the TMI-1 end of the runway; i.e., using HIA runway 31.
A_L, A_T	=	the effective target area of the plant upon landing and takeoff, respectively.
C_{SL}, C_{NL}, C_{ST}, and C_{NT}	=	the applicable accident rate of scheduled landing, nonscheduled landing, scheduled takeoff, and nonscheduled takeoff.

and, finally,

$S_L(r,\theta)$ = frequency, per unit area, of the crash occurring at coordinates r,θ from end of runway, given that the crash is on landing.

$S_T(r,\theta)$ = frequency, per unit area, of the crash occurring at r,θ, given the crash is on takeoff.

A visual aid to understanding the physical meaning of these spatial distributions is provided in Figure 7-2. It is assumed that $S_L(r,\theta)$ and $S_T(r,\theta)$ are separable into radial and angular components.

More explicitly, let

$R_L(r) \equiv$ the fraction of landing crashes that occur at radius r or greater.

$\Theta_L(\theta) \equiv$ the fraction of landing crashes that occur at angle θ or greater.

Then,

$$S_L(r,\theta) = \left[\frac{d}{dr} R_L(r)\right]\left(\frac{360}{2\pi r}\right)\left[\frac{d}{d\theta}\Theta_L(\theta)\right]\left(\frac{1}{2}\right) \qquad (7.8)$$

where θ is measured in degrees, r in miles, and S_L in fraction per square mile.

Similarly, for takeoffs

$$S_T(r,\theta) = \left[\frac{d}{dr} R_T(r)\right]\left(\frac{360}{2\pi r}\right)\left[\frac{d}{d\theta}\Theta_T(\theta)\right]\left(\frac{1}{2}\right) \qquad (7.9)$$

The final 1/2 in these formulas corrects for the fact that in calculating the function, Θ, we will lump both positive and negative values of θ together--thus, in effect, treating all accidents as if they occurred on the TMI side of the runway.

The issue of separability of $S(r,\theta)$ has been discussed in Reference 7-3. The conclusion was that the assumption of separability does not introduce any significant error in terms assessing the spatial distribution.

In this analysis, following the method presented in Reference 7-4, uncertainty distributions are developed for all the frequencies using Bayesian techniques. The final results are presented for the total crash frequency as well as the frequency of crash for each of the four categories represented by Equations (7.4) through (7.7).

7.3.1 STATISTICAL INFORMATION FOR MODEL PARAMETERS

The data needed to quantify the various parameters of the model are presented in this section. These data include the number of aircraft movements at Harrisburg International Airport and the national aerial crash statistics.

7.3.1.1 Number of Movements of Heavy Aircraft at Harrisburg International Airport

In this analysis, we are concerned with the number of heavy aircraft movements; i.e., aircraft weighing 200,000 pounds or more. A conservative estimate puts the number of such operations at less than 1% of the total operations (Reference 7-5). For instance, based on the data presented in Table 7-1 for the year 1984, this number was estimated to be less than 1,411.

To estimate the number of movements of heavy aircraft in the scheduled and nonscheduled categories, we first observe that air taxi and general aviation aircraft, by definition, do not include heavy aircraft. The total number of movements, excluding these two categories for the year 1984, was 26,684 (see Table 7-1). A total of 8,549 of these operations was scheduled. Therefore, the fraction of scheduled operations is 0.32. The fraction of nonscheduled operations (including military) is then 0.68. Therefore, the breakdown of heavy aircraft movements based on these percentages is

Scheduled: $N_S = (0.32)(1,411) = 452$

Nonscheduled: $N_N = (0.68)(1,411) = 959$

The threat to the TMI site is from operations at the south end of the Runway 13/31; that is, from landings taking place in the northwest direction (Runway 31) and takeoffs in the southeast direction (Runway 13). Of the operations on this strip, 70% use Runway 31 and 30% use Runway 13. The number of landings and takeoffs are approximately equal on each runway. Thus, if N is the number of operations per year on the strip, then

$.35N$ = number of landings at south end $\equiv N_L$

$.15N$ = number of takeoffs at south end $\equiv N_T$

Based on the above data on the use of runways at the airport, we calculate the following values for the number of scheduled and nonscheduled landings and takeoffs in the TMI-1 direction of the runways.

$$N_{SL} = (0.35)N_S = 158$$

$$N_{ST} = (0.15)N_S = 68$$

$$N_{NL} = (0.35)N_N = 335$$

$$N_{NT} = (0.15)N_N = 144 \quad (7.10)$$

7.3.1.2 National Aerial Crash Statistics

Table 7-2 lists U.S. air carrier landing and takeoff accidents in the contiguous U.S. involving destruction of the aircraft for the years 1956 to 1982. The data for the years 1956 to 1977 were taken from Reference 7-6. The additional data for the years 1978 to 1982 were obtained from the National Transportation Safety Board computerized briefs of accidents and the detailed accident reports available from NTSB. Detailed reports for accidents beyond 1982 were not available at the time of this analysis. Table 7-2 also lists hit locations (r,θ) for each of the accidents and the phase and type of operation for the aircraft involved.

Tables 7-3 and 7-4 provide the number of takeoffs and landings for scheduled and nonscheduled operations for the period 1956 to 1982 (References 7-7 through 7-10). The takeoff and landing crash frequencies, plotted by year in Figure 7-3, show a downtrend in the accident frequencies.

Figure 7-4 is a plot of the radial distribution of crashes based on the data in Table 7-2. The angular distribution for takeoffs and landings is presented in the form of scatter diagrams in Figures 7-5 and 7-6, respectively.

7.3.1.3 Plant Target Area

The critical structures of the TMI-1 plant are designed to withstand crashes of aircraft having gross weights of up to 200,000 pounds during landing and takeoff operations. These structures are (see Figure 7-7):

- Reactor Building
- Fuel Handling Building
- Control Building
- Intake Screen House and Pump House
- Designated Portions of the Intermediate Building
- Designated Portions of the Auxiliary Building
- Heat Exchanger Vault
- Air Intake Structure (below ground)
- Access Tunnel Vault to Auxiliary Building

In calculating the target area for heavy aircraft, these and other structures of the plant were included and the following estimates for landing and takeoff hits were developed (Reference 7-1):

A_L = Landing Target Area = 0.0224 Square Mile

A_T = Takeoff Target Area = 0.0066 Square Mile

These areas were calculated by considering "shadow effect" to account for the dependence of the potential target area on the glide angle of the crashing aircraft and the "skid effect" to account for airplanes that might crash in front of the plant and slide into it. The calculated landing and takeoff target areas are based on glide angles of 10° and 45°, respectively.

The above values include the effective target area of both units to account for the fact that most of the critical structures of the two units are closely connected, so the crash of a large aircraft into the structures of one unit might have some impact on the structures of the other unit. This, of course, is a conservative assumption.

7.3.2 ASSESSMENT OF MODEL PARAMETERS

In this section, we will use the data presented in the previous section to estimate various components of the aircraft crash frequency model.

7.3.2.1 Prediction of Accident Rates from Historical Data

In this section, we develop an estimate of the aircraft accident rate, f, applicable to the plant in 1985 and beyond. Since, of course, we do not know the value of f exactly, we express our estimate in the form of a probability curve against f. The location and shape of this curve will then communicate our state of knowledge about the "true" value of f.

The historical data curve in Figure 7-3 shows, beginning in the early 1960s, a clear downward trend in accident rates reflecting, presumably, a steady improvement in aircraft equipment, flight safety technology, and safety consciousness.

A direct linear extrapolation of the curve to the years beyond 1982, however, would yield a crash rate very close to zero. A further extrapolation would go negative. Clearly, then, our extrapolation must reflect a leveling out of the curve. The approach followed in this study for extrapolating the crash frequency is based on Bayesian methods as described in the following:

1. We regard the historical data curve in Figure 7-3 as the result of sampling from an underlying population whose crash frequency is assumed to vary with time according to the functional form:

$$f(t) = a + (b-a)e^{-\lambda(t-t_0)} \qquad (7.11)$$

which reflects a gradual decrease and a leveling out at value a. In other words, we are saying that the "true" frequency in 1965, for example, is f(1965) as calculated from Equation (7.11). In that year, we selected (see Tables 7-3 and 7-4) a sample of 3,867 departures (7,734 operations) out of which we had a total of 4 accidents.

The parameter b controls the initial or starting value of f(t), λ defines its rate of decrease in time, and, finally, a determines its asymptotic behavior for large t.

2. In this form, Equation (7.11), we shall fix the year t_0, the starting point in time for the fit, and assign a value to b that would be the value of $f(t_0)$. We then determine or "fit" the remaining two parameters, a and λ, using Bayes' theorem. That is, we regard the data in Tables 7-3 and 7-4, the experience of the past, as evidence. On the basis of this evidence, we derive by Bayes' theorem a probability distribution on the space of a, λ pairs.

3. From this probability distribution of a, λ pairs, we shall derive a probability distribution for the crash frequency for any given year in the future. For instance,

   $$f(1985) = a + (b-a)^{-\lambda(1985 - t_0)} \tag{7.12}$$

   is the accident rate in 1985, given a, λ, and b. The probability distribution of f(1985) is found from the distribution of a, λ pairs.

To obtain the quantity in which we are interested; namely, the expected annual crash frequency over the remaining life of the plant, we calculate

$$\bar{f} = \frac{1}{31} \sum_{t=1985}^{t=2015} f(t) \tag{7.13}$$

The following provides the details of this "Bayesian Extrapolation" process.

Tables 7-3 and 7-4 give us for each year, t, a doublet (n_t, m_t) that tells the numbers of crashes and the number of operations in that year. Denote by B the set of such doublets from the year t_0 on:

$$B = \left\{(n_t, m_t)\right\}_{t=t_0}^{1982} \tag{7.14}$$

B, then, is the experience of the past. Next, we assume that the underlying frequency has the time dependence represented by Equation (7.11) with b and t_0 fixed from inspection of the data. We now ask: What can we say about the values of a, λ in light of the experience B?

For this purpose, we write Bayes' theorem in the form

$$p(a,\lambda|B) = p(a,\lambda)\left[\frac{p(B|a,\lambda)}{p(B)}\right] \tag{7.15}$$

where

$p(a,\lambda)$ = the probability we assign to the pair a,λ 'prior' to having information B.

$p(a,\lambda|B)$ = our probability of a,λ after having information B (the posterior).

$p(B|a,\lambda)$ = the probability of experiencing B, given the values a,λ.

$p(B)$ = the prior probability of B based on the knowledge represented by $p(a,\lambda)$.

$$p(B) = \iint_{a\ \lambda} p(a,\lambda)\ p(B|a,\lambda)\ da\ d\lambda. \tag{7.16}$$

To evaluate $p(B|a,\lambda)$, we note that each pair a, λ implies a specific function of time $f(t)$ through Equation (7.11). In any particular year, then, the probability of observing the pair (n_t, m_t) is

$$p(n_t,m_t|a,\lambda) = \binom{m_t}{n_t}[f(t)]^{n_t}\ [1-f(t)]^{m_t-n_t} \tag{7.17}$$

For the size m_t we are dealing with, the right side of Equation (7.17) may be replaced by

$$p(n_t,m_t|a,\lambda) = \frac{[m_t f(t)]^{n_t}}{n_t!}\ e^{-[m_t f(t)]} \tag{7.18}$$

The probability of experiencing the entire set B is then

$$p(B|a,\lambda) = \prod_{t=t_0}^{1982} \frac{[m_t f(t)]^{n_t}}{n_t!}\ e^{-[m_t f(t)]} \tag{7.19}$$

To carry out the process numerically, we established a discrete grid over the values of a and λ as follows:

$$a: \{a_1, a_2 \cdots , a_I\}$$

$$\lambda: \{\lambda_1, \lambda_2, \cdots , \lambda_j\}\ (\text{yrs}^{-1}) \tag{7.20}$$

We then chose a uniform prior over the set of discrete points (a_i, λ_j), saying thus that as far as our knowledge goes, each such pair is as likely as any other within the grid. With this choice, Equation (7.15) becomes

$$P_{ij} = p(a_i, \lambda_j | B) \quad \frac{p(B | a_i, \lambda_j)}{\sum_{i,j} p(B | a_i, \lambda_j)} \tag{7.21}$$

with the right side computed from Equation (7.19) using the f(t) given by Equation (7.11).

We now calculate the crash frequency for four different categories of aircraft operation, scheduled landings, nonscheduled landings, scheduled takeoffs, and nonscheduled takeoffs, and repeat the Bayesian analysis for each category. The historical data in Tables 7-3 and 7-4 are displayed graphically for each data category in Figures 7-8 through 7-11. The a, λ, and b values used for each category are as follows:

- Scheduled Landings

  $b = 1.0 \times 10^{-6}$ $t_o = 1955$

  $a = \{0.0, 0.05, 0.1, 0.2, 0.3, 0.4, 0.5, 0.6\}$ $(\times 10^{-6})$

  $\lambda = \left\{\frac{1}{5}, \frac{1}{6}, \frac{1}{7}, \ldots \frac{1}{25}\right\}$ (yrs^{-1})

- Nonscheduled Landings

  $b = 16 \times 10^{-6}$ $t_o = 1955$

  $a = \{0.0, 0.05, 0.1, 0.2, 0.3, 0.4, 0.5, 0.6\}$ $(\times 10^{-6})$

  $\lambda = \left\{\frac{1}{3}, \frac{1}{4}, \frac{1}{5}, \frac{1}{6}, \frac{1}{7}, \ldots \frac{1}{17}, \frac{1}{18}, \frac{1}{20}\right\}$ (yrs^{-1})

- Scheduled Takeoffs

  $b = 0.8 \times 10^{-6}$ $t_o = 1955$

  $a = \{0.0, 0.025, 0.05, 0.075, 0.1, 0.2\}$ $(\times 10^{-6})$

  $\lambda = \left\{\frac{1}{1.0}, \frac{1}{2.0}, \frac{1}{3.0}, \ldots \frac{1}{14}, \frac{1}{15}, \frac{1}{17}, \frac{1}{20}\right\}$ (yrs^{-1})

- Nonscheduled Takeoffs

$$b = 10 \times 10^{-6} \qquad t_0 = 1955$$

$$a = \{0.0,\ 1.0,\ 2.0,\ 30.0,\ 4.0,\ 5.0,\ 6.0\}\ (\times 10^{-6})$$

$$\lambda = \left\{\frac{1}{0.1}, \frac{1}{0.5}, \frac{1}{1.0}, \frac{1}{2.0}, \frac{1}{3.0}, \cdots \frac{1}{10}\right\}(\text{yrs}^{-1})$$

The resulting expected distributions for the predicted average crash frequency between 1985 and 2015 are displayed at the right of Figures 7-8 through 7-11. Each of these distributions was calculated by obtaining a distribution for the value of $f(t)$ for each value of t in the period 1985 through 2015, using the probability distribution on the a, λ grid and then obtaining the value of $\overline{f}$ based on Equation (7.13). The smooth curve on Figures 7-8 through 7-11 is a plot of Equation (7.11), using the mean values of a and λ from the discrete probability distribution for the period 1955 to 1982.

The mean annual crash rates for various cases are summarized as follows:

Scheduled Landings = 1.27×10^{-7} Crashes per Year

Nonscheduled Landings = 1.13×10^{-6} Crashes per Year

Scheduled Takeoffs = 4.57×10^{-8} Crashes per Year

Nonscheduled Takeoffs = 3.11×10^{-6} Crashes per Year

7.3.2.2 The Radial Density $\left[\frac{d}{dr} R(r)\right]_{r = r_0}$

The data shown in Figure 7-4 suggests that $R(r)$ may be well fit by a step at $r = 0$, followed by a decaying exponential, i.e.,

$$R(r) = \begin{cases} 1.0 & , r = 0 \\ ae^{-\lambda r}, & r > 0 \end{cases} \tag{7.22}$$

This being so, the derivative of $R(r)$ contains a delta function at $r = 0$

$$\left[\frac{-d}{dr} R(r)\right] = (1-a)\ \delta(r) + \lambda a e^{-\lambda r} \tag{7.23}$$

We seek to estimate the value of this derivative at the radius of the plant. Thus, we seek

$$D(r) = \left[\frac{-d}{dr} R(r)\right]_{r_0} = \lambda a e^{-\lambda r_0} \tag{7.24}$$

where r_o = 2.7 miles. We will obtain this estimate by first obtaining a discretized probability distribution on the space of doublets, (a,λ), and then converting this to a DPD against the desired derivative through Equation (7.24). To begin this process, we discretize the sets of possible a's and λ's

$$a:\{a_1, a_2, a_3 \ldots, a_I\} \tag{7.25}$$

$$\lambda:\{\lambda_1, \lambda_2, \lambda_3, \ldots, \lambda_J\} \tag{7.26}$$

We then consider the space of a, λ doublets

$$\{(a_i, \lambda_j)\} \tag{7.27}$$

On this space, we will establish a discrete probability distribution by assigning a probability, p_{ij}, to each such doublet, i.e.,

$$\{<p_{ij}, (a_i,\lambda_j)>\} \tag{7.28}$$

To explain the next step, let us introduce the notation

$$g(a,\lambda) = \lambda a e^{-\lambda r_o} \tag{7.29}$$

and

$$g_{ij} = g(a_i,\lambda_j) = \lambda_j a_i e^{-\lambda_j r_o} \tag{7.30}$$

Then, the DPD Equation (7.28) converts through Equation (7.30) to a DPD for g:

$$\{<p_{ij}, g_{ij}>\} \tag{7.31}$$

This is then the DPD for our desired derivative in Equation (7.24).

We obtain the DPD on (a,λ) space by applying Bayes' theorem in the form

$$p(a_i,\lambda_j|B) = p(a_i,\lambda_j) \frac{p(B|a_i,\lambda_j)}{\sum_{i,j} p(a_i,\lambda_j)\ p(B|a_i,\lambda_j)} \tag{7.32}$$

where

B = the information we get from our historical data.

$p(a_i,\lambda_j|B)$ = the probability we assign to the doublet (a_i,λ_j) after we have the information B.

$p(a_i,\lambda_j)$ = the probability we assign to the doublet (a_i,λ_j) prior to having the information B.

$p(B|a_i,\lambda_j)$ = the likelihood of event B happening, given that a_i,λ_j are true.

In our case, B is the set of radii at which crashes occurred.

In the case of landings, B is the set of radii at which landing crashes occurred. Thus, from Table 7-2 we have

$B = \{0, 0, 3.5, 0.8, 0.4, \ldots, \text{etc.}\}$

We note that B contains a total of 70 points, 27 points have $r = 0$, and the remainder have the sum,

$$\sum_{n=1}^{43} r_n = 73.8 \text{ miles} \tag{7.33}$$

Then, from Equation (7.23), the probability of these 70 crashes occurring as they did is

$$p(B|a_i,\lambda_j) = (1-a_i)^{27} (a_i\lambda_j)^{43} e^{\{-\lambda_j 73.8\}} \tag{7.34}$$

For this calculation, the following values are given for a_i and λ_j:

$$\{a_i\} \equiv \{0.4, .45, .5, .55, .6, .65, .7\} \tag{7.35}$$

$$\{\lambda_j\} \equiv \frac{1}{.75}, \frac{1}{1.0}, \frac{1}{1.25}, \ldots, \frac{1}{3.25} \tag{7.36}$$

The result is shown in Figure 7-12a. The Bayes' fit using the mean a, λ is shown as the straight line in Figure 7-13. The staircase function is the historical data.

In the case of takeoff crashes, B, from Table 7-2, is the set,

$B = \{0, 4.7, 0.9, 4.0, 3.1, 0.6, \ldots, \text{etc.}\}$

B for takeoff contains a total of 40 points, 18 having $r = 0$.

The remainder have the sum,

$$\sum_{n=1}^{22} r_n = 34.9 \text{ miles} \tag{7.37}$$

The probability of these 40 takeoff crashes occurring as they did is

$$p(B|a_i,\lambda_j) = (1-a_i)^{18} (a_i\lambda_j)^{22} e^{\{-\lambda_j 34.9\}} \tag{7.38}$$

The result for the takeoff calculation is shown in Figure 7-12b. Figure 7-14 compares the mean Bayes' fit with the historical takeoff crash data.

The mean value of the distribution of the radial density for various cases are summarized as follows:

Radial Density, Landings = 7.39×10^{-2}

Radial Density, Takeoffs = 6.40×10^{-2}

7.3.2.3 The Angular Density $\left[\frac{d}{d\theta}\Theta(\theta)\right]_{\theta = \theta_0}$

The same kind of reasoning can now be applied to determine the θ dependence, using as data only those crashes occurring at a radius of $\geq$ .5 mile. However, it is evident from Figure 7-15 that a simple exponential is not going to give a good fit to the angular data. Therefore, we need to modify the procedure used for the radial dependence. In doing this, we need to recognize that the important point is that the fit be good in the neighborhood of 34°, the location of the plant. At the same time, we wish to include the experience at the extremes (0° and 90°) of the θ range. Finally, if we can, we prefer to retain a fitting fraction with two parameters, rather than the complication of a three or four-parameter form.

The following approach appears to satisfy these requirements. We define $\Theta(\theta)$ as the fraction of crashes occurring at angle θ or more.

We then choose the form,

$$\Theta(\theta) = e^{-\lambda\theta} + b, \quad 0° \leq \theta \leq 70° \tag{7.39}$$

and use it to fit the data within the 0° to 70° range. Within this range, we may expect from Figure 7-15 that this form has the flexibility to give a good fit. Outside the range, of course, it cannot fit since it levels off, whereas the actual data go to zero. To blend in appropriately at $\theta = 70°$ and to account for the data of 90°, we choose $b = 0.098$.

We then use a Bayesian procedure to establish probability distributions on a, λ in the following way.

From Equation (7.39) we have the frequency density

$$\left[\frac{-d}{d\theta}\Theta(\theta)\right] = (1-a-b)\,\delta(\theta) + a\lambda e^{-\lambda\theta} \tag{7.40}$$

We now take B to be the set of crash points in the 0° to 70° range (and having $r \geq$ .5 mile). Thus, from Table 7-2,

$B = \{0, 01, 47, 61, 0, 26, 0, \ldots\}$

a total of 46 crashes with 19 at $\theta = 0$. Thus,

$$p(B|a,\lambda) = (1-a-b)^{19}(a\lambda)^{27}\ e^{-\lambda} \quad \sum_{1}^{27} \Theta_i \tag{7.41}$$

where

$$\sum_{1}^{27} \theta_i = 486 \qquad (7.42)$$

The resulting distribution for the desired derivative quantities is shown in Figure 7-16a. As a matter of interest, Figure 7-15 shows the goodness of fit using the mean a,λ, to the experimental data.

We now apply the analysis of the previous section to the landing and takeoff data separately. For landings, there is a total of 34 crashes; 15 at θ = 0 and 2 at 90°. We, therefore, set

$$b = \frac{2}{34} = .059 \qquad (7.43)$$

and summing over the points less than 90°, we have

$$\sum_{i=1}^{17} \theta_i = 230 \qquad (7.44)$$

The histogram for the desired derivative is plotted as Figure 7-16b. The Bayes' fit with average a,λ is plotted with the historical data in Figure 7-17.

For takeoffs, there is a total of 17 crashes; 4 at θ = 0 and 3 at 90°. In this case

$$b = \frac{3}{17} = .176 \qquad (7.45)$$

The sum of the angles in this case is

$$\sum_{i=1}^{10} \theta_i = 256 \qquad (7.46)$$

The results are given in Figures 7-16c and 7-18. The mean value of the distribution of the angular density for various cases is summarized as follows:

Angular Density, Landings = 3.31×10^{-3}

Angular Density, Takeoff = 5.75×10^{-3}

7.3.3 TOTAL IMPACT FREQUENCY

Using the values of the model parameter calculated in the previous section in Equations (7.4) through (7.7) results in the annual impact frequencies for the various aircraft operation categories as summarized in Table 7-5.

7.4 MODERATE WEIGHT AIRCRAFT CRASH FREQUENCY

This category of aircraft includes large civilian and military aircraft that weigh less than 200,000 pounds. Almost all such aircraft use Harrisburg International Airport, and only under very special conditions is the capital city airport used.

The crash frequency into TMI-1 for this category is also calculated from an equation similar to Equation (7.3), with the individual terms defined by Equations (7.4) through (7.7). Also, except for the number of operations in different categories, we will use the same values for the crash model parameters as those calculated in the previous section. The impact area in this case is only calculated for Unit 1 because it is not expected that a crash of an aircraft in this category into Unit 2 structures would also impact Unit 1 buildings. To obtain the number of operations in this category, we use the data of Table 7-1 for the year 1984. Subtracting the number of heavy aircraft in the scheduled and nonscheduled groups from the total in the corresponding categories, we get the number of operations for moderate weight aircraft:

$$N_S = 8{,}549 - 452 = 8{,}097$$

$$N_N = 26{,}684 - 959 = 25{,}725$$

Based on the discussion on the use of runways at Harrisburg International Airport, the following values for takeoffs and landings in the TMI-1 direction of the runways are calculated as

$$N_{SL} = 0.35\ N_S = 2{,}834$$

$$N_{ST} = 0.15\ N_S = 1{,}215$$

$$N_{NL} = 0.35\ N_N = 9{,}004$$

$$N_{NT} = 0.15\ N_N = 3{,}859$$

Table 7-6 summarizes the mean value calculations for the frequency of moderate aircraft crashes into the TMI-1 structures.

7.5 SMALL AIRCRAFT CRASH FREQUENCY

The crash frequency of small aircraft into various structures of the TMI-1 plant are calculated in this section. This category includes air taxi and general aviation aircraft operation from both the Harrisburg International Airport and Capital City Airport.

The basic model used is the same as that presented by Equation (7.1). The values of the model parameters are calculated as follows.

7.5.1 NUMBER OF OPERATIONS OF SMALL AIRCRAFT

Table 7-1 provides the number of air taxi and general aviation type of aircraft operations at the Harrisburg International Airport. The number

TABLE 8-3. NUMBER OF SHIPMENTS PER YEAR OF THE IMPORTANT HAZARDOUS CHEMICALS (n_i)

Chemical	Line	Shipments per Year
Acetic Acid	Shocks	79.3
	Roy	26
Acetic Anhydride	Shocks	34.7
	Roy	34.7
Acrylonitrile	Shocks	134.7
Ammonia, Anhydrous	Shocks	180
	Roy	46
Bromine	Shocks	47.3
Chlorine	Shocks	1,046
Chromic Fluoride	Roy	127.3
Coal Tar, Light Oil	Shocks	118.7
Ethyl Acrylate	Shocks	334.7
Ethylene Oxide	Shocks	236.7
Hydrochloric Acid	Shocks	117
Formaldehyde, 37% Weight	Shocks	50.7
Hydrofluoric Acid	Shocks	96
	Roy	42.7
Phosphorus Oxychloride	Shocks	41.3
Propylene Oxide	Shocks	236.7
Vinyl Acetate	Shocks	32
Vinyl Chloride	Shocks	2,888.7
	Roy	42

TABLE 8-4. ANNUAL FREQUENCY OF EXCEEDANCE OF TOXIC LIMITS IN CONTROL ROOM

Chemical	Roy	Shocks
Acetic Acid, Glacial	2.27×10^{-8}	1.82×10^{-8}
Acetic Anhydride	1.88×10^{-9}	0.00×10^{-0}
Acrylonitrile	--	1.58×10^{-7}
Ammonia, Anhydrous	9.62×10^{-8}	3.40×10^{-7}
Bromine	--	3.14×10^{-7}
Chlorine	--	6.54×10^{-6}
Chromic Fluoride	1.04×10^{-6}	--
Coal Tar, Light Oil	--	3.42×10^{-8}
Ethyl Acrylate	--	6.08×10^{-9}
Ethylene Oxide	--	4.44×10^{-7}
Formaldehyde	--	4.06×10^{-12}
Hexane	--	0.00×10^{-0}
Hydrochloric Acid	--	2.50×10^{-7}
Hydrofluoric Acid	3.49×10^{-7}	7.25×10^{-7}
Phosphorus Oxychloride	--	2.63×10^{-7}
Propylene Oxide	--	2.37×10^{-7}
Vinyl Acetate	--	8.76×10^{-8}
Vinyl Chloride	5.32×10^{-8}	2.22×10^{-6}

of landings and takeoffs for the year 1984 was 29,724 for air taxi and 84,693 for general aviation aircraft. Therefore, the total number of operations of small aircraft at Harrisburg International Airport is taken to be

$$N = 84{,}693 + 29{,}724 = 114{,}417 \text{ operations per year} \qquad (7.47)$$

The total number of operations of small aircraft at the Capital City Airport in 1985 was 71,733 (Reference 7-11). According to Reference 7-11, of the total hours flown by general aviation aircraft, nearly 76% involve single-engine and 24% involve multiple-engine aircraft. This breakdown is used to calculate the approximate number of operations for single and multiple-engine small aircraft:

Airport	Single-Engine	Multiple-Engine
Harrisburg	84,669	27,460
Capital City	53,082	17,216

7.5.2 CRASH RATES OF SMALL AIRCRAFT

The accident rates for general aviation aircraft are given in Table 7-7 (References 7-12 and 7-13). Note that these rates are given in units of accidents per mile flown and not per operation. The latter type is not available and must be estimated. This was done in this analysis by assuming an average 1-hour flight duration and an average speed of 100 mph and 250 mph for single and multiple-engine small aircraft, respectively. Furthermore, the annual crash statistics were used to develop uncertainty distributions for the crash rates by using the mean of the crash rates as the mean of the lognormal with a range factor of 5. This range factor is relatively high considering the annual variation of the crash rates and is judged to cover other uncertainties introduced in converting the rates from crashes per mile to crashes per operation. The following values characterize the resulting lognormal distributions:

CRASH RATES PER OPERATION

	Single-Engine	Multiple-Engine
5th Percentile	1.4×10^{-6}	1.1×10^{-6}
50th Percentile	7.1×10^{-6}	5.6×10^{-6}
95th Percentile	3.5×10^{-5}	2.8×10^{-5}
Mean	1.1×10^{-5}	9.0×10^{-6}

7.5.3 SPATIAL CRASH DENSITY

As discussed in Section 7.3, the distribution of crashes in the area surrounding an airport should normally include both radial and angular components. However, for small aircraft, the angular distribution of crash locations is not readily available from the published FAA and NTSB statistics. Furthermore, such detailed information is not needed in this case in which the adverse consequences of the crash of small aircraft are shown to have very small frequencies. Therefore, in this analysis, a uniform angular distribution was assumed.

The radial crash distribution remains to be found.

Table 7-8 shows the fraction of general aviation aircraft crashes for different radial distances, r, from the airport. It is based on 7 years of statistics provided by the FAA in the agency's annual review of aircraft accidents for calendar years 1972 through 1978. For areas with a radial distance, $r \geq 2$ miles, we fit the following exponential function,

$$D(r) = ae^{br} \qquad r \geq 2 \qquad \text{crash per mile}$$

with $a = 0.117$ and $b = 0.344$. $D(r)dr$ is the fraction of crashes in the area with a radial distance between r and $r + dr$ from the airport. Crash density per square mile at radius r is then given by

$$S(r) \approx \frac{D(r)}{2\pi r}$$

The uncertainty about $S(r)$ will be represented by a lognormal distribution with a median value estimated by the above formula and a range factor appropriate for each case.

For Harrisburg International Airport, we have

$$D(r)\Big|_{r = 2.7} = 0.117e^{-0.344 \times 2.7} \quad 4.6 \times 10^{-2} \quad \text{crash per mile}$$

This leads to a value of 2.7×10^{-3} crash per square mile as the median of the distribution of $S(r = 2.7)$. With a 95th to 50th percentile ratio of 2, we obtain the following characteristics of the (lognormal) distribution of S for Harrisburg International Airport:

5th Percentile:	1.4×10^{-3} Crash per Square Mile
95th Percentile:	5.4×10^{-3} Crash per Square Mile
Mean:	3.0×10^{-3} Crash per Square Mile

Similarly, we obtain a spatial crash distribution for Capital City Airport ($r = 8$). This time, we use a 95th to 50th percentile ratio equal to 4 to acknowledge the fact that we become more uncertain about the exponential fit to the data at long distances. The resulting lognormal distribution is the following:

5th Percentile: 3.7×10^{-5} Crash per Square Mile

50th Percentile: 1.5×10^{-4} Crash per Square Mile

95th Percentile: 5.0×10^{-4} Crash per Square Mile

Mean: 2.1×10^{-4} Crash per Square Mile

7.5.4 IMPACT AREA OF CRITICAL STRUCTURES

Due to the difference in the damage caused by the crash of small aircraft on concrete buildings compared with that for some other structures, such as the turbine building and metal tanks (BWST and CST), the crash frequency is calculated for each category of buildings separately. Therefore, effective areas were calculated for concrete buildings, turbine building, metal tanks, and unit transformers. These values are listed in Table 7-9 (same values are used for landing and takeoff operations).

7.5.5 TOTAL IMPACT FREQUENCY FOR SMALL AIRCRAFT

Table 7-9 summarizes the numerical results including the total impact frequency for various structures obtained by using the values of the model parameters obtained in the previous sections in Equation 7.3.

7.6 STRUCTURAL INTEGRITY EVALUATION

The impact frequencies calculated for heavy and moderate weight aircraft in Sections 7.3 and 7.4 are sufficiently low to not necessitate a structural fragility analysis of the target buildings. Based on the design criteria and as stated in Section 7.3, it is assumed that the conditional probability of substantial damage to critical structures is 1.0, given the crash of a heavy aircraft (more than 200,000 pounds). For other types of aircraft, the fragility of the TMI-1 concrete structures can be approximated by the values provided in Reference 7-14, as summarized in Table 7-10 for the perforation mode of damage.

Since the thickness of walls and roofs of TMI-1 concrete structures exceed 2 feet and all operations of moderate weight aircraft are to or from Harrisburg International, which is located less than 5 miles from the site, the probability of perforation is less than 0.28. Also, the numbers in Table 7-10 are based on all large aircraft, including those weighing in excess of 200,000 pounds. Therefore, the probability of perforation for moderate weight aircraft would be even smaller. The value for the collapse mode of failure is obviously smaller than the perforation probability because in this case a substantially higher momentum is typically required to completely destroy the structural integrity of the building. We assume a value of 0.1 and believe it is conservative. For small aircraft, no significant damage to concrete buildings is expected. Even if we use a conservative value of 0.01, which corresponds to the value for 2-foot thick walls and crashes beyond 5 miles (Table 7-10), the damage is expected to be localized with much smaller probabilities for affecting critical components. The conditional probability of the collapse mode of damage for this category of aircraft is vanishingly small (Reference 7-14).

7.7 FREQUENCY OF CORE DAMAGE

Core melt is assumed, given crash of a heavy aircraft. With this assumption, the contribution to the mean core damage frequency from heavy aircraft crash is

$$\phi_{cm}^{H} = f_H \cdot p_H$$

$$= (3.5 \times 10^{-8})(1.0) = 3.5 \times 10^{-8} \text{ per year}$$

For moderate weight aircraft, core damage is assumed for the collapse mode of damage, which has a conditional probability of 0.1. In the perforation mode of damage with a conditional probability of 0.28, the likelihood of damage to several critical components leading to core melt is small. (Reference 7-14 suggests a value of 0.01 for core damage conditional likelihood.) In this analysis, we will use 0.1, which is conservative. Therefore, for moderate weight aircraft, the mean core damage frequency is

$$\phi_{cm}^{M} = f_M \left[0.1 + (0.28)(0.1)\right]$$

$$= (4.62 \times 10^{-7})(1.28 \times 10^{-1}) = 5.91 \times 10^{-8} \text{ per year}$$

In the case of small aircraft, the likelihood of core melt, given damage from perforation of concrete buildings, is very small. We will use 0.01, as suggested in Reference 7-14, and believe it is conservative. The consequences of the crash of a small aircraft into the turbine building are much less severe even if it results in perforation or collapse of the building. The reason is that no safety functions would be impaired by such impact and, consequently, the conditional probability of core melt given the impact is negligible. Also, other scenarios, such as crash into BWST, CST, or transformers, are clearly dominated by similar scenarios involving random failure or unavailability of this equipment due to other causes. For instance, the frequency of damage to BWST due to aircraft crash is several orders of magnitude smaller than the rate of failure of the tank due to other causes (2.15×10^{-4} per year, see Data Analysis Report, Table 3-4). Therefore, the contribution of small aircraft crash to the mean core damage frequency is

$$\phi_{cm}^{S} = f_S \, (0.01)(0.01)$$

$$= (4.1 \times 10^{-5})(10^{-4}) = 4.1 \times 10^{-9} \text{ per year}$$

The total mean core damage frequency is therefore estimated as

$$\phi_{cm} = \phi_{cm}^{H} + \phi_{cm}^{M} + \phi_{cm}^{S}$$

$$= 9.8 \times 10^{-8} \text{ per year}$$

The above frequency is dominated by the frequency of other scenarios by several orders of magnitude. Therefore, no further analysis on the consequences of aircraft crash is needed.

7.8 REFERENCES

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7-8. Federal Aviation Administration, "FAA Statistical Handbook of Aviation," calendar years 1978-1983.

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7-11. Private Communication with Mr. R. Smith, Control Tower, Capital City Airport, 1985.

7-12. National Transportation Safety Board, "Annual Review of Aircraft Accident Rates, Calendar Year 1980," NTSB-ARG-80-1, May 1980.

7-13. Pickard, Lowe and Garrick, Inc., "Midland Probabilistic Risk Assessment," prepared for the Consumers Power Company, May 1984.

7-14. Chelapati, C. V., R. P. Kennedy, and I. B. Wall, "Probabilistic Assessment of Aircraft Hazard for Nuclear Power Plants," Nuclear Engineering and Design, Number 19, 1972.

TABLE 7-1. AIRCRAFT OPERATIONS AT HARRISBURG INTERNATIONAL AIRPORT (1980 - 1984)

Type of Operation	Total Number of Aircraft Movements (Takeoffs and Landings)				
	1980	1981	1982	1983	1984
Commercial, Scheduled	8,227	6,954	6,268	6,747	8,549
Commercial, Nonscheduled	1,422	356	690	233	157
Air Tax	23,010	20,135	22,752	22,437	29,724
Military	12,514	11,552	12,231	12,857	17,978
General Aviation	67,525	60,347	62,732	67,189	84,693
Total	112,698	99,344	104,673	109,463	141,101
Estimated Number of Heavy Aircraft Operations*	1,127	993	1,047	1,095	1,411

*Approximately 1% of the total number of aircraft movements.

TABLE 7-2. LISTING OF U. S. AIR CARRIER LANDING AND TAKEOFF ACCIDENTS IN THE CONTIGUOUS U. S., INVOLVING DESTRUCTION OF THE AIRCRAFT (1956 - 1982)

Sheet 1 of 8

Date	Location	Phase	Aircraft	Injury	Type Operation	Hit Location*	
						r (miles)	θ (degrees)
1956							
2/17	Owensboro, KY	L	M-404	0	SP	0	0
4/1	Pittsburg, PA	T	M-404	F	SP	0	0
4/2	Seattle, WA	T	B-377	F	SP	4.7	0
11/14	Las Vegas, NV	L	M-404	0	SP	0	0
1957							
1/6	Tulsa, OK	L	CV-240	F	SP	3.5	0
2/1	Rikers Island, NY	T	DC-6	F	SP	0.9	47
9/15	New Bedford, MA	L	DC-3	F	SP	0.8	6
1958							
2/13	Palm Springs, CA	T	CV-240	0	SP	4.0	0
3/25	Miami, FL	T	DC-7	F	SP	3.1	26
4/6	Freeland, MI	L	Viscount	F	SP	0.4	0
6/4	Martinsburg, WV	L	DC-3	F	Training	0.3	90
8/15	Nantucket, MA	L	CV-240	F	SP	0.3	22
8/28	Minneapolis, MN	T	DC-6	0	SP	0.6	0
11/10	New York, NY	T	L-1049	0	Training	0	0

NOTE: Footnotes and legend appear on the last sheet of this table.

TABLE 7-2 (continued)

Sheet 2 of 8

Date	Location	Phase	Aircraft	Injury	Type Operation	Hit Location*	
						r (miles)	θ (degrees)
1959							
2/3	New York, NY	L	L-188	F	SP	0.8	0
2/20	San Francisco, CA	L	DC-7	0	NS/C	0	0
3/15	Chicago, IL	L	CV-240	0	SC	1.2	28
5/12	Charleston, WV	L	L-1049	F	SP	0	0
8/15	Calverton, NY	L	B-707	F	Training	3.0	13
9/2	Abilene, TX	L	C-46	F	NS/C	0	0
11/24	Chicago, IL	L	L-1049	F	SC	0.2	0
12/1	Williamsport, PA	L	M-202	F	SP	1.4	90
10/26	Santa Maria, CA	T	DC-3	F	SP	1.5	NA
1960							
5/23	Atlanta, GA	T	CV-880	F	Training	0	0
9/14	New York, NY	L	L-188	0	SP	0	0
10/4	Boston, MA	T	L-188	F	SP	1.0	20
10/29	Toledo, OH	T	C-46	F	NS/P	1.1	4
1961							
7/11	Denver, CO	L	DC-8	F	SP	0	0
9/17	Chicago, IL	T	L-188	F	SP	0.8	90
11/8	Richmond, VA	L	L-1049	F	NS/P	1.1	26

NOTE: Footnotes and legend appear on the last sheet of this table.

TABLE 7-2 (continued)

Sheet 3 of 8

Date	Location	Phase	Aircraft	Injury	Type Operation	Hit Location*	
						r (miles)	θ (degrees)
1962							
3/1	Jamaica Bay, NY	T	B-707	F	SP	2.7	90
4/18	Dallas, TX	T	DC-3	F	Test	0	0
7/8	Amarillo, TX	T	V-812	0	SP	1.2	21
8/22	Wilmington, NC	L	M-404	0	Training	0	0
11/30	New York, NY	L	DC-7	F	SP	0.8	9
12/14	Hollywood, CA	L	L-1049	F	SC	1.5	0
12/21	Grand Island, NE	L	CV-340	0	SP	0.8	0
1963							
1/29	Kansas City, MO	L	V-812	F	SP	0	0
2/3	San Francisco, CA	L	L-1049	F	SC	0	0
2/16	Puyallup, WA	L	C-46	0	NS/C	0.5	0
5/28	Manhattan, KS	L	L-1049	0	NS/P	0.1	0
7/2	Rochester, NY	T	M-404	F	SP	0	0
11/29	Morgantown, WV	L	DC-3	F	Ferry	2.5	18
1964							
3/10	Boston, MA	L	DC-4	F	SC	1.3	0
3/12	Miles City, MT	L	DC-3	F	SP	1.9	0
11/20	Detroit, MI	T	C-46	0	NS/C	0.4	0
12/24	San Francisco, CA	T	L-1049	F	SC	4.3	31
12/30	Detroit, MI	L	C-46	F	NS/C	2.3	13

NOTE: Footnotes and legend appear on the last sheet of this table.

TABLE 7-2 (continued)

Sheet 4 of 8

Date	Location	Phase	Aircraft	Injury	Type Operation	Hit Location*	
						r (miles)	θ (degrees)
1965							
4/16	Las Vegas, NV	T	F-27	0	Training	0	0
5/18	Knob Knoster, MO	L	DC-6	0	NS/C	0.8	10
7/23	Montoursville, PA	T	CV-440	0	SP	2.8	45
9/13	Kansas City, MO	T	CV-880	0	Training	0.2	27
11/8	Constance, KY	L	B-727	F	SP	2.0	0
11/11	Salt Lake City, UT	L	B-727	F	SP	0.1	0
1966							
3/21	Norfolk, VA	L	CL-44	0	SC	0	0
4/22	Ardmore, OK	L	L-188	F	NS/P	2.3	90
7/28	Newark, NJ	T	C-46	0	NS/C	1.1	90
11/20	New Bern, NC	L	M-404	F	SP	4.0	9
1967							
1/31	San Antonio, TX	L	DC-6	F	NS/C	4.5	0
3/30	Kenner, LA	L	DC-8	F	Training	0.4	27
11/6	Erlanger, KY	T	B-707	F	SP	0	0
11/20	Constance, KY	L	CV-880	F	SP	1.8	3
12/21	Denver, CO	T	DC-3	F	NS/C	0	0

NOTE: Footnotes and legend appear on the last sheet of this table.

TABLE 7-2 (continued)

Sheet 5 of 8

Date	Location	Phase	Aircraft	Injury	Type Operation	Hit Location*	
						r (miles)	θ (degrees)
1968							
1/1	Oxford, MS	L	M-404	0	Ferry	0	0
3/21	Chicago, IL	T	B-727	0	SC	0	0
4/28	Atlantic City, NJ	L	DC-8	0	Training	0	0
8/10	Charleston, WV	L	F-227	F	SP	0	0
9/27	Cherry Point, NC	L	DC-7	0	NS/C	0.4	17
12/24	Bradford, PA	L	CV-580	F	SP	2.8	8
12/27	Sioux City, IA	T	DC-9	0	SP	0	0
12/27	Chicago, IL	L	CV-580	F	SP	0.3	86
1969							
1/6	Bradford, PA	L	CV-440	F	SP	5.0	0
7/15	Jamaica, NY	T	DHC-6	F	SP	0	0
7/26	Pomona, NJ	L	B-707	F	Training	0	0
10/11	Stockton, CA	T	DC-8	0	Training	0	0
1970							
8/24	Hill AFB, UT	T	L-188	0	NS/C	0	0
9/8	Jamaica, NY	T	DC-8	F	Ferry	0	0
10/10	Wrightstown, NJ	L	GA-382	F	NS/C	1.0	0
11/14	Huntington, WV	L	DC-9	F	NS/P	1.1	0

NOTE: Footnotes and legend appear on the last sheet of this table.

TABLE 7-2 (continued)

Sheet 6 of 8

Date	Location	Phase	Aircraft	Injury	Type Operation	Hit Location*	
						r (miles)	θ (degrees)
1971							
3/31	Ontario, CA	L	B-720	F	Training	0	0
6/7	New Haven, CN	L	CV-580	F	SP	0.9	6
1972							
3/3	Albany, NY	L	F-227	F	SP	3.8	0
5/18	Ft. Lauderdale, FL	L	DC-9	0	SP	0	0
5/30	Ft. Worth, TX	L	DC-9	F	Training	0	0
12/8	Chicago, IL	L	B-737	F	SP	1.8	10
12/20	Chicago, IL	T	DC-9	F	SP	0	0
1973							
7/23	St. Louis, MO	L	F-227	F	SP	2.6	4
7/31	Boston, MA	L	DC-9	F	SP	0.6	4
11/3	Boston, MA	L	B-707	F	SC	0	0
11/27	Akron, OH	L	DC-9	0	SP	0	0
1974							
1/16	Los Angeles, CA	L	B-707	0	SP	0	0
9/11	Charlotte, NC	L	DC-9	F	SP	3.4	0

NOTE: Footnotes and legend appear on the last sheet of this table.

TABLE 7-2 (continued)

Sheet 7 of 8

Date	Location	Phase	Aircraft	Injury	Type Operation	Hit Location*	
						r (miles)	θ (degrees)
1975							
6/24	Jamaica, NY	L	B-727	F	SP	0	0
11/12	Jamaica, NY	T	DC-10	0	NS/P	0	0
1976							
2/8	Van Nuys, CA	T	DC-6	F	Ferry	1.5	0
6/23	Philadelphia, PA	L	DC-9	0	SP	0	0
1977							
7/6	St. Louis, MO	T	L-188	F	NS/C	0	0
1978							
03/1	Los Angeles, CA	T	DC-10	F	SP	0.1	0
9/25	San Diego, CA	L	B-727	F	SP	3.5	28
1979							
2/9	Miami, FL	T	DC-9	0	Training	0.15	30
1/5	Amiat, AK	L	188A	0	NS/CTR	0	0
5/25	Chicago, IL	T	DC-10	F	SP	0.87	17
6/22	Daggett, CA	T	DC-7	F	M	1.0	20
5/15	Mesa, AZ	T	C-54D	0	Test	0**	0
11/19	McCormick, SC	L	C-54D	F	M	2.5	35

NOTE: Footnotes and legend appear on the last sheet of this table.

TABLE 7-2 (continued)

Sheet 8 of 8

Date	Location	Phase	Aircraft	Injury	Type Operation	Hit Location*	
						r (miles)	θ (degrees)
1980							
6/19	Atlanta, GA	L	SUD AVN SE-210	0	Cargo Service	0	0
11/28	Pecos, TX	T	DC-7B	F	M	†	†
6/22	Columbus, IN	T	1049-H	F	Ferry	0.87	25
1981							
2/17	Santa Ana, CA	L	B-737	F	SP	0	0
1982							
3/13	Glendale, AZ	L	KC-135A	F	Military	3.5	0
1/23	Boston, MA	L	DC-10-30	F	SP	0	0

*Hit location: r = radial distance of the hit to the end of the runway in use. θ is the angle to the runway centerline. r = 0 is considered if the hit occurred within 0.05 mile of the runway, and θ = 0 is considered if the hit occurred within 200 feet of the extended runway center-line. Note that we do not distinguish between a positive or negative angle (θ).

**This plane ran off the runway after aborted takeoff. The radial distance would be 0.25 mile (1,300 feet) if final resting place is considered.

†Sufficient information unavailable to determine r or θ.

LEGEND:

Phase: L = landing; T = takeoff.
Injury: F = one or more occupant fatalities; 0 = none.
Type operation: SC = scheduled cargo; SP = scheduled passenger; NS/C = nonscheduled cargo; NS/CTR = nonscheduled charter; NS/P = nonscheduled passenger; M = smuggling.

TABLE 7-3. U.S. AIR CARRIER ACCIDENT RATE FOR SCHEDULED AND NONSCHEDULED LANDINGS IN THE CONTIGUOUS U.S.*

Year	Scheduled			Nonscheduled			Total Landings		
	Operations (10^3)	Accidents	Accident Rate** (10^{-6})	Operations (10^3)	Accidents	Accident Rate** (10^{-6})	Operations (10^3)	Accidents	Accident Rate** (10^{-6})
1956	3,188	2	.627	90	0	0	3,278	2	.610
1957	3,444	2	.581	90	0	0	3,534	2	.566
1958	3,302	2	.606	90	0	0	3,392	2	.590
1959	3,551	5	1.406	90	2	22.2	3,641	7	1.92
1960	3,501	1	.286	125	0	0	3,626	1	.276
1961	3,400	1	.294	140	1	7.14	3,540	2	.565
1962	3,303	3	.908	175	0	0	3,478	3	.863
1963	3,414	2	.586	155	2	12.9	3,569	4	1.12
1964	3,554	2	.563	95	1	10.5	3,649	3	.822
1965	3,772	2	.530	95	1	10.5	3,867	3	.776
1966	3,926	2	.509	85	1	11.8	4,011	3	.748
1967	4,478	1	.223	90	1	11.8	4,568	2	.438
1968	4,836	3	.620	105	1	9.52	4,941	4	.810
1969	4,934	1	.203	115	0	0	5,049	1	.198
1970	4,669	0	0	125	2	16.0	4,794	2	.417
1971	4,558	1	.219	155	0	0	4,713	1	.212
1972	4,601	3	.652	135	0	0	4,736	3	.633
1973	4,651	4	.860	130	0	0	4,781	4	.837
1974	4,275	2	.468	105	0	0	4,380	2	.457
1975	4,269	1	.234	110	0	0	4,379	1	.228
1976	4,411	1	.227	115	0	0	4,526	1	.221
1977	4,560	0	0	125	0	0	4,685	0	0
1978	4,608	1	.217	116	0	0	4,724	1	.212
1979	4,852	0	0	122	2	16.4	4,974	2	.402
1980	4,892	0	0	123	1	8.13	5,015	1	.199
1981	4,664	1	.214	110	0	0	4,774	1	.209
1982	4,455	1	.224	114	1	8.77	4,569	2	.438

*Destruct accidents on or off runway but within 5 miles.

**Accidents per landing.

TABLE 7-4. U.S. AIR CARRIER ACCIDENT RATE FOR SCHEDULED AND NONSCHEDULED TAKEOFFS IN THE CONTIGUOUS U.S.*

Year	Scheduled			Nonscheduled			Total Takeoffs		
	Operations (10^3)	Accidents	Accident Rate** (10^{-6})	Operations (10^3)	Accidents	Accident Rate** (10^{-6})	Operations (10^3)	Accidents	Accident Rate** (10^{-6})
1956	3,188	2	.627	90	0	0	3,278	2	.610
1957	3,444	1	.290	90	0	0	3,534	1	.283
1958	3,302	3	.909	90	0	0	3,392	3	.884
1959	3,551	1	.281	90	0	0	3,641	1	.275
1960	3,501	1	.286	125	1	8.00	3,626	2	.552
1961	3,400	1	.294	140	0	0	3,540	1	.282
1962	3,303	2	.606	175	0	0	3,478	2	.575
1963	3,414	1	.293	155	0	0	3,569	1	.280
1964	3,554	1	.281	95	1	10.5	3,649	2	.548
1965	3,772	1	.265	95	0	0	3,867	1	.259
1966	3,926	0	.0	85	1	11.8	4,011	1	.249
1967	4,478	1	.223	90	1	11.1	4,568	2	.438
1968	4,836	2	.414	105	0	0	4,941	2	.405
1969	4,934	1	.203	115	0	0	5,049	1	.198
1970	4,669	0	0	125	1	8.0	4,794	1	.209
1971	4,558	0	0	155	0	0	4,713	0	0
1972	4,601	1	.217	135	0	0	4,736	1	.211
1973	4,651	0	0	130	0	0	4,781	0	0
1974	4,275	0	0	105	0	0	4,380	0	0
1975	4,269	0	0	110	1	9.09	4,379	1	.228
1976	4,411	0	0	115	0	0	4,526	0	0
1977	4,560	0	0	125	1	8.00	4,685	1	.213
1978	4,608	1	.217	116	0	0	4,724	1	.212
1979	4,852	1	.206	122	1	8.20	4,974	2	.402
1980	4,892	0	0	123	1	8.13	5,015	1	.199
1981	4,664	0	0	110	0	0	4,774	0	0
1982	4,455	0	0	114	0	0	4,569	0	0

*Destruct accidents on or off runway but within 5 miles.

**Accidents per takeoff.

TABLE 7-5. MEAN ANNUAL HIT FREQUENCY RESULTS FOR VARIOUS TYPES AND MODES OF HEAVY AIRCRAFT OPERATION (10^{-9} CRASHES PER YEAR)

Type of Operation	Mode of Operation		Total
	Landing	Takeoff	
Scheduled	1.20	0.08	1.28
Nonscheduled	22.3	11.5	33.8
Total	23.5	11.6	35.1

TABLE 7-6. MEAN VALUES OF MODEL PARAMETERS AND ANNUAL IMPACT FREQUENCY FOR MODERATE WEIGHT AIRCRAFT

Type of Operation	Number of Operations	Crash Rate	Spatial Distribution		Impact Area	Impact Frequency
			Radial	Angular		
Scheduled Landing	2,834	1.27-7	7.39-2	3.31-3	0.0112	1.05-8
Scheduled Takeoff	1,215	4.57-8	6.40-2	5.75-3	0.0033	7.15-10
Nonscheduled Landing	9,004	1.13-6	7.39-2	3.31-3	0.0112	2.96-7
Nonscheduled Takeoff	3,859	3.11-6	6.40-2	5.75-3	0.0033	1.55-7
					Total	4.62-7

NOTE: Exponential notation is indicated in abbreviated form; i.e., 1.27-7 = 1.27 x 10^{-7}.

TABLE 7-7. FATAL ACCIDENT RATES FOR U.S. GENERAL AVIATION AIRCRAFT

Year	Fatal Accident Rates Per Miles Flown		
	Single Engine	Multiple Engine	All Types
1972	2.63-7	8.7-8	2.11-7
1973	2.52-7	8.2-8	2.09-7
1974	2.45-7	7.6-8	1.88-7
1975	2.30-7	6.9-8	1.71-7
1976	2.02-7	6.4-8	1.66-7
1977	2.03-7	5.1-8	1.59-7
1978	2.02-7	7.7-8	1.59-7

NOTE: Exponential notation is indicated in abbreviated form; i.e., 2.63-7 = 2.63×10^{-7}.

TABLE 7-8. FRACTION OF GENERAL AVIATION AIRCRAFT CRASHES AS A FUNCTION OF DISTANCE FROM THE AIRPORT

Distance (miles)	Fraction (percent)
On the Airport	16.61
1/4	5.17
1/2	4.03
3/4	1.32
1	3.54
2	5.90
3	4.25
4	2.73
5	2.08
> 5	54.37

TABLE 7-9. MEAN ANNUAL IMPACT FREQUENCY FOR VARIOUS TYPES OF SMALL AIRCRAFT

Structure	Area Miles^2	Airport	Aircraft Type	Number of Operations	Spatial Distribution	Crash Rate	Impact Frequency	Total
Concrete Structures	1.1-2	Harrisburg	Single Engine	84,669	3.0-3	1.1-5	3.1-5	
			Multiple Engine	27,460		9.0-6	8.3-6	
		Capital City	Single Engine	53,082	2.1-4	1.1-5	1.4-6	
			Multiple Engine	17,216		9.0-6	3.6-7	4.1-5
Turbine Building	3.8-3	Harrisburg	Single Engine	84,669	3.0-3	1.1-5	1.1-5	
			Multiple Engine	27,460		9.0-6	2.8-6	
		Capital City	Single Engine	53,082	2.1-4	1.1-5	4.7-7	
			Multiple Engine	17,216		9.0-5	1.2-7	1.4-5
BWST and CST	1.2-4	Harrisburg	Single Engine	84,669	3.0-3	1.1-5	3.4-7	
			Multiple Engine	27,460		9.0-6	8.9-8	
		Captial City	Single Engine	53,082	2.1-4	1.1-5	1.5-8	
			Multiple Engine	17,216		9.0-6	3.9-9	4.5-7
Unit Transformers	3.9-5	Harrisburg	Single Engine	84,669	3.0-3	1.1-5	1.1-7	
			Multiple Engine	27,460		9.0-6	2.9-8	
		Captial City	Single Engine	53,082	2.1-4	1.1-5	4.8-9	
			Multiple Engine	17,216		9.0-6	1.3-9	1.5-7

NOTE: Exponential notation is indicated in abbreviated form; i.e., 3.0-3 = 3.0 x 10^{-3}.

TABLE 7-10. CONDITIONAL PROBABILITY OF PERFORATION MODE OF DAMAGE TO CONCRETE STRUCTURES

Distance from Airport	Aircraft Type	Wall Thickness			
		1 Foot	1.5 Feet	2 Feet	6 Feet
Within 5 Miles	Small	0.003	0	0	0
	Large	0.96	0.52	0.28	0
Beyond 5 Miles	Small	0.28	0.06	0.01	0
	Large	1.00	1.00	0.84	0.32

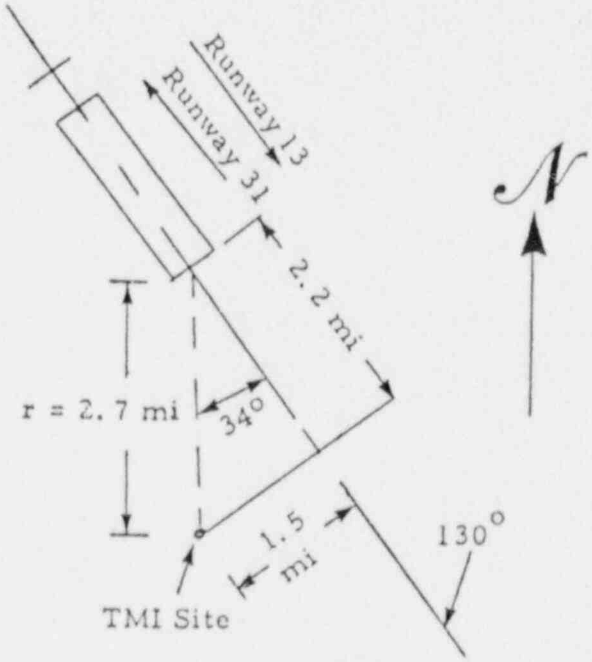

FIGURE 7-1. LOCATION OF TMI SITE WITH RESPECT TO HARRISBURG AIRPORT

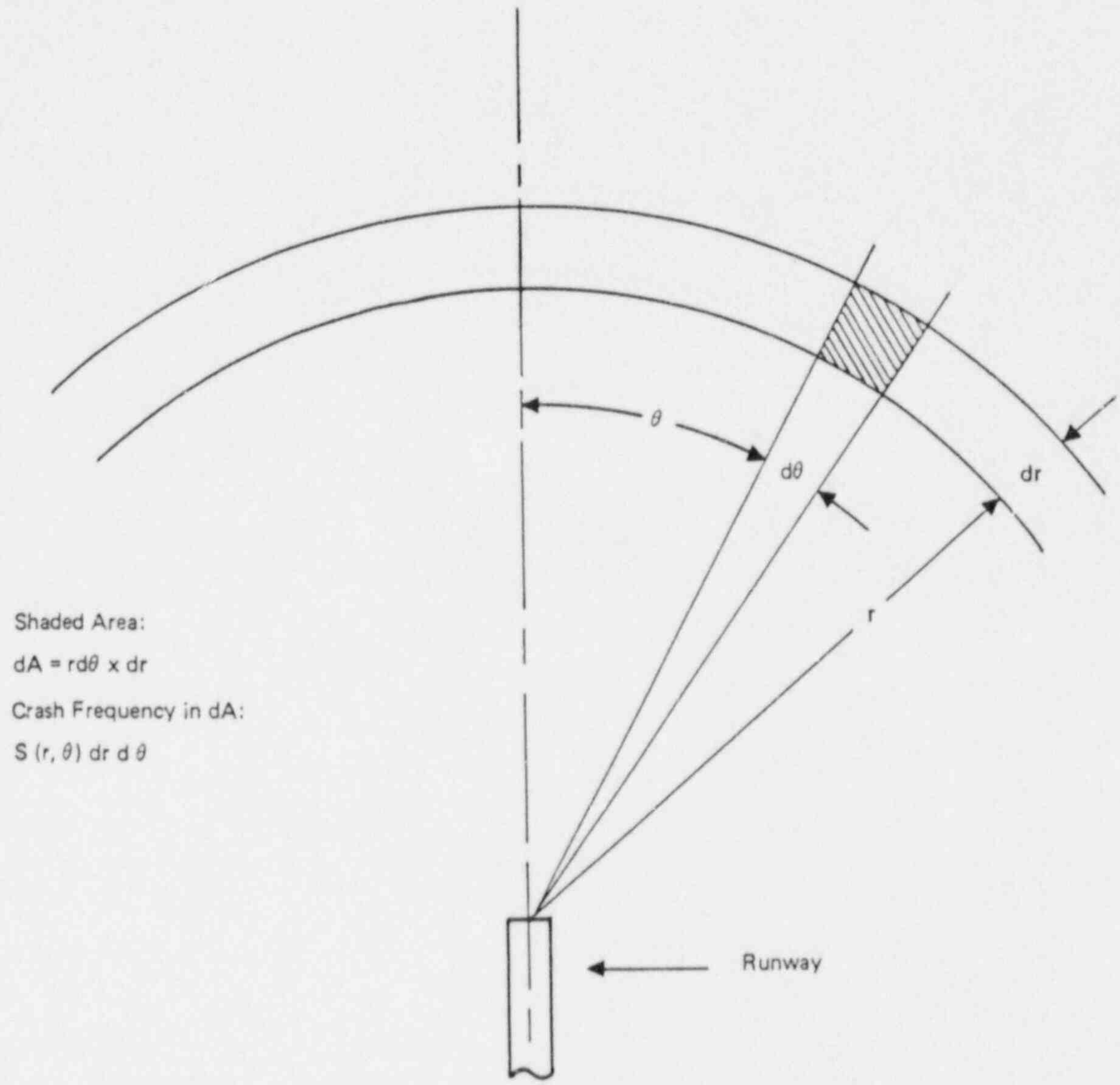

FIGURE 7-2. REPRESENTATION OF SPATIAL CRASH FREQUENCY DISTRIBUTION

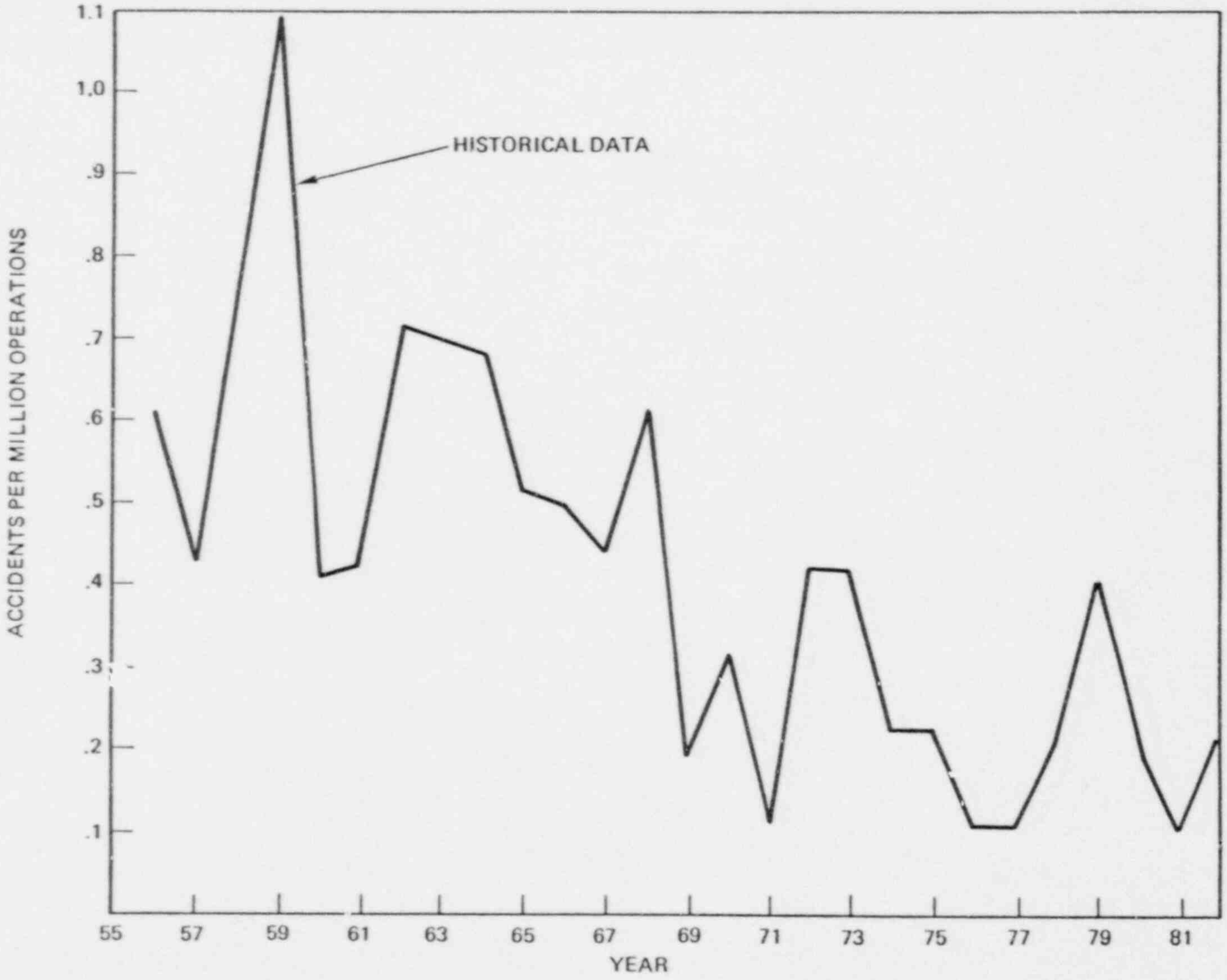

FIGURE 7-3. HISTORICAL ACCIDENT RATE VERSUS TIME - LANDINGS AND TAKEOFFS COMBINED

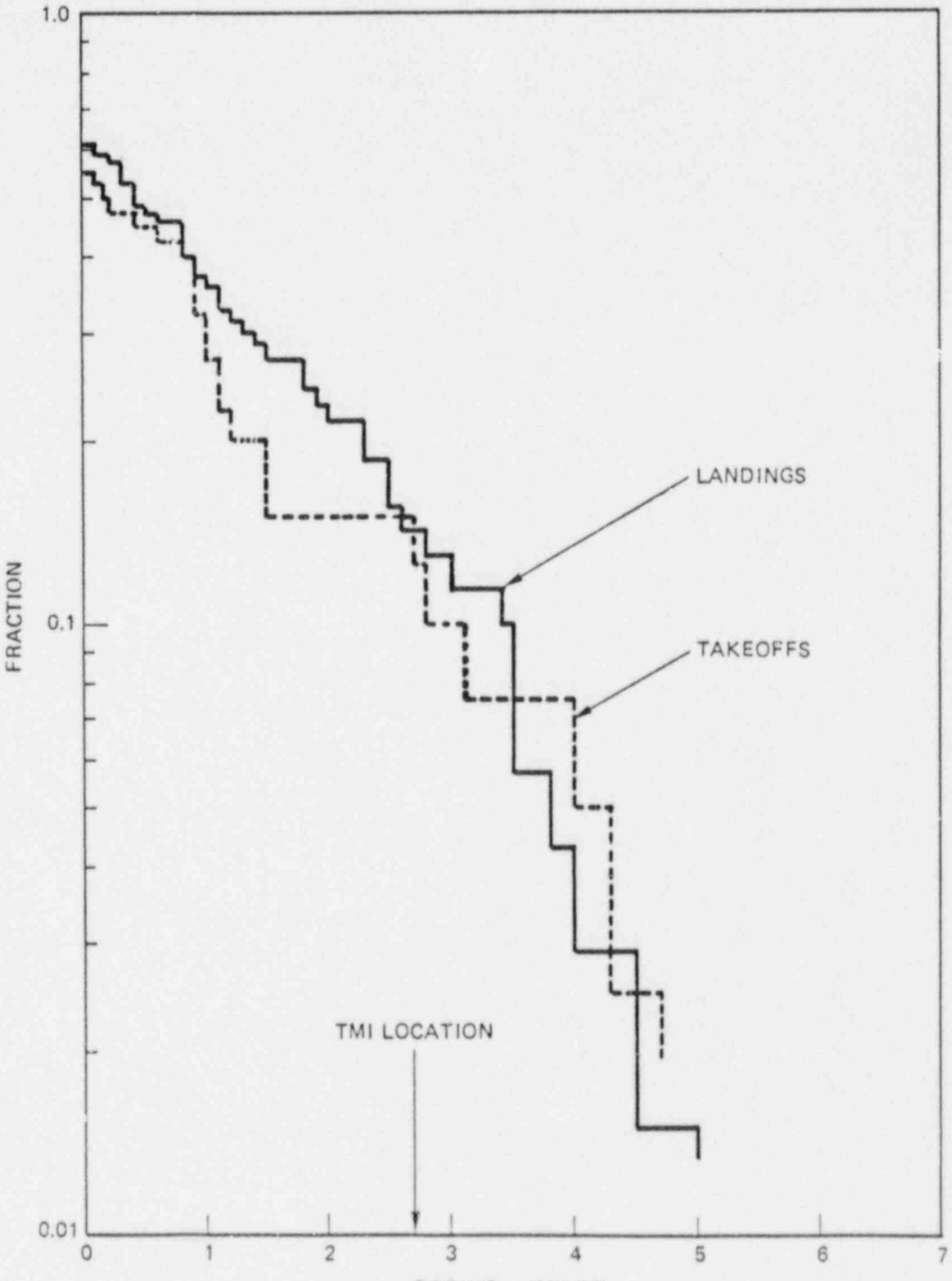

FIGURE 7-4. FRACTION OF CRASHES OCCURRING AT RADIUS r OR GREATER

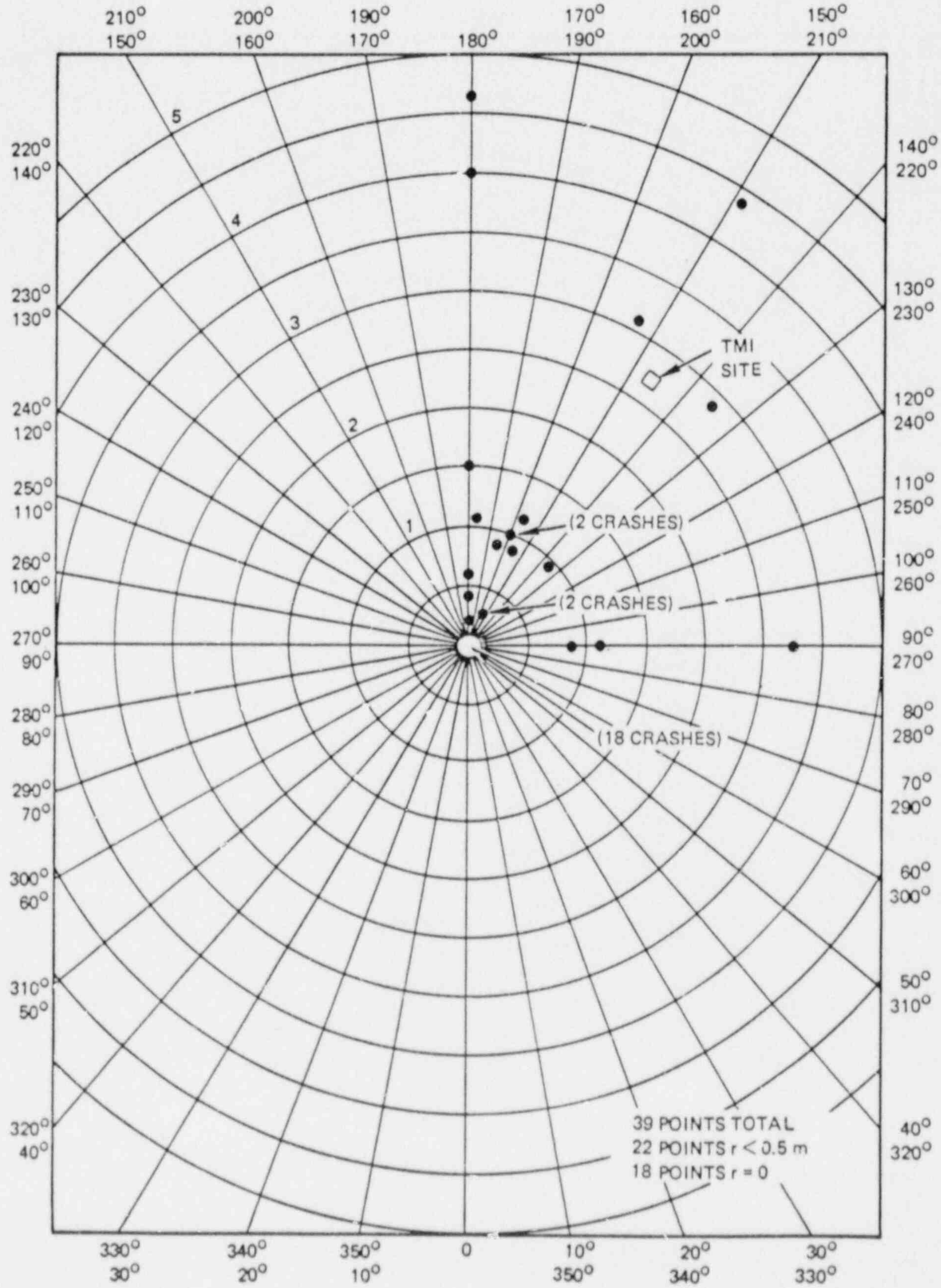

FIGURE 7-5. SCATTER PATTERN FOR TAKEOFF ACCIDENTS (Radius in Miles)

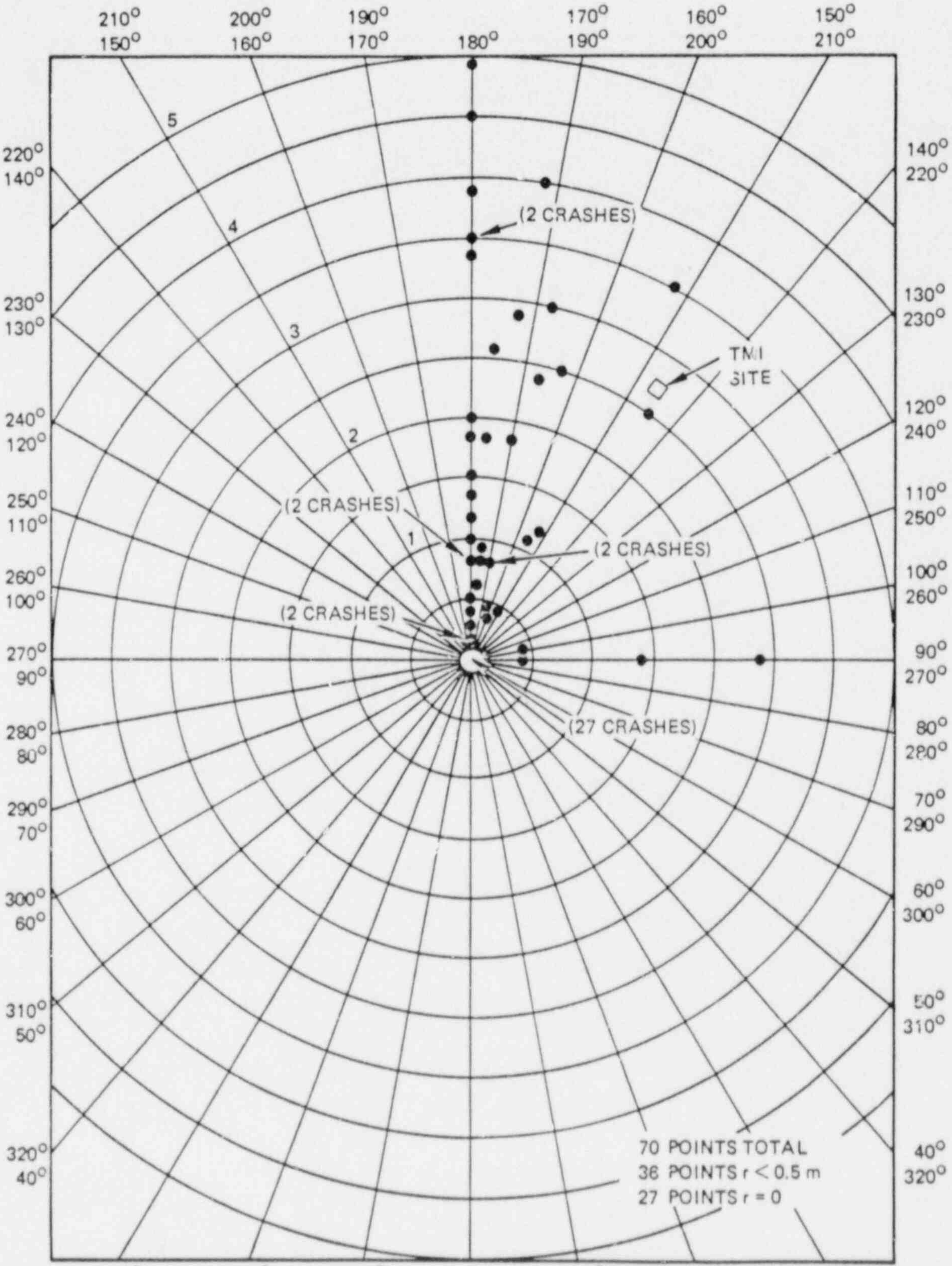

FIGURE 7-6. SCATTER PATTERN FOR LANDING ACCIDENTS (Radius in Miles)

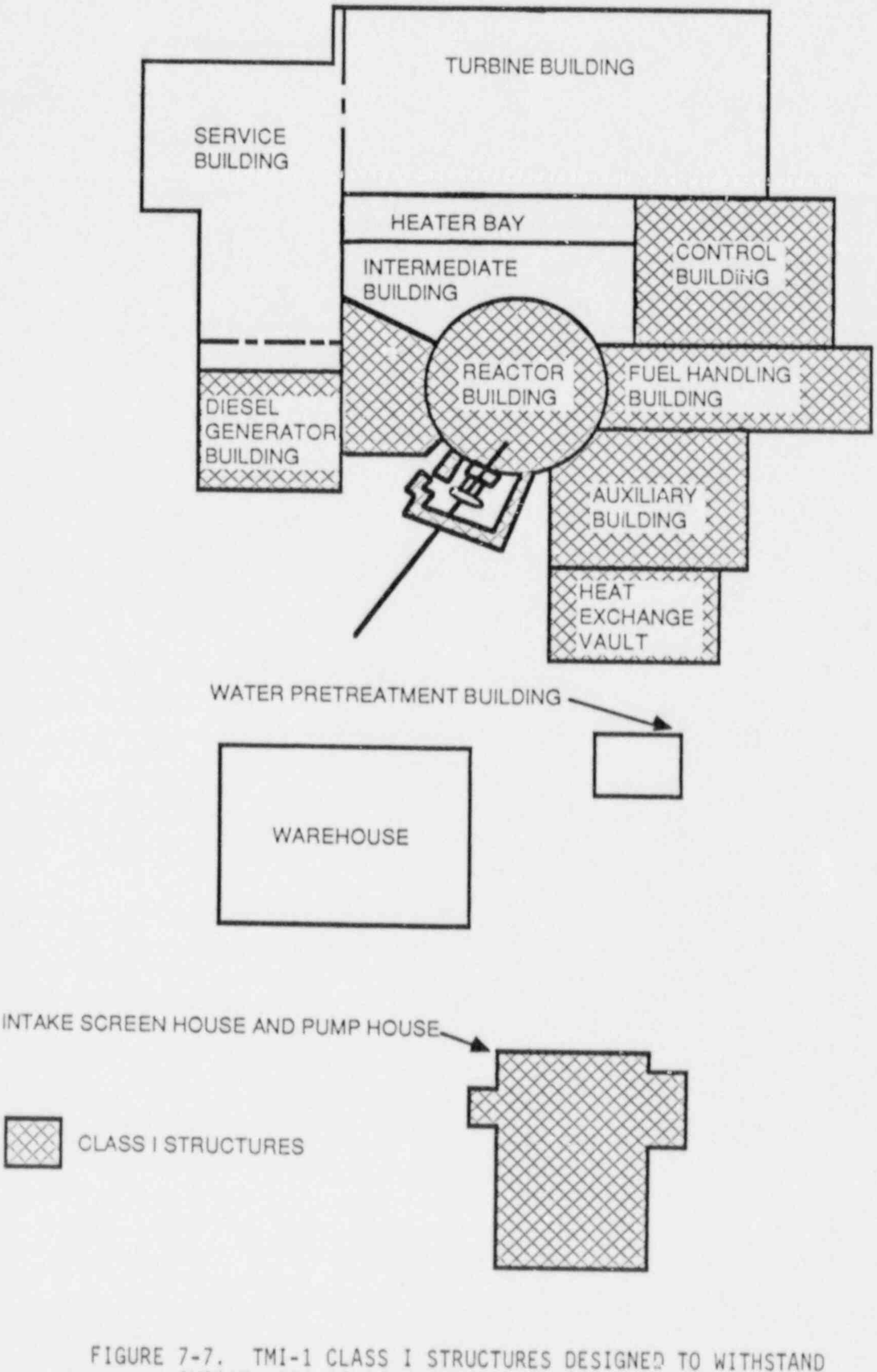

FIGURE 7-7. TMI-1 CLASS I STRUCTURES DESIGNED TO WITHSTAND IMPACT LOAD OF AIRCRAFT UP TO 200,000 POUNDS

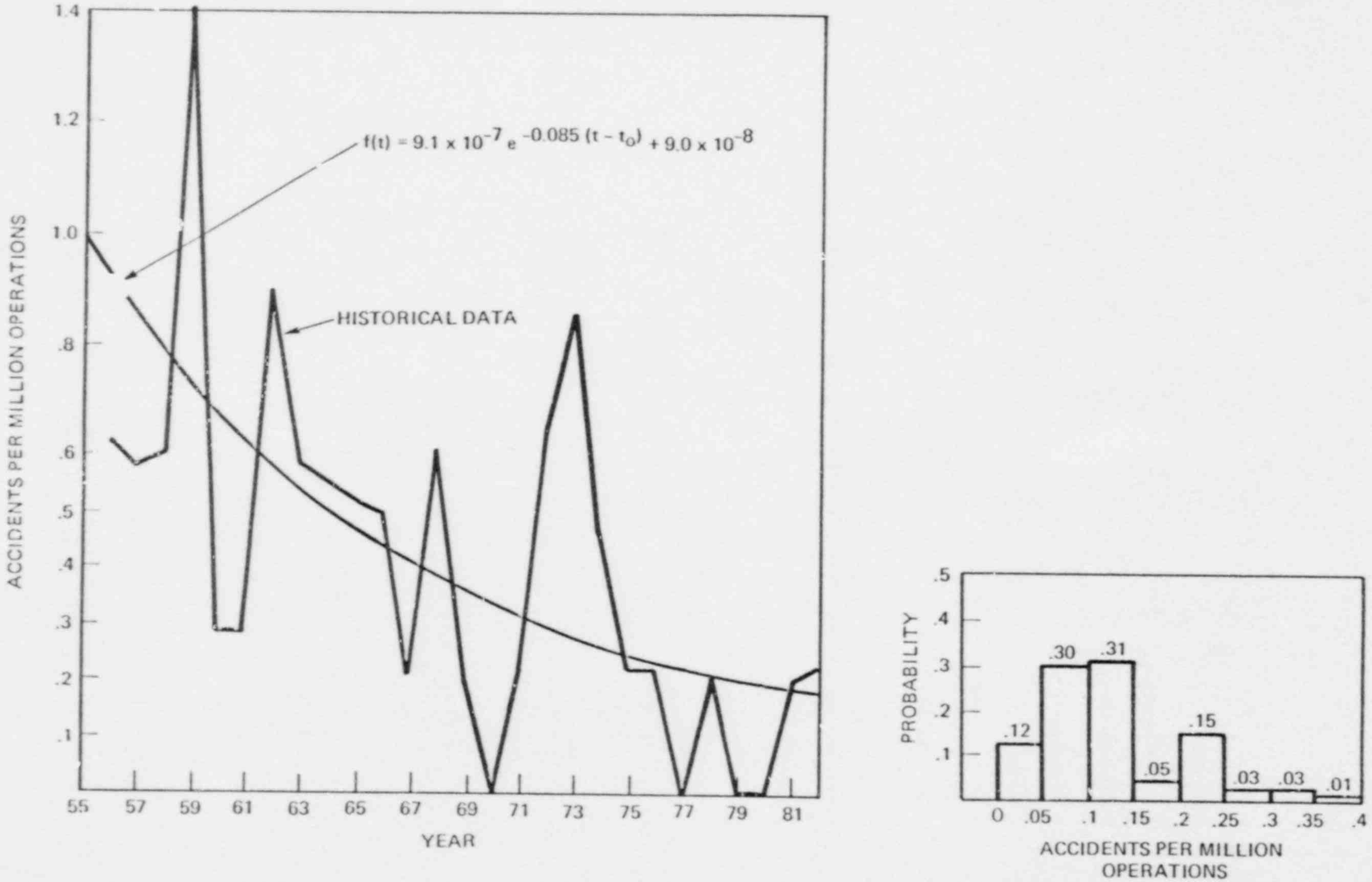

FIGURE 7-8. CRASH RATE VERSUS TIME - SCHEDULED LANDINGS

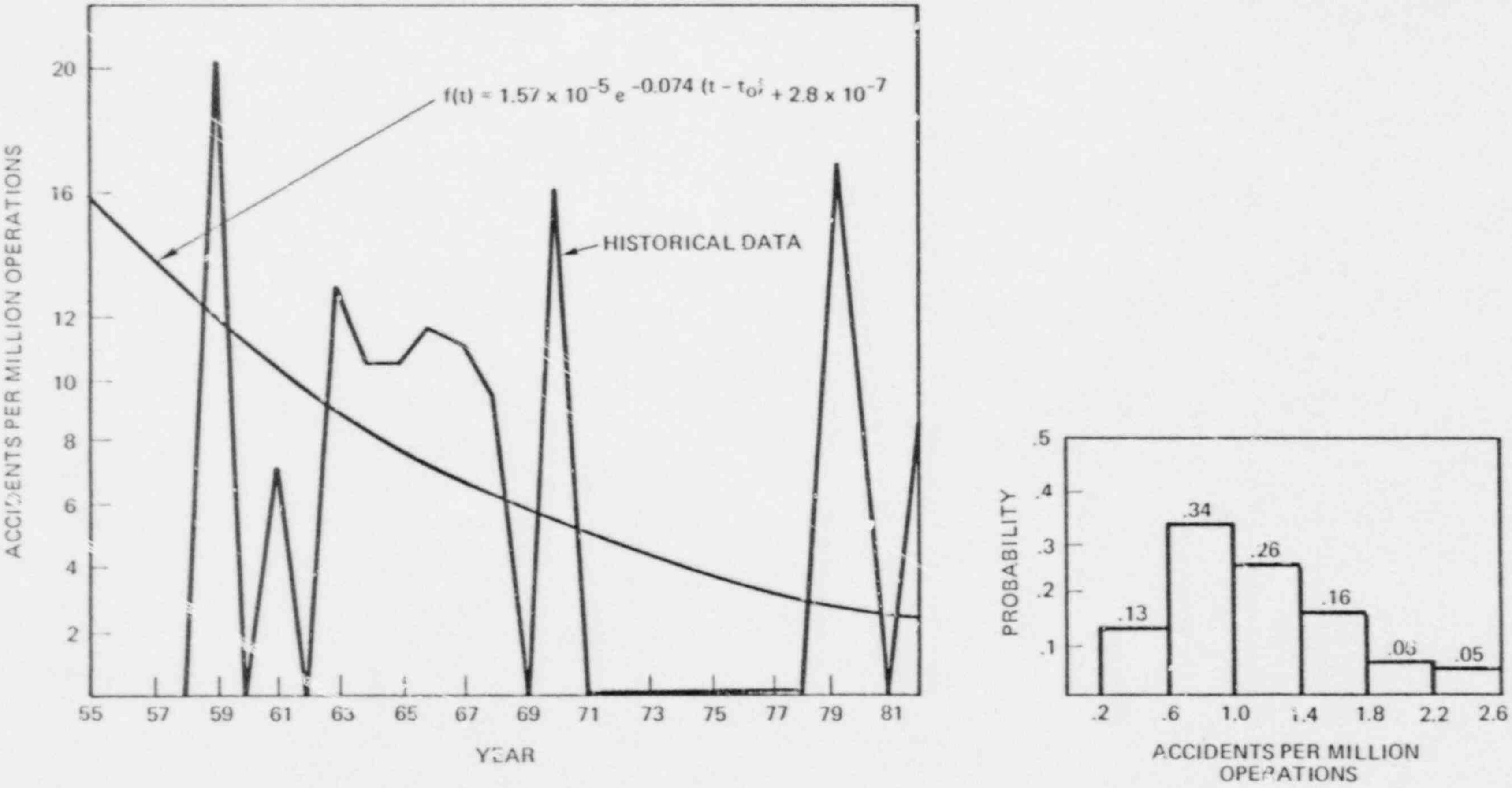

FIGURE 7-9. CRASH RATE VERSUS TIME - NONSCHEDULED LANDINGS

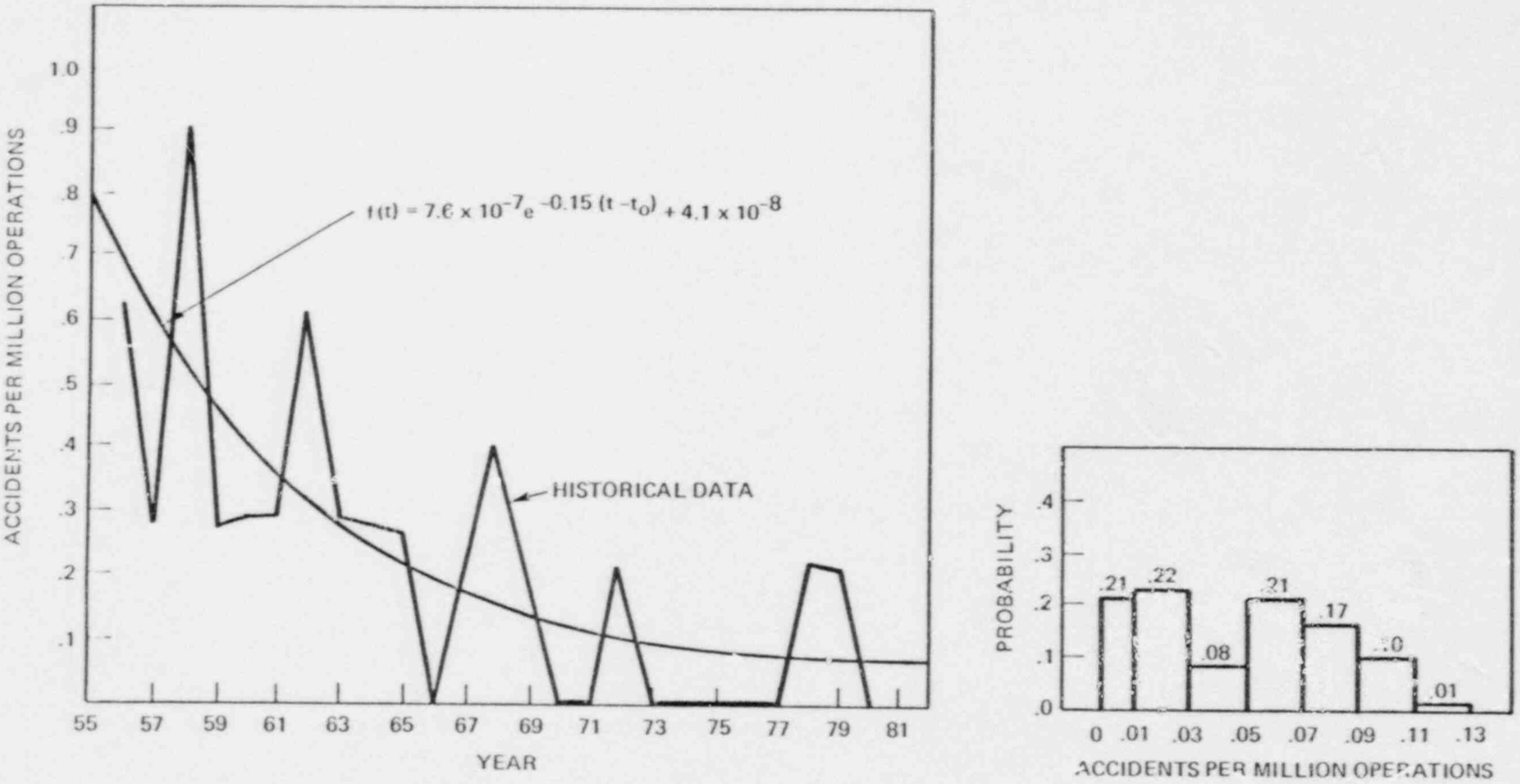

FIGURE 7-10. CRASH RATE VERSUS TIME - SCHEDULED TAKEOFFS

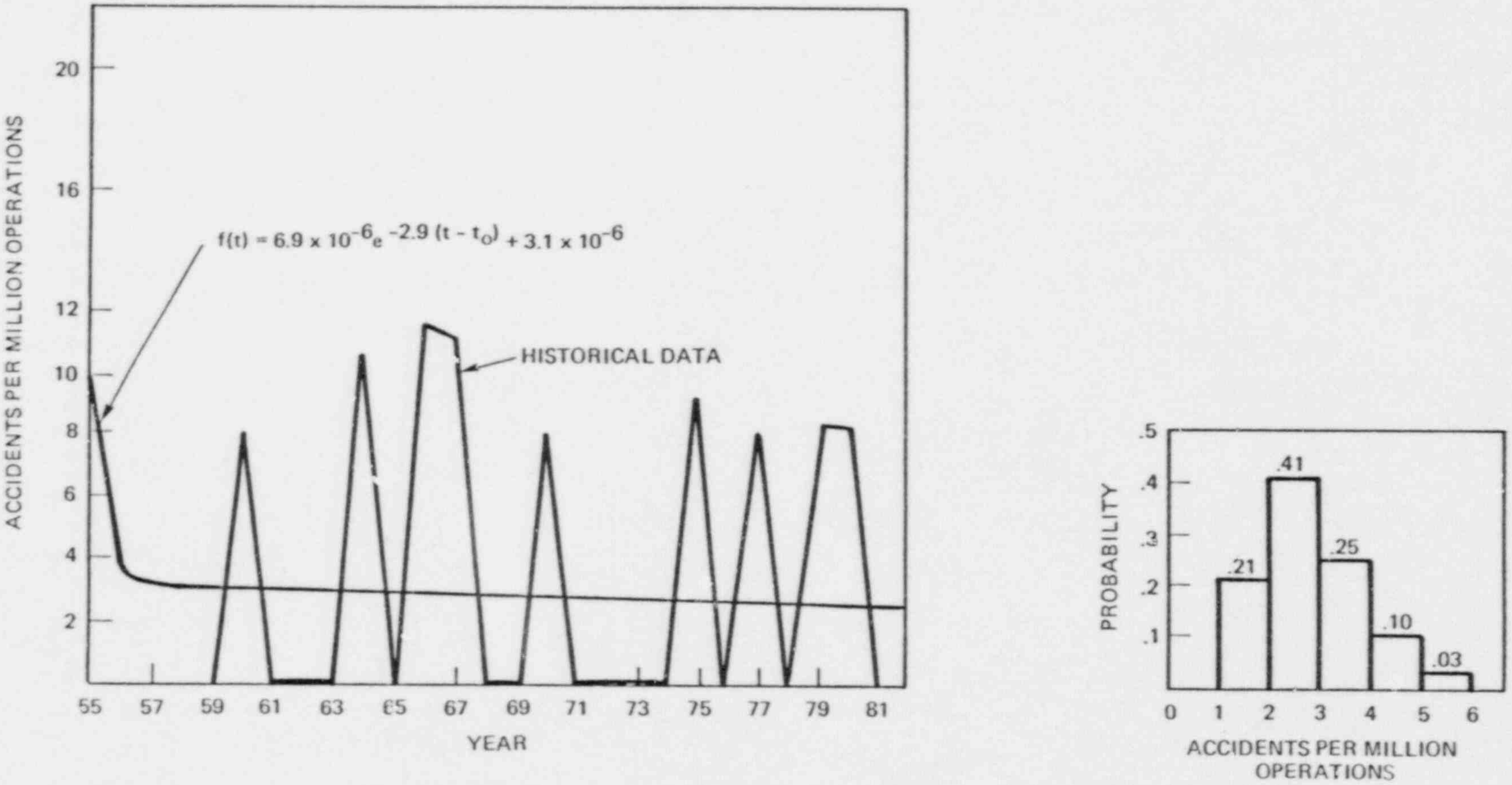

FIGURE 7-11. CRASH RATE VERSUS TIME - NONSCHEDULED TAKEOFFS

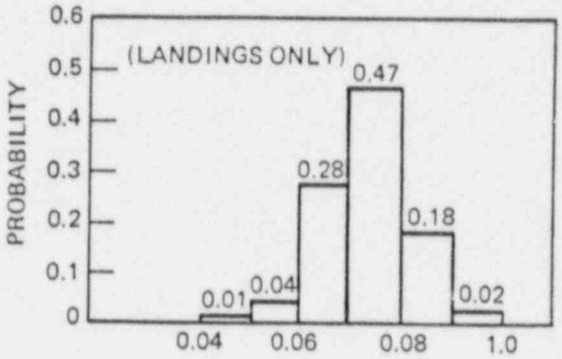

FIGURE 7-12a. THE QUANTITY $\left[\frac{-d}{dr} R_L(r)\right]_{r = 2.7}$

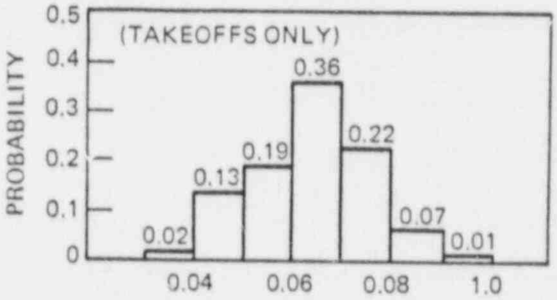

FIGURE 7-12b. THE QUANTITY $\left[\frac{-d}{dr} R_T(r)\right]_{r = 2.7}$

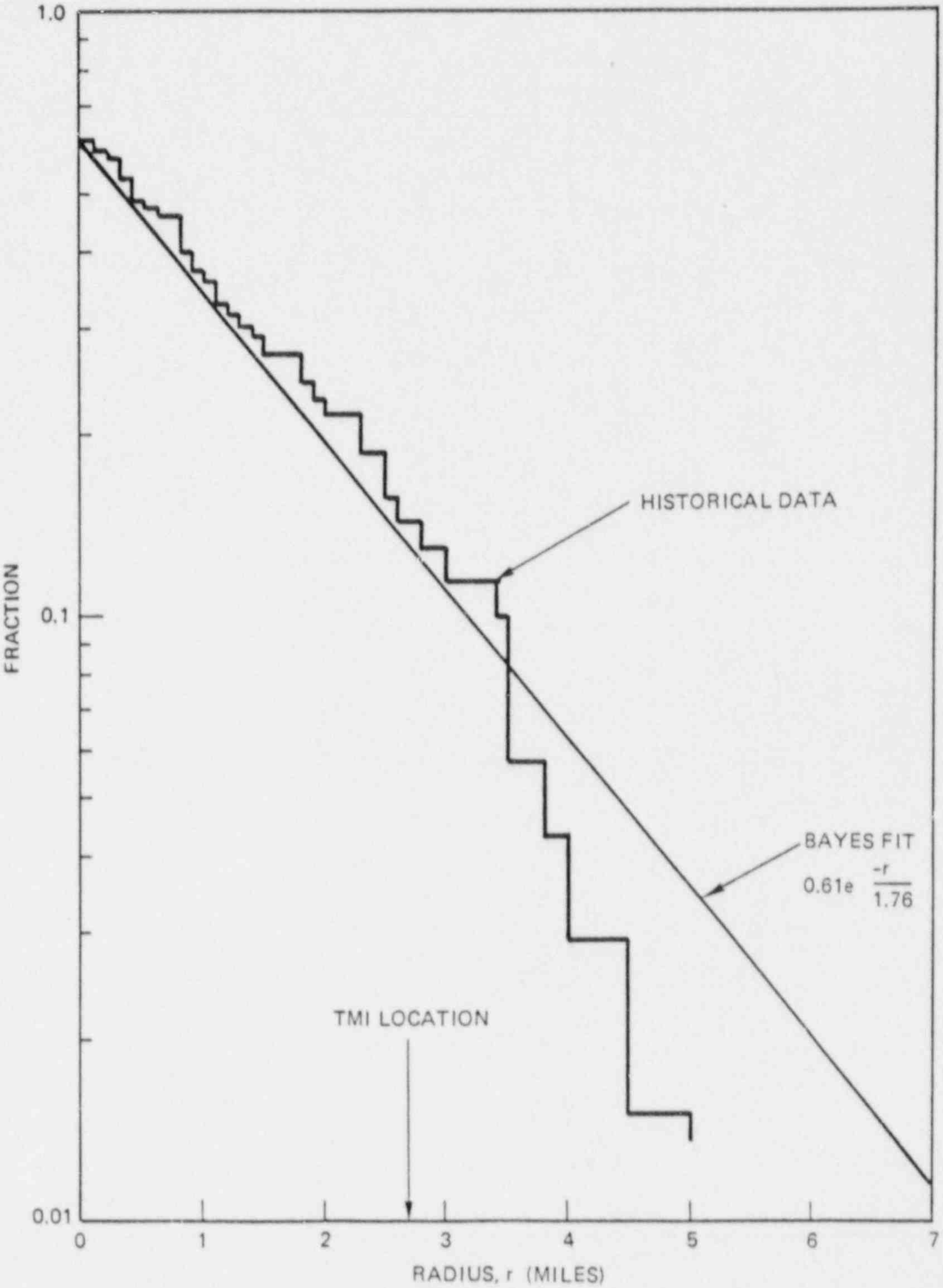

FIGURE 7-13. FRACTION OF LANDING CRASHES OCCURRING AT RADIUS r OR GREATER

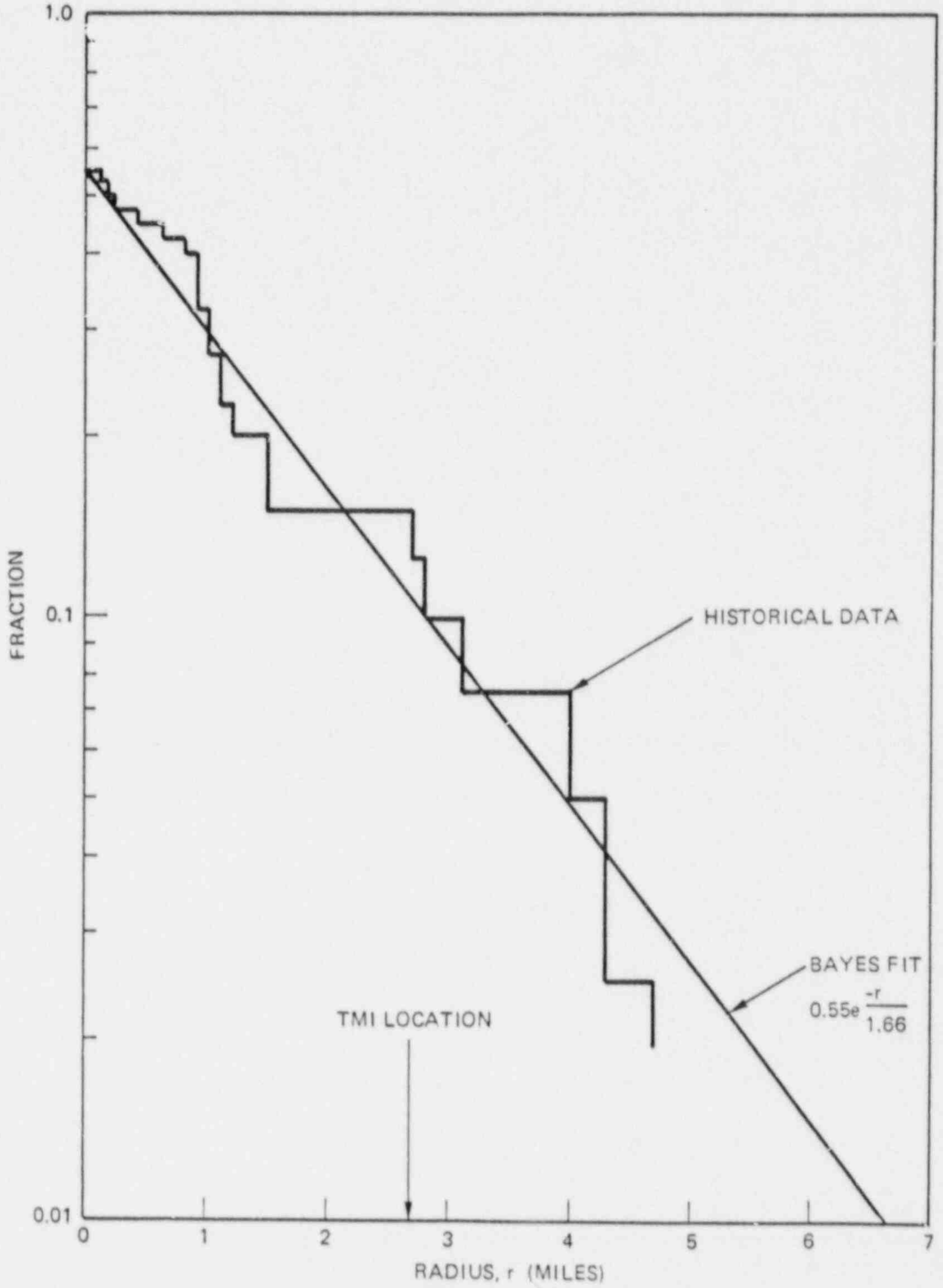

FIGURE 7-14. FRACTION OF TAKEOFF CRASHES OCCURRING AT RADIUS r OR GREATER

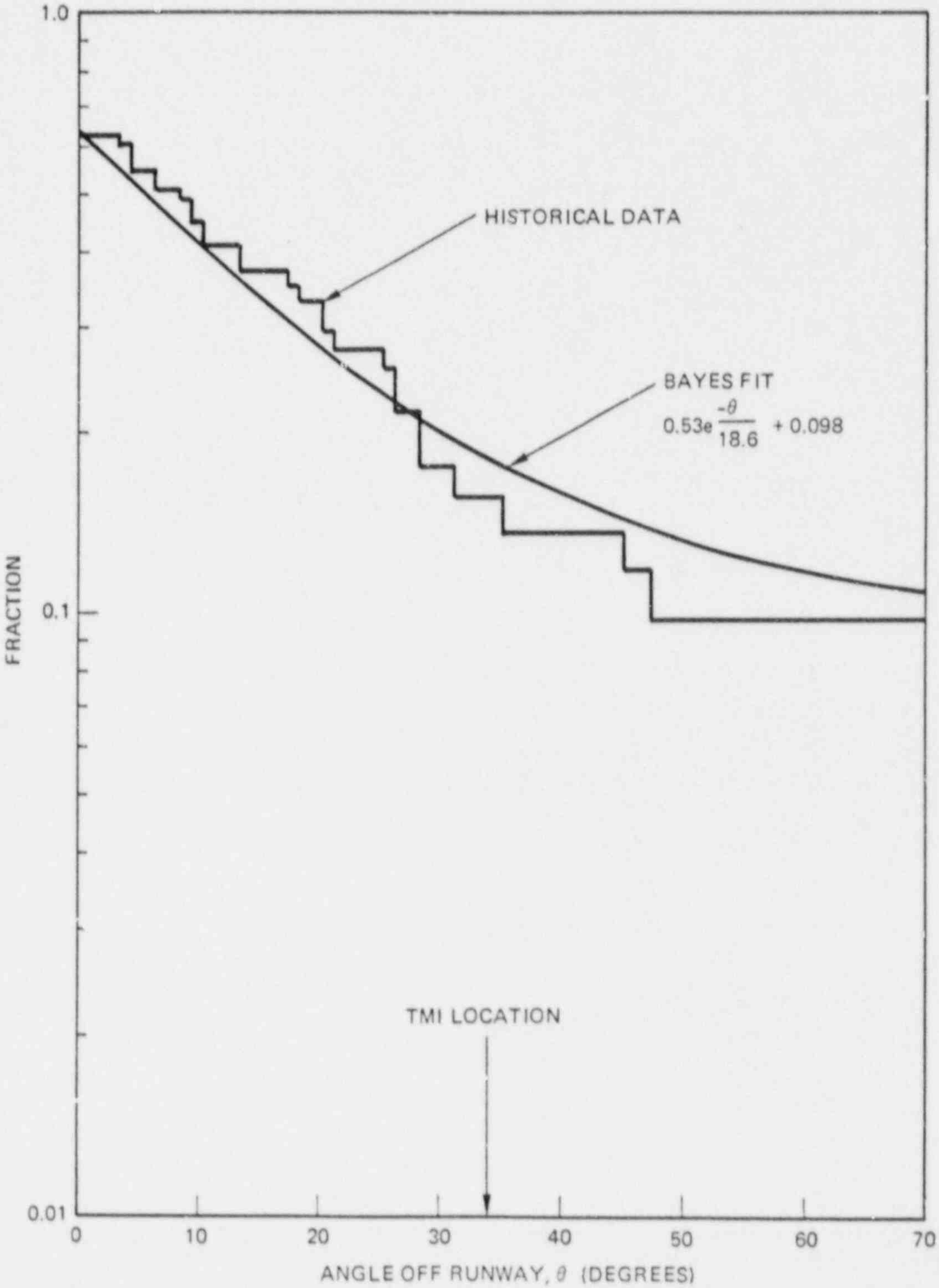

FIGURE 7-15. ANGULAR DISTRIBUTION OF CRASHES - LANDINGS AND TAKEOFFS COMBINED

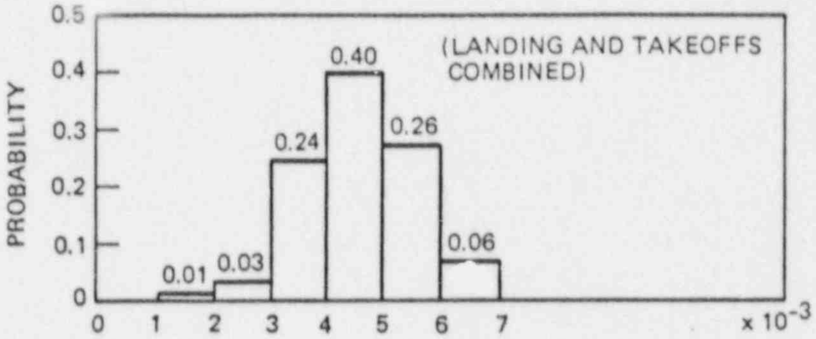

FIGURE 7-16a. THE QUANTITY $\left[\frac{-d}{d\theta}\Theta(\theta)\right]_{\theta = 34^\circ}$

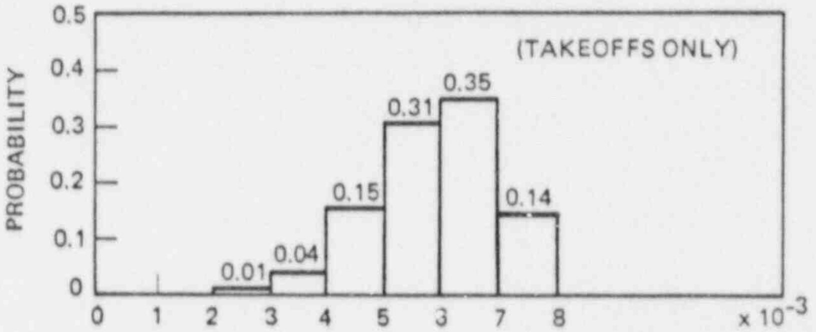

FIGURE 7-16b. THE QUANTITY $\left[\frac{-d}{d\theta}\Theta_T(\theta)\right]_{\theta = 34^\circ}$

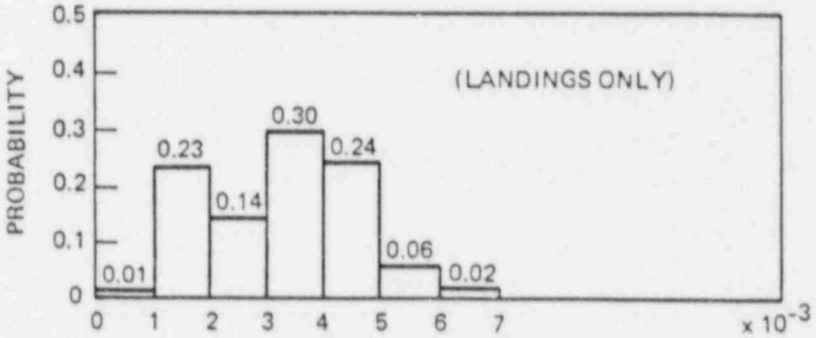

FIGURE 7-16c. THE QUANTITY $\left[\frac{-d}{d\theta}\Theta_L(\theta)\right]_{\theta = 34^\circ}$

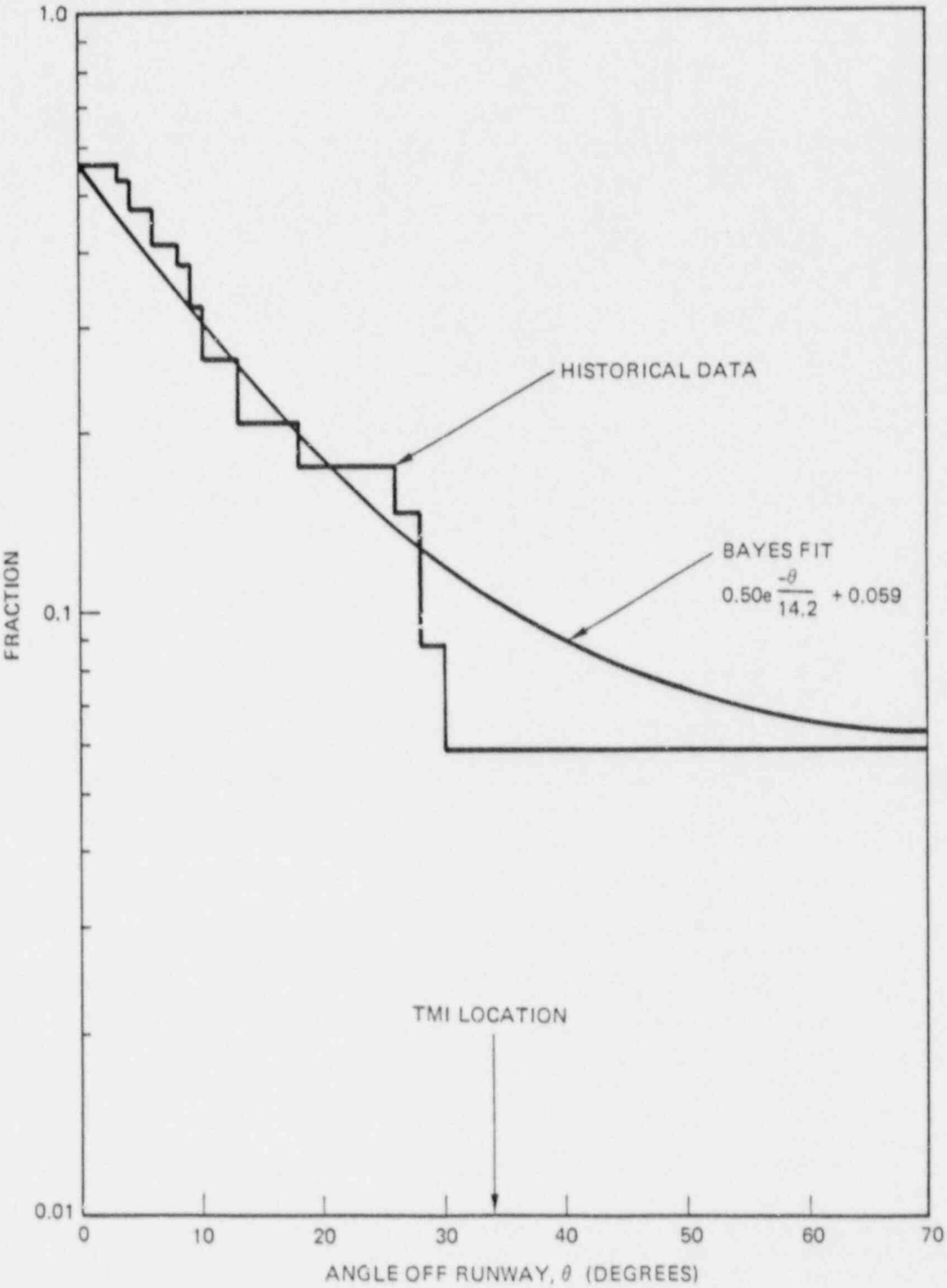

FIGURE 7-17. ANGULAR DISTRIBUTION OF LANDING CRASHES

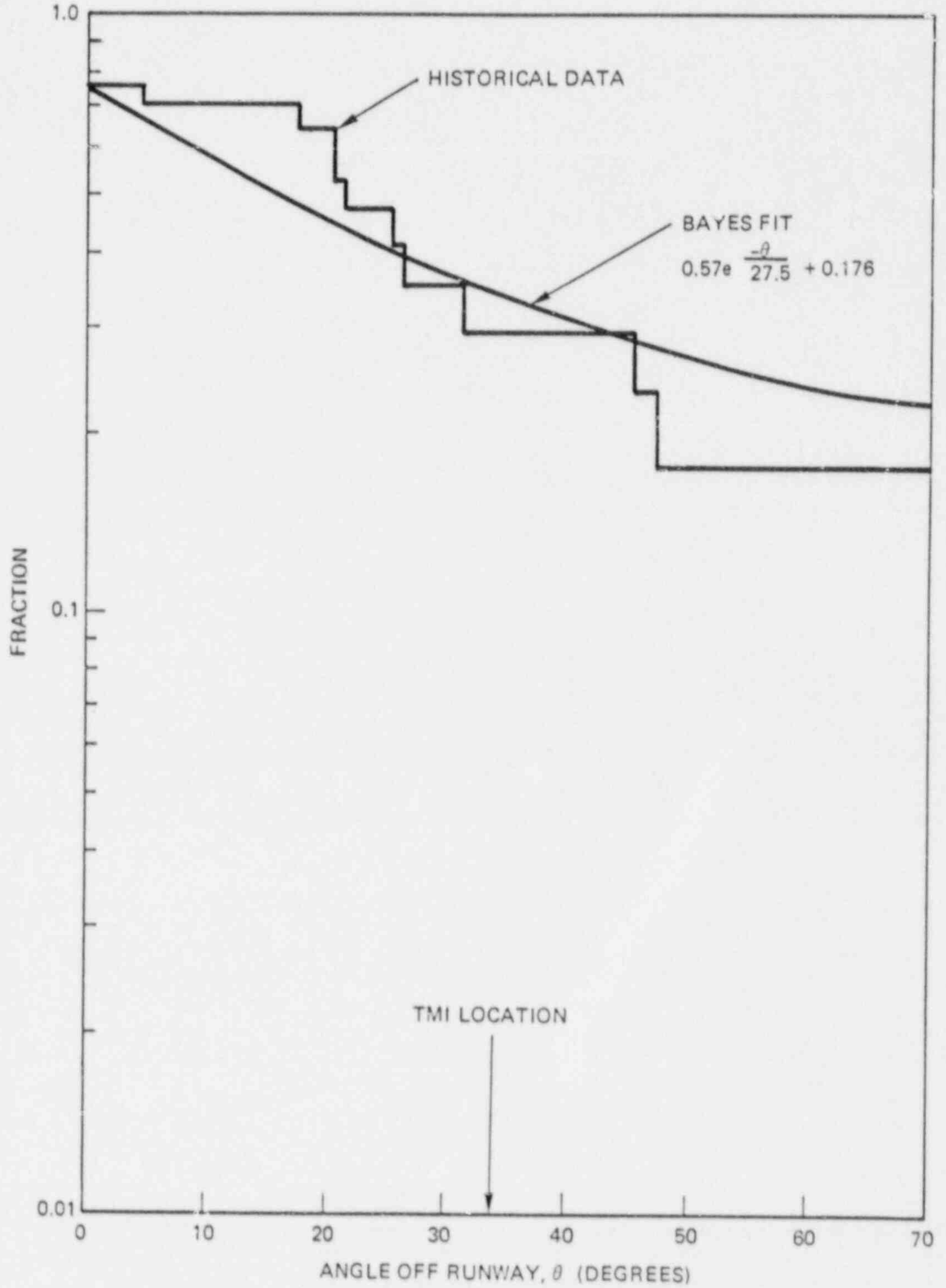

FIGURE 7-18. ANGULAR DISTRIBUTION OF TAKEOFF CRASHES

8. HAZARDOUS CHEMICALS EVALUATION

8.1 INTRODUCTION AND SUMMARY

In this section, the results of the analysis to determine the contribution to the risk from hazardous chemicals in the area surrounding the TMI-1 plant are presented. The chemical hazard to the plant is dominated by toxic chemical releases caused by the rupture of one tank car transporting chemicals on either of the two rail lines adjacent to the plant. The two rail lines are the Roy line to the east and the Shocks line to the west of the plant (Reference 8-1). As shown in Reference 8-2, other sources of hazardous chemicals release, such as rupture of a large ammonia tank located 2.7 miles north of the site and other chemicals stored onsite in very small quantities in fixed tanks, would not generate a high enough concentration of toxic gases at the control room intake to cause control room inhabitability and lead to significant disruption of normal activities of the operators. Scenarios initiated by such releases, therefore, are negligible contributors to the frequency of accident scenarios initiated by the release of toxic chemicals.

This analysis consists of three parts:

1. Estimation of the frequency of major release of different chemicals.
2. Determination of the conditional probability of the inhabitability of the control room, given a major release.
3. Determination of the conditional probability of core damage, given control room inhabitability.

For steps 1 and 2, this analysis relies on methods and results of a similar analysis performed for TMI-1 (Reference 8-2).

The frequency of core melt scenarios initiated by an offsite hazardous chemical release can be calculated in the following way:

$$\phi_{CM} = \sum_{i}^{M} \left[\lambda_{T_i} \; f_{R-T_i} \left\{ \sum_{J}^{N} f_{o_j} \right\} \cdot P_{FR} \right] \quad (8.1)$$

where

λ_{T_i} = frequency per year of a major offsite hazardous chemical release of chemical i.
= $\lambda_T \cdot n_i$.

λ_T = frequency of major releases per tank-car mile.

n_i = number of tank cars shipped per year on either Shocks or Roy of chemical i.

f_{R-T_i} = integral over the track length of the conditional probability of exceeding the toxic limit, given a major release of chemical i on either Shocks or Roy.

f_{o_j} = total fraction of the time when operator action is required to mitigate the scenario caused by tank car rupture.

P_{FR} = fraction of all such operator actions that are unsuccessful and lead to core damage.

M = number of chemical-rail line combinations considered.

N = number of mitigating actions possibly required.

The variables in Equation (8.1) are each calculated or discussed in the following sections.

A list of the chemicals considered in this analysis is given in Table 8-1 (Reference 8-2). The criteria used to determine which chemicals pose the greatest threat to the TMI-1 are:

1. The chemical's toxicity to humans.
2. The relative volatility (vapor pressure) of chemicals that are normally liquids at ambient temperatures and the potential for flashing of chemicals that are normally gases at ambient temperatures. Only catastrophic releases (entire tank contents) were considered since preliminary studies have shown that this type of release always results in highest concentrations downwind. Furthermore, the degree of impact increases as the amount released and rate of release increases.
3. The quantity of material contained in the railroad car.

Certain physical phenomena that could be inferred to occur were not included in the model due to the unavailability of the data required to properly evaluate them. The primary example of this is the assumption that a hydrofluoric acid release would simply evaporate, rather than reacting chemically with the surface, when in fact it is highly reactive. On the one hand, the reason for this line of approach is that there are insufficient data on the composition of the roadbed, the reactions to be expected, the reaction rates, and a variety of other subjects to produce a valid model. Thus, the simplifying assumption was made that no reaction occurs. On the other hand, since such a reaction would decrease the amount of hydrogen fluoride available for release, it is conservative.

The following section describes how the frequency of major releases were calculated. The determination of the conditional probability of exceedance of control room habitability limits, given a major release of each chemical is presented in Section 8.3. Section 8.4 describes the calculation of the conditional probability of core damage, given a control room concentration in excess of toxic limits. Finally, Section 8.5 presents the calculation of core damage frequency due to scenarios initiated by toxic chemical release. It is shown that based on conservative assumptions, this frequency is about 2.6 x 10^{-7} per year which is a negligible contributor to the overall core melt frequency for TMI-1.

8.2 FREQUENCY OF A MAJOR RELEASE

In this section, the frequency of a toxic chemical release from the accidental rupture of a railroad tank car is calculated. This frequency was calculated from Accident/Incident Bulletins 146 (1977) and 151 (1982) (References 8-3 and 8-4). Other data sources were considered but rejected because they were not sufficiently well defined. For instance, a study performed for the Department of Commerce by Systems Laboratory, Inc. (Reference 8-5), and quoted in the Limerick PRA (Reference 8-6) insufficiently documented the source of its numbers and therefore could not be used. Reference 8-7 provides data analyzed by Sandia National Laboratories, which uses an accident rate of 1.5 x 10^{-6} per car mile. It is also apparent that the quality of railroad tank cars is improving. However, the mix of new and old cars used by CONRAIL for shipments on the Shocks and Roy lines is unknown. Therefore, national averages through 1982 for track of the same type were used. The historic data include rail cars of all vintages used during that year, including the new cars. The mix of new and old cars is not expected to be any different on Shocks and Roy than elsewhere in the country. The statistics for 1983 and 1984, when available, are expected to be better as more new cars are brought into service.

Most track in the U.S. (80%) is Class IV; therefore, without more specific information, it was assumed that the portions of Shocks and Roy considered here are also Class IV.

CONRAIL provided GPUN with a list of shipping frequencies (number of tank cars) for the TMI plant for an 18-month period from January 1978 to June 1979 (Reference 8-8). The availability of this list made it possible to calculate release frequencies for each chemical considered.

The frequency, λ_T, of a major release from a tank car rupture was calculated using the following relationship:

$$\lambda_T = \lambda_t \, n_{HMR} \, f_{RHM-M} / n_{HM} \qquad (8.2)$$

where

λ_t = total rate of accidents per train mile.

n_{HMR} = number of cars that release some or all of their contents per accident.

f_{RHM-M} = fraction of such releases that are major.

n_{HM} = number of cars of hazardous material per train.

8.2.1 TOTAL RATE OF ACCIDENTS PER TRAIN MILE (λ_t)

Data from Accident/Incident Bulletins were used to calculate the total rate per train mile of railroad accidents. Some of these data are reproduced in Reference 8-2. The latest data for the years 1977 through 1982 show a range of from 8 x 10^{-6} to 13.8 x 10^{-6} accidents per train mile based on a range of between 10,362 and 4,589 accidents and between 7.3 x 10^8 and 5.7 x 10^8 total train miles. About 75% of these accidents involve derailments and about 40% are due to track defects.

The data are also divided according to speed on three different track locations: main line, yard, and industry siding/unknown. In 1982, the fractions attributed to these types were 48%, 43% and 9%, respectively. Shocks and Roy were assumed to be main line track. For this year of main line data, 32% of all accidents occurred when the trains were traveling at 10 miles per hour or less and 52% at 20 miles per hour or less.

During the period from 1968 to 1982 covered by References 8-3 and 8-4, the threshold value for declaring a rail problem an "accident," went up by a factor of three, from $750 to $4,100. This probably had some effect in reducing the frequency of accidents per train mile. According to Reference 8-2, the rate of accidents varied from 9.2 x 10^{-6} in 1968 to a peak rate of 1.5 x 10^{-5} in 1978 back down to a rate of 8 x 10^{-6} in 1982 after the change in reporting criteria. Based on these data, a mean value of 1 x 10^{-5} was used for the total frequency of accidents per train mile.

8.2.2 NUMBER OF CARS OF HAZARDOUS MATERIAL PER TRAIN (n_{HM})

The Accident/Incident Bulletin data excerpted in Reference 8-2 shows that between 504 and 842 trains were involved in accidents during 1975 to 1982 while carrying hazardous materials. In these trains were a total of between 2,297 and 4,711 cars containing hazardous material. This produces an average 5 cars per train for this whole period, with a yearly average of between 4.6 and 7.4.

8.2.3 NUMBER OF CARS THAT RELEASE HAZARDOUS MATERIALS PER ACCIDENT (n_{HMR})

From the 3,884 trains that transported hazardous materials between 1975 and 1982 and were involved in accidents, 850 cars produced releases. This results in an average over this period of 0.22 cars carrying hazardous materials releasing some or all of their contents per accident. The yearly average over this period varied between 0.18 and 0.23.

8.2.4 FRACTION OF TANK CAR RELEASES THAT ARE MAJOR (f_{RHM-M})

Following the technique used in Appendix H.2 of the Midland PRA (Reference 8-9), the fraction of tank car releases that are major was implied from the number that required evacuation of the area around the accident. From the data given in Reference 8-7, this ratio ranges from 10% to 26%, with a mean value of 18%. The mean value was used in this calculation.

The fact that a release was major (i.e., requiring evacuation) was used in turn to imply that the tank was not just leaking; it was ruptured and released its entire contents rapidly. The release rate from a tank car that was sufficient to prompt an evacuation would, of course, depend on the local authorities and on the toxicity of the chemical released. No site-specific data of this type could be obtained for TMI.

8.2.5 FREQUENCY OF MAJOR RELEASES PER TANK-CAR MILE (λ_T)

The numbers discussed in the last four sections are to be inserted into Equation (8.2) so that the frequency of a major release per tank-car mile is as follows:

$$\lambda_T = \lambda_t \, n_{HMR} \, f_{RHM\text{-}M} \, / n_{HM}$$

$$= (1 \times 10^{-5}) \, (0.22) \, (0.18)/(5)$$

$$= 8.0 \times 10^{-8} \text{ per car mile}$$

8.3 DETERMINATION OF THE CONDITIONAL PROBABILITY OF EXCEEDANCE OF CONTROL ROOM HABITABILITY LIMITS, GIVEN A MAJOR RELEASE, FOR EACH CHEMICAL

This section summarizes the methodology, data, and procedures used in Reference 8-2 to determine the habitability of the TMI-1 control room for various chemical releases and meteorological conditions. The objective is to determine the conditional probability that a major release will result in the exceedance of control room habitability limits. This conditional probability varies along the track. Therefore, the conditional probability is integrated along the track.

8.3.1 EVAPORATION AND DISPERSION MODELS

The evaporation and dispersion of contaminants resulting from a hazardous chemical spill were analyzed using a modification of the methods suggested in NUREG-0570 (Reference 8-10) and Regulatory Guide 1.78 (Reference 8-11). The most significant modifications were:

1. Plumes resulting from the spill of chemicals whose vapors are much lighter than air are treated as both buoyant and nonbuoyant plumes.

2. Enhanced dispersion due to plume meandering during neutral and stable low wind speed meteorological conditions is accounted for.

3. Enhanced dispersion due to interaction of the plume with the reactor building complex is accounted for if tall structures are in the path of the plume as it travels from its source toward the control room air intake vent.

The various components of the evaporation and dispersion models are presented below. It is shown that the modified models still provide conservative estimates of control room toxic vapor concentrations. The model assumes that the entire contents of a single railroad tank car or stationary storage container is released to the environment instantaneously. Preliminary analysis showed that this assumption results in a "worst case" scenario in the control room (Reference 8-2).

8.3.1.1 Evaporation Models

The evaporation model contains the following components:

1. A model to calculate the time-dependent surface area of a liquid spill.
2. A model to calculate the initial flashing of a compressed gas or pure low boiling point liquid release and the boiloff of the remaining liquid pool (Vaporization Class I).
3. A model to calculate the evaporation rate of a chemical that is a pure liquid at ambient conditions (Vaporization Class II).
4. A model to calculate the evaporation rate of the toxic components of a liquid mixture (Vaporization Class III).

8.3.1.2 Dispersion Models

Gaussian plume models were employed in this analysis to account for the dispersion of the instantaneous puff formed by instantaneous flashing of a Vaporization Class I chemical and the continuous plume formed from boiloff evaporation of the liquid spills. The models presented in NUREG-0570 were modified to account for plume rise, meandering, and plume-building wake interactions.

The concentration of toxic chemical, $C_0(t)$, at any time, t, at a downwind receptor is the sum of the instantaneous puff (if it occurs) and continuous plume concentrations. That is,

$$C_0(t) = C_{puff}(t) + C_{plume}(t)$$

For a Vaporization Class II chemical, no flashing occurs on exposure to the atmosphere, so $C_{puff}(t) = 0$ always.

In applying both the instantaneous puff model and the continuous plume model, it is assumed that the wind is always blowing from the accident source directly toward the control room air intake vent.

For toxic vapors much lighter than air, such as ammonia, the rise of the continuous plume center line was calculated using the Briggs plume rise formulas (Reference 8-12).

For all buoyant releases, the release height, was assumed equal to zero. The gradient of potential temperature was assumed equal to .02, .0375, and .05 °C per meter for E, F, and G stabilities, respectively. For instantaneous puff releases, the plume center line height was assumed equal to continuous plume center line height at time zero. This is a conservative assumption for the cases considered since the instantaneous puff has considerably more buoyant potential than the continuous plume.

It should be noted that no credit was taken for plume meandering or plume-building wake interactions for buoyant plumes that rise above the reactor building complex.

For vapors much heavier than air, the plume center line was assumed to be at ground level. For vapors whose density does not differ significantly from that of air, the plume center line height was assumed equal to the air intake vent height. These assumptions are not substantially different since the TMI Unit 1 air intake vent is only about 16 feet above ground level.

There is ample evidence to confirm the existence of plume meandering in the vicinity of the TMI site during stable, low wind speed conditions. A series of SF_6 tracer gas atmospheric diffusion experiments were conducted on Three Mile Island during 1971. The results of these experiments are reported in Reference 8-13. They confirm the existence of plume meandering for releases in open areas and for releases affected by building wake interactions. As a result, the continuous plume dispersion model was modified to account for plume meandering.

Plume meander factors were not applied to the instantaneous puff model since the effect of meandering on puff dispersion is not presently well understood.

Due to the arrangement of the buildings and structures of the TMI Nuclear Station, plumes approaching the Unit 1 control room air intake vent from the west and south are unobstructed while plumes approaching from the other directions must pass around or over some portion of the reactor building complex and cooling towers to reach the vent. Dispersion in the vicinity of these structures is too complex to model accurately. As a result, a relatively simple but conservative modification was applied to the instantaneous puff and continuous plume dispersion models. The modification involves adjusting the plume standard deviations (sigmas) to reflect interaction with the reactor building complex. No credit was taken for interaction with the cooling towers even though they can significantly enhance plume dilution.

8.3.1.3 Modeling of Toxic Gas Concentrations in the Control Room Isolation Zone

A time-dependent model was used to calculate the concentration of toxic gases in the control room isolation zone. The details of the model and the computer code used to perform the calculations are presented in Reference 8-2.

The intake tunnel model converts the rate of introduction of the toxic gas (evaporation or leakage in grams per second) into a concentration at the mouth of the intake tunnel at a later time, the delay being equal to the ratio of the distance between the source and the mouth of the intake tunnel to the wind speed. This concentration is tracked from the mouth of the tunnel to the intake damper, moving forward by a volume equal to the product of the length of the time step and the intake flow rate. If this volume is greater than the intake tunnel volume, the appropriate time delay is used instead. The model considers the fact that, at the intake damper, a portion of the flow is diverted to the halls and machine shop, while the remainder goes into the control room ventilation system.

8.3.2 METHODOLOGY EMPLOYED TO FIND THE CONDITIONAL PROBABILITY OF EXCEEDANCE

The maximum concentration of a chemical in the control room atmosphere after a spill is a strong function of four meteorological variables: wind direction, wind speed, stability, and temperature. The evaporation rate is a function of temperature and, in many cases, windspeed. The dispersion of the plume is determined by the stability and windspeed, while the plume rise for chemicals lighter than air is determined by windspeed, stability, and evaporation rate. Finally, the difference in the wind direction and the direction from the spill to the intake, along with the dispersion of the plume, determine what fraction of the peak concentration is present at the intake. A method was developed to systematically take these factors into account in determining the conditional probability of exceeding the toxic limits in the control room, given a chemical spill of a given amount of a given chemical at a given location.

Two methods were used for determining the ambient temperature at the time of the spill. The conservative method assumes that the evaporation takes place at the highest temperature consistent with the stability; 100°F for stability Classes A through D, and 80°F for stability Classes E through G. A more realistic method, used only for hydrofluoric acid spills, is to find the control room concentrations as a function of temperature. For both methods, the peak concentrations are found as a function of windspeed for a fixed atmospheric dilution factor.

The assessment of the conditional probability of exceedance will be considered first for the conservative method. For each combination of windspeed and stability, the peak control room concentration, C_{max}, evaluated at an atmospheric dispersion factor of $(X/Q)_{ref}$, is compared to the toxic limit for that chemical, C_{lim}. The limiting value of the atmospheric dispersion factor, $(X/Q)_{lim}$, is found using

$$(X/Q)_{lim} = \frac{(X/Q)_{ref}\, C_{lim}}{C_{max}} \qquad (8.3)$$

Only atmospheric dispersion factors greater than $(X/Q)_{lim}$ at the vent will result in exceedance of the toxic limit in the control room. Using the meteorological methods in Reference 8-11, the plume standard deviations, σ_y and σ_z, and the atmospheric dilution factor at the vent height and plume center line, $(X/Q)_{CL}$ is found. If this value is less than $(X/Q)_{lim}$, the plume presents no possibility of exceeding the toxic limit for this stability and windspeed. Otherwise, a further step is required. The atmospheric dispersion factor, X/Q, has the following function form in the cross-wind direction:

$$X/Q = (X/Q)_{CL} \exp\left[-y^2/2\sigma_y^{\,2}\right] \qquad (8.4)$$

where y is the lateral distance between the plume center line and the vent, measured perpendicular to the wind direction at the vent height, and X/Q is the atmospheric dilution at that point. Thus the plume only presents a hazard within a band within y_{lim} of the center line, where y_{lim} is the solution of Equation (8.4) at $(X/Q)_{lim}$:

$$y_{lim} = \sigma_y \; 2\ell n \; [(X/Q)_{CL}/(X/Q)_{lim}]^{1/2} \qquad (8.5)$$

The half-width of the sector of the plume for which exceedances occur is thus

$$A = \tan^{-1} (y_{lim}/x) \qquad (8.6)$$

where x is the distance from the spill to the intake. If the wind direction that would carry the vapor directly toward the vent is B, the wind directions between B - A and B + A lead to exceedances. Using meteorological data for a sample year, tabulated in the form of the number of occurrences of a given stability with a given range of windspeeds and a given range of directions, the number of occurrences of wind directions between B - A and B + A for the given stability and windspeed are found. These results are summed over all windspeeds and stabilities and the sum divided by the total number of hours of meteorological data in the sample year, yielding the conditional frequency of exceedance of toxicity limits in the control room, given a spill.

For the more realistic method, the same procedure is followed except that meteorological data are grouped into 10°F ranges and the conditional probability is found for that temperature range. These are multiplied by the probabilities of their respective groups and summed over all temperature groups to give the conditional frequency of exceedance.

The track was broken into segments, with each segment represented by its central point. The conditional probability of exceedance at that point was then multiplied by the length of the segments and the resulting values summed over the length of the rail line considered. The portion of the track considered was that within 5 miles of the plant. The resulting line integral of the conditional probability was multiplied by the frequency of major releases of that chemical per mile per year to find the frequency of exceedance for that chemical.

8.3.3 RESULTS

The results of the conditional probability of exceedance calculations are given in Table 8-2. The chemical data and meteorological information needed for these calculations are presented in Tables 3-1 through 3-16 of Reference 8-2. Note that the values given are integrals over the track within 5 miles from the plant and that the values are given in miles.

To obtain the annual frequency of exceedance, these probabilities must be multiplied by the corresponding number of shipments per year and the frequency of major release per tank car accident. Table 8-3 provides the total number of shipments for each of the chemicals analyzed in this analysis. Table 8-4 lists the annual frequency of toxic limit exceedance in the control room for each chemical.

The resulting total (mean) annual frequency of toxic limit exceedance in the control room is then [see Equation (8.1)]:

$$\phi_E = \sum_{i=1}^{M} \lambda_{T_i} \, f_{R-T_i}$$

$$= 1.3 \times 10^{-5} \text{ per year}$$

8.4 CONDITIONAL PROBABILITY OF CORE DAMAGE, GIVEN A CONTROL ROOM CONCENTRATION IN EXCESS OF TOXIC LIMITS

In the scenarios considered so far, a railroad car filled with a toxic chemical has ruptured, and the resulting toxic plume has made it to the control room air intake and has infiltrated the control room in a concentration in excess of the toxic limit value. To be concentrated enough, the toxic plume half-width will be between 50 and 150 feet. For many of these chemicals, the operator will isolate the control room prior to the TLV being reached, based on smell or skin irritation. In some cases, however, he will not be aware of the situation in time. It was estimated that, depending on the chemical, the conditional probability P_{FI} of failing to isolate ranges between 0.1 and 1.0, with a mean value of approximately 0.4.

This value is comparable to some of the highest values calculated in the human actions analysis report for high stress situations. In cases in which the control room remains unisolated, one of two situations may evolve from the operator's extreme discomfort at being exposed to the TLV:

1. Most likely, the operator will trip the plant because of his apprehension about his ability to perform.

2. He will become incapacitated prior to being able to trip the plant.

If the operator trips the plant, normally operating systems will insert the control rods, trip the turbine, ramp back the feedwater, and dump steam, thereby leveling off at the steam dump and feed flow rates required to remove decay heat. No operator action is required. If the plant continues to run, it will do so until some onsite or offsite disturbance causes the plant to trip automatically. On the average, this happens 3.5 times per year (see Data Analysis Report, Table 3-8), which means that the likelihood per operating hour of the plant tripping is about 5.3×10^{-4} (based on 75% plant availability). Therefore, automatic trip is not nearly as likely as the operator tripping the plant manually. In either case, one of the systems that must respond automatically will need to fail for operator response to be required to prevent core damage release. It was assumed that the operator tripped the plant. That is, the probability, P_{OT}, that the operator trips the plant is 1.

The plant response to a trip is modeled by the reactor trip event tree, as presented in the Plant Model Report. Reference to this tree indicates that the sum of all split fractions corresponding to the failure of systems whose automatic response will be needed is less than 0.1. This value is used as the conditional probability, P_{AR}, that, given that the operator tripped the plant, additional manual action will be needed.

In those situations requiring manual action after the control room concentration has exceeded the TLV, the operator may don a Scott-AirPack and still be able to act or operators not in the control building may enter it to help out. The plume half-widths must be fairly narrow if the concentration is to exceed the TLV in the control room. Any operators outside the part of the plume that exceeds the TLV will not be incapacitated. Since the maximum plume half-width is about 150 feet, operators may come from most locations onsite other than the control building or from offsite. These operators would don breathing apparatus and/or protective clothing and enter the control building to, for instance, actuate high pressure injection to keep the core covered.

Note that in the notation of Equation (8.1)

$$P_{FI} \cdot P_{OT} \cdot P_{AR} = \sum_{j=1}^{N} f_{oj}$$

Based on the time available to act (about 1 hour), the distance from which the new operators must come, and the stress involved in the situation, a conservative estimate of $P_{FR} = 0.5$ for the conditional probability of failing to perform the required manual actuations was made. This number is higher than the likelihood of failure to recover from the worst case loss of offsite power scenario in which there is a comparable level of stress and the time available.

8.5 TOTAL FREQUENCY OF CORE DAMAGE INITIATED BY TOXIC CHEMICAL RELEASE

The total mean frequency of core damage due to scenarios initiated by rail car accidents releasing toxic chemicals is calculated from using the frequencies and conditional probabilities calculated in the previous sections in the following equation [see Equation (8.1)]

$$\begin{aligned} \phi_{CM} &= \phi_E \cdot P_{FI} \cdot P_{OT} \cdot P_{AR} \cdot P_{FR} \\ &= (1.3 \times 10^{-5})(0.4)(1.0)(0.1)(0.5) \\ &= 2.6 \times 10^{-7} \text{ per year} \end{aligned}$$

This number is several orders of magnitude smaller than the core melt frequency due to other TMI-1 accident scenarios. It is therefore concluded that toxic chemical hazard is a negligible contributor to the overall risk.

8.6 REFERENCES

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TABLE 8-1. CHEMICALS ANALYZED AND SOURCES OF RELEASE

Chemical	Sources*
Acetic Acid, Glacial	Roy, Shocks
Acetic Anhydride	Roy, Shocks
Acrylonitrile	Shocks
Ammonia, Anhydrous	Roy, Shocks
Ammonia, 29.4% Weight, Aqueous	Manly-Regan
Bromine	Shocks
Chlorine	Shocks
Chromic Fluoride, 20% Weight in HF	Roy
Coal Tar, Light Oil	Shocks
Ethyl Acrylate	Roy, Shocks
Ethylene Oxide	Shocks
Formaldehyde, 37% Weight, Aqueous	Shocks
Hexane	Shocks
Hydrochloric Acid, 36% Weight, Aqueous	Shocks
Hydrofluoric Acid, Anhydrous	Roy, Shocks
Phosphorus Oxychloride	Shocks
Propylene Oxide	Shocks
Vinyl Acetate	Shocks
Vinyl Chloride	Roy, Shocks

*The Roy line runs to the east of the plant; the Shocks lines to the west. The Manly-Regan tank is 4,400 meters north of the plant.

TABLE 8-2. CONDITIONAL PROBABILITY OF EXCEEDANCE OF TOXIC LIMITS IN CONTROL ROOM, GIVEN A MAJOR RELEASE, INTEGRATED OVER TRACK WITHIN 5 MILES OF TMI-1

Chemical	Roy	Shocks
Acetic Acid, Glacial	.010934	.002872
Acetic Anhydride	.000679	.000000
Acrylonitrile	--	.014661
Ammonia, Anhydrous	.026145	.023608
Bromine	--	.082288
Chlorine	--	.078189
Chromic Fluoride	.102293	--
Coal Tar, Light Oil	--	.003605
Ethyl Acrylate	--	.000227
Ethylene Oxide	--	.023453
Formaldehyde	--	.000001
Hexane	--	.000000
Hydrochloric Acid	--	.002672
Hydrofluoric Acid	.102293	.094398
Phosphorus Oxychloride	--	.079749
Propylene Oxide	--	.012521
Vinyl Acetate	--	.034216
Vinyl Chloride	.015827	.009610

TABLE 8-3. NUMBER OF SHIPMENTS PER YEAR OF THE IMPORTANT HAZARDOUS CHEMICALS (n_i)

Chemical	Line	Shipments per Year
Acetic Acid	Shocks	79.3
	Roy	26
Acetic Anhydride	Shocks	34.7
	Roy	34.7
Acrylonitrile	Shocks	134.7
Ammonia, Anhydrous	Shocks	180
	Roy	46
Bromine	Shocks	47.3
Chlorine	Shocks	1,046
Chromic Fluoride	Roy	127.3
Coal Tar, Light Oil	Shocks	118.7
Ethyl Acrylate	Shocks	334.7
Ethylene Oxide	Shocks	236.7
Hydrochloric Acid	Shocks	117
Formaldehyde, 37% Weight	Shocks	50.7
Hydrofluoric Acid	Shocks	96
	Roy	42.7
Phosphorus Oxychloride	Shocks	41.3
Propylene Oxide	Shocks	236.7
Vinyl Acetate	Shocks	32
Vinyl Chloride	Shocks	2,888.7
	Roy	42

TABLE 8-4. ANNUAL FREQUENCY OF EXCEEDANCE OF TOXIC LIMITS IN CONTROL ROOM

Chemical	Roy	Shocks
Acetic Acid, Glacial	2.27×10^{-8}	1.82×10^{-8}
Acetic Anhydride	1.88×10^{-9}	0.00×10^{-0}
Acrylonitrile	--	1.58×10^{-7}
Ammonia, Anhydrous	9.62×10^{-8}	3.40×10^{-7}
Bromine	--	3.14×10^{-7}
Chlorine	--	6.54×10^{-6}
Chromic Fluoride	1.04×10^{-6}	--
Coal Tar, Light Oil	--	3.42×10^{-8}
Ethyl Acrylate	--	6.08×10^{-9}
Ethylene Oxide	--	4.44×10^{-7}
Formaldehyde	--	4.06×10^{-12}
Hexane	--	0.00×10^{-0}
Hydrochloric Acid	--	2.50×10^{-7}
Hydrofluoric Acid	3.49×10^{-7}	7.25×10^{-7}
Phosphorus Oxychloride	--	2.63×10^{-7}
Propylene Oxide	--	2.37×10^{-7}
Vinyl Acetate	--	8.76×10^{-8}
Vinyl Chloride	5.32×10^{-8}	2.22×10^{-6}

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PLG-0525
Volume 7
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Three Mile Island Unit 1 Probabilistic Risk Assessment

ENVIRONMENTAL AND EXTERNAL HAZARDS REPORT

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SUMMARY OF CONTENTS

CONTENTS

Section Page

BOOK 1 OF 2

CONTENTS (continued)

CONTENTS (continued)

BOOK 2 OF 2

LIST OF TABLES

LIST OF TABLES (continued)

LIST OF FIGURES

Figure Page

BOOK 1 OF 2

LIST OF ACRONYMS

Abbreviation	Definition
ACR	air-cooled reactor
ADV	atmospheric dump valve
AEC	U.S. Atomic Energy Commission
AOV	air-operated valve
ATOG	abnormal transient operational guidelines
ATWS	anticipated transient without scram
B&W	Babcock & Wilcox Company
BOP	balance of plant
BRP	Big Rock Point
Btu	British thermal unit
BWR	boiling water reactor
BWST	borated water storage tank
CAR	corrective action report
CARS	condenser air removal system
CAS	chemical addition system
CBVS	control building ventilation system
CCF	common cause failure
CDF	cumulative distribution function
CFT	core flooding tank
CIV	containment isolation valve
CSF	conditional split fraction
CST	condensate storage tank
CRO	control room operator
CWS	circulating water system
DHCCW	decay heat closed cooling water
DHR	decay heat removal
DHRS	decay heat removal system
DHRW	decay heat river water
DPD	discrete probability distribution
EFW	emergency feedwater
EEHR	Environmental and External Hazards Report
EHC	electrohydraulic control
EOF	emergency operations facility
EPRI	Electric Power Research Institute
ESD	event sequence diagram
ESAS	engineered safeguards actuation system
ETC	event tree code
FAA	Federal Aviation Administration
FHA	fire hazards analysis
FSAR	Final Safety Analysis Report
FTAP	Fault Tree Analysis Program

LIST OF ACRONYMS (continued)

Abbreviation	Definition
GCR	gas-cooled reactor
GE	General Electric Company
GPUN	GPU Nuclear Corporation
HCR	human cognitive reliability
HCLPF	high confidence low probability of failure
HIA	Harrisburg International Airport
HPI	high pressure injection
HPIS	high pressure injection system
HSPS	heat sink protection system
HTM	high trajectory missile
HVAC	heating, ventilating, and air conditioning
ICCS	intermediate closed cooling system
ICCW	intermediate closed cooling water
ICS	integrated control system
IREP	Interim Reliability Evaluation Program
LBIS	line break isolation system
LCO	limiting condition for operation
LER	Licensee Event Report
LOCA	loss of coolant accident
LOFW	loss of main feedwater
LONS	loss of nuclear services
LORI	loss of reactor coolant system inventory
LORW	loss of river water
LOSP	loss of offsite power
LPI	low pressure injection
LPIS	low pressure injection system
LSS	low speed stop
LTM	low trajectory missile
MCC	motor control center
MFPT	main feedwater pump trip
MFW	main feedwater
MGL	multiple Greek letter
MOV	motor-operated valve
MSIV	main steam isolation valve
MSLB	main steam line break
MSS	main steam system
MSSV	main steam safety valve
MSV	main steam valve
MUP	makeup and purification
MUT	makeup tank

LIST OF ACRONYMS (continued)

Abbreviation	Definition
NPE	Nuclear Power Experience
NRC	U.S. Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSCCS	nuclear services closed cooling system
NSCCW	nuclear services closed cooling water
NSRW	nuclear services river water
NSSS	nuclear steam supply system
NTSB	National Transportation Safety Board
NUS	NUS Corporation
OPM	Operations Plant Manual
OTSG	once-through steam generator
P&ID	piping and instrumentation drawing
PCL	panel center left
PCR	panel center right
PDF	probability density function
PDS	plant damage state
PLF	panel left front
PLG	Pickard, Lowe and Garrick, Inc.
PMF	probable maximum flood
PMR	Plant Model Report
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PRF	panel right front
PSHX	primary to secondary heat transfer
PSV	pressurizer safety valve
PWR	pressurized water reactor
RBCU	reactor building cooler unit
RBEC	reactor building emergency cooling
RBD	reliability block diagram
RBS	reactor building spray
RBSS	reactor building spray system
RCDT	reactor coolant drain tank
RCP	reactor coolant pump
RCS	reactor coolant system
RPS	reactor protection system
RSS	Reactor Safety Study
RSSM	Reactor Safety Study Methodology Application Program
SAI	Science Applications, Inc.
SCCW	secondary closed cooling water
SCM	subcooled margin
SGTR	steam generator tube rupture
SLB	steam line break
SLRDS	steam line rupture detection system
SRO	senior reactor operator

LIST OF ACRONYMS (continued)

Abbreviation	Definition
SRV	safety relief valve
SRW	secondary river water
SSCCS	secondary services closed cooling system
SSE	safe shutdown earthquake
SSS	support state system
STA	shift technical advisor
TBV	turbine bypass valve
TLV	toxic limit valve
TMI-1	Three Mile Island Nuclear Generating Station, Unit 1
TMI-2	Three Mile Island Nuclear Generating Station, Unit 2
TPRA	Three Mile Island Probabilistic Risk Assessment
TVA	Tennessee Valley Authority
UCS	The Union of Concerned Scientists
ULD	unit load demand

APPENDIX A

RISK ENGINEERING, INC., REPORT

Seismic Ground Motion Hazard at Three Mile Island Nuclear Generating Station, Unit 1

GPU-768-DOC-250
c.3 - HFPerla

SEISMIC GROUND MOTION HAZARD
AT
THREE MILE ISLAND
NUCLEAR GENERATING STATION, UNIT 1

Prepared for
Pickard, Lowe and Garrick, Inc.

by
Risk Engineering, Inc.
Golden, Colorado

May 31, 1985

TABLE OF CONTENTS

1.0 INTRODUCTION

This report was prepared for Pickard, Lowe and Garrick, Inc., and GPU Nuclear, Inc. The purpose of this study is to make a probabilistic assessment of the frequency of exceedance of various ground acceleration levels at the Three Mile Island Nuclear Generating Station, Unit 1 (TMI-1). The results of this study will be used as input to a Probabilistic Risk Assessment to assess the seismic response of equipment and components in the plant.

Experience and judgment play an important role in guiding assumptions and drawing conclusions for seismic hazard calculations. In addition to our own expertise, the work of Bernreuter et al (1984) summarizes a wide range of opinion and expertise on seismicity in the central and eastern U.S. Other studies of eastern U.S. seismicity include Hadley and Devine (1974), Tera Corp (1980), and numerous other documents included in the list of references. The earthquake catalogs of Chiburis (1981) and the U.S. Geological Survey, updated with more recent information, are the sources of historical earthquake data used here. Figure 1 shows the seismicity in the vicinity of TMI-1 (as derived from references reported in Section 4 of this report).

The formal mathematical procedures used to calculate seismic hazard (described in Section 2) are standard ones for seismic hazard assessment of nuclear power plant safety, as documented in the TERA Corp. (1980) report of the Lawrence Livermore National Laboratory work, in the USNRC Probabilistic Risk Assessment guide (American Nuclear Society, 1981), and in Bernreuter et al. (1984). The computer program used for calculations (McGuire, 1976) is a standard one in the industry and has been used for many seismic risk studies.

The specific facility examined in this study is the Three Mile Island Nuclear Generating Station, Unit 1, Pennsylvania. The assumptions and hypotheses examined are appropriate for this site, but may not be for other sites. As an example, certain alternate configurations of seismogenic zones in the eastern U.S. may be appropriate for the evaluation of seismic hazard at other sites in the eastern U.S. These alternate configurations were not examined here because they would have no appreciable effect on the conclusions drawn for seismic hazard at the Three Mile Island facility.

2.0 SEISMIC HAZARD MODEL

Probabilistic seismic hazard spectra at TMI-1 could be developed in several ways. The most direct is to estimate spectral amplitudes at different frequencies, draw spectra corresponding to preselected frequencies of exceedance, and use these to compute responses of components and equipment. However, because of the lack of strong motion data in the eastern U.S., the estimation of spectral amplitudes requires substantial judgment and is subject to large uncertainties. An alternative procedure is to estimate seismic hazard for various accelerations and to anchor appropriate spectral shapes to these accelerations. This procedure has the advantage that numerous methods have been published to estimate acceleration in the eastern U.S., and spectral shapes can be derived from studies of west coast strong motion data. This is the procedure used in this study.

The seismic hazard model used in this study to estimate frequency-of-exceedance versus ground motion level has been described in detail elsewhere (Cornell, 1968, 1971; McGuire, 1976), and the steps involved are depicted in Figure 2. As shown in Figure 2a, the first step is to delineate zones of potential future earthquake occurrences, using seismicity, geology, and tectonic evidence. For each zone, data on historical earthquake occurrences are gathered, and earthquake magnitudes are estimated from historical earthquake intensities using relationships proposed by Nuttli and Herrmann (1978). The data are plotted to indicate the number of earthquakes per unit time occurring in specific magnitude intervals, as illustrated in Figure 2b. A truncated exponential distribution is assumed to adequately represent the relative frequency of earthquake magnitudes in each zone, and the rate of earthquake occurrence is assumed to be accurately estimated by historical occurrences.

After delineating seismic zones and analyzing earthquake statistics, the third step is to adopt or derive an "attenuation function", shown in Figure 2c. This equation estimates ground motion amplitude (peak acceleration) as a function of earthquake magnitude and distance between the source of seismic energy and the site. It is assumed that the ground motion amplitude predicted by the attenuation function is a median value, and that actual values are lognormally distributed. The final step in the analysis consists of mathematically integrating over all possible earthquake magnitudes and locations, calculating for each magnitude and location the distribution of ground motion at the site, to evaluate the annual frequencies that various levels of ground motion will be exceeded. A standard computer program (McGuire, 1976) is used for calculations. The output from this program is frequency of exceedance (number per unit time) as a function of ground motion level which can be plotted as illustrated in Figure 2d.

Assumptions used in the seismic ground motion hazard analysis are listed in Table 1 for reference. The most basic assumptions are that seismogenic zones can be drawn to represent occurrences of future earthquakes, and that those occurrences can be represented probabilistically using the statistics of historical earthquakes in those zones. These assumptions, while quite gross, yield quite accurate estimates of seismic hazard (see, for example, McGuire, 1979, and McGuire and Barnhard, 1981). These are standard assumptions for seismic hazard analyses in regions where tectonic faults cannot be identified at the earth's surface.

There are several assumptions required to describe seismicity within each seismogenic zone. The first is that successive earthquakes are independent in time, location, and size. This means that the frequency of occurrence of an earthquake at a specific location in any year is not affected by seismicity (or lack of it) in prior years in the same general area. While this is physically unrealistic (any physical explanation of seismic events would account for the release of crustal stress, making future events at the same location unlikely in the short term), there are simply not enough data available in the short historical record to justify or calibrate more sophisticated models. Also, the readjustment of crustal stresses during major earthquakes means that events of similar size are possible at adjacent locations without any quiescent period; the New Madrid earthquake series of 1811-1812 is a good example of this process. Comparisons in areas where longer historical records are available indicate that the independence assumption is accurate if we are interested in estimating seismic hazard for periods on the order of 50 years (see the aforementioned references). Finally, we observe that derived ground motion levels are not very sensitive to errors in frequencies of occurrence: an error in frequency of occurrence of a factor of two implies an error in ground motion amplitude of only about thirty percent. Thus, we can misestimate earthquake frequencies by a large amount and only expect a relatively small error in the associated ground motion amplitude.

The typical probability distribution used to represent earthquake size is the double-truncated exponential distribution. This is an accurate representation of historical seismicity data; its use to characterize future seismicity is appropriate if (as is the case in the eastern United States) no change in the character of tectonic strain accumulation or release is suspected. Parameters required to define this distribution are the lower bound, upper bound, and b-value. A lower-bound body-wave magnitude m_b of 4.5 was used in this study, based on the observation that earthquakes smaller than this are not known to cause damage to engineered structures. In fact, this may be a conservative assumption: if, for example, seismic events in the range of 4.5 to 5.0 could also be shown to cause no damage, regardless of the ground motions generated, they should also be excluded from consideration. At present, such a demonstration is not possible quantitatively. The upper-bound magnitude is a realistic representation, based on all seismologic, geologic and geophysical data available. The method used in this study to examine the b-value is described below.

The random process used to represent earthquake occurrences in time is not critical to seismic hazard results. The levels of ground motion and their frequencies are such that only the mean rate of activity (number of earthquakes per year) is important. Thus the selection of a Poisson (or other) process does not seriously affect the results.

Other assumptions required in the analysis are that ground motion levels can be represented as a function only of earthquake magnitude and source-to-site distance, and that the uncertainty in predicted ground motion can be represented by a lognormal distribution with a logarithmic standard deviation of 0.6. Both assumptions are standard for this type of analysis, and are appropriate to characterize available earthquake ground motion data.

On balance, the assumptions used in seismic hazard analyses provide realistic estimates for the frequency of occurrence of peak ground acceleration. Not considered explicitly here are conservatisms associated with assuming that damage to structures is well-related to peak acceleration. This ground motion parameter is used here to anchor standard response spectrum shapes, not as a measure of earthquake-induced damage.

The response spectrum recommended to characterize earthquake ground motion for the rock and stiff soil foundation conditions existing at TMI-1 is anchored to peak acceleration at high frequencies. The appropriate spectrum is one such as derived by Newmark and Hall (1982) which represents a median, broad-banded spectrum for moderate earthquakes (M_L = 6.5, m_b = 6.3). Events of this size generally dominate the hazard calculations for the high accelerations (about 0.5g and higher) which dominate the hypothesized core melt and radionuclide release sequences.

3.0 SEISMOGENIC ZONES

The seismic hazard analysis requires the delineation of seismogenic zones, within which earthquakes are considered to be of similar tectonic origin so that future seismic events can be modeled by a single function describing earthquake occurrences in time, space, and size. Several sets of seismogenic zones were adopted from Bernreuter et al (1984), each set representing a different hypothesis on the crustal stress mechanisms causing earthquake occurrences in the vicinity of the site. These sets of zones represent a reasonable range of the types and sizes of seismogenic zones which govern earthquake occurrences in the eastern U.S. In our experience, the seven sets of zones used here encompass the general range of zones that might be used to represent seismicity in the vicinity of TMI-1. Thus it is not necessary to model other similar sets of zones, including the zones used by other experts in the Bernreuter et al. study. The seismic zones located within several tens of km of a site dominate the hazard except in special cases, so it is not necessary to model zones which are more distant, particularly if they are relatively aseismic.

3.1 SEISMOGENIC ZONATION NO. 1

This zonation was adopted from seismicity expert No. 1 in Bernreuter et al (1984), and is shown in Figure 3. This expert used a large zone encompassing the central and southern Appalachian mountains to govern seismicity; TMI-1 lies within this zone. The largest historical earthquake in this zone is the 1897 shock which occurred in Giles County, Virginia, with Modified Mercalli (MM) intensity VIII.

3.2 SEISMOGENIC ZONATION NO. 2

Expert 2 of the Bernreuter et al (1984) study provided a different set of zones which were adopted for this study (Figure 4). In this case the central Appalachians were modeled with one zone, and a separate, smaller zone was used to represent seismicity in eastern Pennsylvania, Virginia, Maryland, Delaware, and New Jersey. The TMI-1 site lies within the latter zone. The largest historical events in this smaller zone are several MM intensity VII earthquakes which occurred in the 19th century and in the 1920's.

3.3 SEISMOGENIC ZONATION NO. 3

The third set of zones used in this study were adopted from expert No. 4 of Bernreuter et al (1984). In this case, seismicity in the central Atlantic states was modeled using three northeast-trending zones plus a fourth zone around Giles County, Virginia (see Figure 5) which does not contribute to seismic hazard at TMI-1. The site lies in the most eastern zone of those shown in Figure 5; the largest historical events are several MM intensity VII shocks which occurred in the 19th century.

3.4 SEISMOGENIC ZONATION NO. 4

Zonation No. 4 (Figure 6) was adopted for this study from the zones of expert No. 6 in the Bernreuter et al (1984) study. This expert represented seismicity in the vicinity of the site with two zones, one encompassing eastern Pennsylvania (and including the site), the other encompassing central Virginia. In the former zone, the largest historical events are several MM intensity VII earthquakes.

3.5 SEISMOGENIC ZONATION NO. 5

Expert 10 of Bernreuter et al (1984) chose to zone the central Atlantic states with a seismogenic zone which extends from central Virginia to eastern New York, and with a separate zone east of that which comprises the coastal plain excluding the Charleston, South Carolina, area (see Figure 7). The site lies within the latter zone but near the border between the two. In both zones the largest historical event is of MM intensity VII.

3.6 SEISMOGENIC ZONATION NO. 6

This zonation was adopted from expert No. 11 of Bernreuter et al (1984). In the vicinity of the TMI-1 site, this expert used a single zone encompassing parts of Virginia, much of eastern Pennsylvania, northern New Jersey and parts of New York, Connecticut and Massachusetts. The largest historical events in this zone are several MM intensity VII earthquakes which occurred in the 19th century.

3.7 SEISMOGENIC ZONATION NO. 7

The last zonation considered here is that of expert No. 12 of the Bernreuter et al (1984) study. This expert used a large zone extending from the southern Appalachians northeast to new Brunswick to represent seismicity in eastern North America, and an adjacent zone to the west (see Figure 9). The site lies in the former zone near its border with the latter zone. The largest historical events in the large zone are of MM intensity VIII or m_b = 5.8, most recently in 1982 in New Brunswick. In the smaller zone the largest event is an MM intenisty VII shock which occurred in 1954 and which is thought to be related to mining activity in Wilkes-Barre, Pennsylvania.

3.8 SUBJECTIVE WEIGHTS ON ZONES

For the purpose of deriving the relative likelihood associated with hazard curves, subjective weights were assigned to the sets of seismogenic zones described above. These sets of zones represent a range of interpretation of seismogenic potential in the eastern U.S., from relatively small zones around the site (zonation Nos. 2 and 4) to broad, continental-scale zonations (Nos. 1 and 7). There is no apparent reason to prefer one zonation over another with respect to seismic hazard at TMI-1; therefore we choose to assign equal credibilities to each zonation (a relative weight of 0.143 each).

4.0 SEISMICITY PARAMETERS

For the probabilistic calculation of seismic hazard, several parameters describing seismicity are required for each seismogenic zone. These parameters, and the methods used to estimate mean values and to quantify uncertainty, are discussed below. The seismicity data, base was obtained from Chiburis (1981) the U.S. Geological Survey (published as state seismicity maps, for example Reagor et al., 1980), Bollinger and Sibol (1983), and Dewey and Gordon (written communication, 1983). For statistical data analysis, earthquakes with an epicentral MM intensity I_e but without a magnitude estimate (predominantely pre-instrumental seismicity) were converted to a body-wave magnitude m_b using the relationships (Nuttli and Herrmann, 1978):

$$m_b = 1.75 + 0.5\ I_e \qquad (1)$$

Equation 1 was derived for the central U.S. and is considered reliable for the eastern U.S. as well (Bollinger, personal communication, 1983).

4.1 RICHTER b-VALUE

The Richter b-value describes the slope of the log-number versus magnitude relation:

$$\log 10\ n(m_b) = a- bm_b \qquad (2)$$

where $n(m_b)$ is the annual number of earthquakes of body-wave magnitude m_b, and a and b are parameters fit to seismicity data. Parameter a is related to the seismic activity rate as discussed in Section 4.2.

The b-value from equation (2) was determined from historical data analyzed in the manner described in the next section. The method of maximum-likelihood (Weichert, 1980) was used to calculate the b-value from historical data. The calculated values for b were generally within one standard deviation of 0.9, leading us to use that value as a best estimate for all zones. (This value typically was specified, for example, by many of the experts in the Bernreuter et al, 1984, study). Uncertainty in the b-value was modeled by modifying the best estimate by ± 15%, a typical coefficient of variation. Weights assigned to the best estimate and alternatives were 0.4 and 0.3 each, respectively.

4.2 SEISMIC ACTIVITY RATE

The rate of earthquake occurrence was determined for each seismogenic zone by the maximum likelihood method (Weichert, 1980), using as data the historical earthquakes in that zone. Magnitudes, when not reported in seismicity catalogs, were estimated from MM intensities using equation 1, and the number of events per decade were counted into magnitude intervals centered on magnitudes estimated from even MM intensity values. Periods of historical completeness were determined in a manner designed to give the highest observed rate of occurrence; these were generally as follows:

Magnitude (m_b) (Equation 1)	MM Intensity	Period
3.3	III	1980-present
3.8	IV	1950-present
4.3	V	1950-present
4.8	VI	1950-present
5.3	VII	1900-present
5.8	VIII	1800-present

Where alternate intervals gave higher observed rates of seismicity (due, for example, to incompleteness in earlier times), those higher rates were used. Activity rates were calculated for occurrences of earthquakes with $m_b \geq 4.5$ (where m_b is body-wave magnitude) which corresponds to MM intensity V-VI. This lower bound was based on the observation that earthquakes of smaller magnitude rarely cause structural damage, even if peak accelerations are high, due to the short duration, impulsive-type ground motions associated with these small events. The activity rates calculated for each zone are shown in Table 2.

Uncertainty from the maximum likelihood method of determining activity rates using the macroseismic data (historical catalog) is relatively small, because of the several tens of earthquakes used to estimate the rate. However, to account for uncertainty in intensity-to-magnitude conversion, stationarity assumptions, and incompleteness of the historical catalog, multiplicative factors of 2 and 1/2 were used to represent 10% and 90% confidence bounds on the activity rate for each zone. These values were assigned subjective weights of 0.2 each, with 0.6 weight given to the best estimate. These choices are based on our judgment and experience on values that other analysts might derive, given the same set of data.

4.3 MAXIMUM MAGNITUDE

The maximum magnitude earthquake $m_{b,max}$ assumed for each seismogenic zone was chosen to be 0.5 magnitude (m_b) units above the largest historical event. This is a value typical of those indicated by the seismicity experts in both the Bernreuter et al (1984) and the TERA (1980) studies; it corresponds to about one intensity unit above the maximum historical event. These best estimate values are shown in Table 2.

Uncertainty in $m_{b,max}$ was represented by varying the best estimate value by ± 0.5 magnitude units (this corresponds to ± one intensity unit). These alternate values were considered to be one standard deviation values, and were assigned a subjective weight of 0.3 each. The best estimate value was assigned a subjective weight of 0.4. Uncertainties in $m_{b,max}$ account for hypotheses that large earthquakes (equivalent, for example, to the 1886 Charleston event) may occur in regions that have not experienced these shocks during historic times.

5.0 ESTIMATION OF SEISMIC GROUND MOTION

5.1 ACCELERATION

Estimates of peak single-component horizontal ground acceleration, a_g, were made for this study using three methods. These are described in the following paragraphs.

The first attenuation equation is from Nuttli (1983):

$$a_g = \exp\left(1.3158 + 1.15\ m_b - 0.833\ \ln\left(\sqrt{\Delta^2 + h^2}\right) - 0.0028\Delta\right) \quad (3)$$

where Δ is epicentral distance and h is focal depth. Two estimates of focal depth were examined:

$$h = 10\ \text{km} \quad (4)$$

$$h = 10\ (-1.730 + 0.456\ m_b) \quad (5)$$

where the latter equation is a minimum depth designated by Nuttli (1983).

Equation 3 is plotted in Figures 10 and 11 for the two depth estimates. There is not a large difference between the two except at close distances. The two depth estimates are weighted equally in hazard calculations because (a) equation 5 from Nuttli is a minimum depth estimate and therefore is conservative for the smaller magnitudes, and (b) the hazard for the two depth estimates is not greatly different.

The second acceleration equation was derived from intensity data reported for the 1944 Cornwall-Massena earthquake (m_b = 5.8). First, accelerations were estimated from site intensity I_s for this event using the relation (McGuire, 1984):

$$\ln a_g = -.430 + 232\ I_s - .968\ \ln R \quad (6)$$

where R is hypocentral distance. Next, an equation was fit to these estimated acceleration values using least-squares regression analysis. This produced:

$$\ln a_g = 2.405 - 1.30 \ln R - .00012 R \qquad (7)$$

Finally, it was assumed that scaling of acceleration to magnitudes other than 5.8 could be accomplished by a factor equal to exp $(1.1\ m_b)$. This led to the final equation:

$$a_g = \exp(2.91 + 1.1\ m_b - 1.30 \ln R - .00012\ R) \qquad (8)$$

which is herein designated the "Cornwall-Massena" attenuation. This equation, assuming a focal depth of 10 km, is plotted in Figure 12.

The third equation is taken from work of Atkinson (1984) for eastern North America:

$$a_g = 61.9 \exp(.673\ m_b)\ R^{-1} \exp(-.001\ R) \qquad (9)$$

Equation 9 is plotted in Figure 13.

For calculation of seismic hazard, all three attenuation equations were used. The Nuttli (1983) and Atkinson (1984) equations were developed from theoretical considerations of wave propagation, but consider the ground-motion estimation problem from different viewpoints. The Cornwall-Massena attenuation estimates ground motions from MM intensity observations, a different procedure. For hazard calculations the three methods are given equal weight (0.333 each); the weight assigned to the Nuttli equation is equally divided between the two depth estimates (equations 4 and 5).

For calculations of seismic hazard, a lognormal distribution of acceleration about the mean value was assumed, with a standard deviation of $\ln a_g$ equal to 0.6, corresponding to a factor of 1.8 uncertainty in the estimate. This distribution is widely used to represent uncertainty in ground motion estimates. The uncertainty modeled is typical of the scatter exhibited by strong motion data sets, when the data are restricted to a specific area. The uncertainty chosen is typical of that expressed by attenuation experts in the Bernreuter et al (1984) study.

The distribution of peak ground acceleration was truncated to reflect the notion that, for a given-sized earthquake, "effective" peak accelerations must be limited. Whether or not instrumental peak accelerations are limited is problematic; the idea is that a small or moderate earthquake can only cause a limited amount of damage to real structures. The bounds used in this analysis are shown in Table 3.

The third column of Table 3 shows upper bound values of sustained acceleration, where this corresponds to the third highest peak. These upper bound values for MM intensity VI, VII, VIII, and IX were obtained from R. P. Kennedy (personal communication, 1981). The values of sustained acceleration shown in Table 3 for half values of MM intensity were derived by observing that a decrease of sustained acceleration of 20 percent for each half intensity unit is consistent with the limits suggested by Kennedy. These limits on sustained acceleration must be multiplied by the factor 1.25 to convert to a peak acceleration. The basis for this factor is experience with the relationship between sustained ground motion which causes damage by several cycles of induced motion, and the associated peak acceleration for earthquakes of large enough magnitude (>6) to cause long durations of strong shaking (Kennedy, personal communication, 1981).

These limits were applied to all calculations of seismic hazard in this study. For example, in the numerical integration over magnitude, the occurrence of magnitude 6 (corresponding to MM intensity VIII-IX) implies that the resulting distribution of peak accelerations was truncated at 0.8g, as shown in Table 3. If $m_{b,max}$ is 6 for the zone dominating hazard at TMI-1, the calculated annual frequencies of exceedance of peak accelerations greater than 0.8 g is zero.

6.0 RESULTS OF ANALYSIS

Figure 14 shows the calculated annual probability of exceedance at TMI-1 for Zonation No. 1, the best estimates of seismicity parameters, and the four versions of the acceleration attenuation equations (the Nuttli, 1983, equation with two focal depths and the other two equations with focal depths of 10 km). The results of different focal depth assumptions for the Nuttli equation are negligible.

The sensitivity to zonation is shown in Figure 15. There is some dependence of calculated hazard on the zones used; this results from the location of TMI-1 in a zone with a low $m_{b,max}$ and low upper-bound acceleration in some zonations, and in a zone with a high $m_{b,max}$ and high upper-bound acceleration in other cases.

Figure 16 illustrates the sensitivity of seismic hazard to $m_{b,max}$ for Zonation No. 1 using the Nuttli (1983) attenuation with 10 km focal depth. There is substantial sensitivity to $m_{b,max}$, particularly at the higher accelerations, resulting from the dependence of upper-bound acceleration on $m_{b,max}$.

Figure 17 shows sensitivity to b-value, again for zonation no. 1, the Nuttli attenuation with 10 km depth, and best-estimate values of $m_{b,max}$. The influence of the b-value is small at all ground motion levels, relative to the other parameters influencing the hazard calculations.

In all, 756 seismic hazard curves for acceleration were generated in this study: seven sets of zonations, times four attenuation functions, times three activity rates, times three values of $m_{b,max}$, time three b-values. These results were aggregated into ten representative hazard curves, with weights equal to the sum of weights of the original curves which make up each aggregate. These aggregate curves are shown in Figure 18.

Numerical results corresponding to the eight aggregate curves of Figure 18 are presented in Table 4. These are in the form of annual frequency of exceedance as a function of peak acceleration.

7.0 SUMMARY

We present here a seismic hazard analysis for peak ground acceleration at the Three Mile Island Nuclear Generating Station, Unit 1. The analysis is primarily dependent on the attenuation equations used, on seismic activity rates, and on the maximum magnitudes assumed. For the purposes of reporting, eight aggregate hazard curves are obtained. These curves illustrate the range and uncertainty in results obtained from uncertainties in seismicity and attenuation.

Uniform hazard response spectra can be estimated by anchoring a standard spectral shape to the peak ground accelerations reported here. These spectra should be broad-banded for accelerations above about 0.5g, because such ground motions are generally caused by moderate-sized earthquakes (m_b = 6.3, M_L = 6.5).

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TABLE 1
GENERAL ASSUMPTIONS USED IN SEISMIC HAZARD ANALYSIS

Assumption	Comment
1. Earthquake locations represented by seismogenic zones with a homogeneous location distribution	1. In general, reasonable. Conservative for sites located away from active fault zones, unconservative for sites located near active fault zones.
2. Historic earthquake magnitudes can be estimated by Modified Mercalli intensities.	2. Reasonable
3. Truncated exponential distribution represents earthquake sizes.	3. Reasonable
4. $m_{b,max}$ represents largest earthquakes.	4. In general, reasonable. Conservative for zones with lower $m_{b,max}$; unconservative for zones with higher $m_{b,max}$.
5. Rate of activity represented by historic rate of occurrence.	5. Reasonable
6. Peak acceleration estimated by attenuation function as a function of magnitude and distances.	6. Reasonable
7. Uncertainty in peak acceleration represented by lognormal distribution with $_{\ln a} = 0.6$.	7. Reasonable

TABLE 2

SEISMOGENIC ZONES AND ASSOCIATED PARAMETERS

Seismogenic Zonation	Zone	Max. Hist. Earthquake (m_b)	Best Estimate $m_{b,max}$	Best Estimate Activity Rate (m_b>4.5)	Best Estimate b-value
No. 1	4*	5.8	6.3	0.22	0.9
No. 2	27	5.8	6.3	0.16	0.9
	28*	5.3	5.8	0.045	0.9
No. 3	8	5.3	5.8	0.12	0.9
	11*	5.5	6.0	0.080	0.9
	12	5.0	5.5	0.038	0.9
No. 4	7*	5.0	5.5	0.058	0.9
	8	5.5	6.0	0.059	0.9
No. 5	4*	5.5	6.0	0.10	0.9
	5	5.0	5.5	0.071	0.9
No. 6	5*	5.3	5.8	0.088	0.9
No. 7	3*	5.8	6.3	0.70	0.9
	4	5.0	5.5	0.070	0.9

* Host zone for this zonation

TABLE 3

BOUNDS ON EFFECTIVE PEAK ACCELERATION

MM Intensity	Corresponding Value of $m_{b,max}$ (Equation 1****)	Upper Bound Sustained Acceleration	Upper Bound Peak Acceleration (g)***
VI	4.8	0.20*	0.25
VI-VII	5.0	0.25**	0.30
VII	5.3	0.30*	0.37
VII-VIII	5.5	0.40**	0.50
VIII	5.8	0.50*	0.62
VIII-IX	6.0	0.65**	0.80
IX	6.3	0.80*	1.00

* From R. P. Kennedy, Personal Communication, 1981

** Obtained by interpolation

*** Calculated as 1.25 times the Upper Bound Sustained Acceleration (see text)

**** See Section 4.1

TABLE 4
ANNUAL FREQUENCIES OF EXCEEDANCE FOR AGGREGATE CURVES

AGGREGATE CURVE NO.	WEIGHT	ANNUAL FREQUENCY FOR 0.1g	0.2g	0.3g	0.4g	0.5g	0.6g	0.7g	0.8g.	0.9g	1.0g
1	.100	1.1×10^{-3}	1.1×10^{-4}	3.6×10^{-6}	0	0	0	0	0	0	0
2	.033	1.5×10^{-3}	2.2×10^{-4}	3.3×10^{-5}	3.4×10^{-6}	2.2×10^{-10}	0	0	0	0	0
3	.138	1.3×10^{-3}	1.9×10^{-4}	3.8×10^{-5}	8.0×10^{-6}	1.1×10^{-6}	0	0	0	0	0
4	.141	8.3×10^{-4}	1.3×10^{-4}	2.9×10^{-5}	7.3×10^{-6}	1.7×10^{-6}	2.9×10^{-7}	3.6×10^{-8}	0	0	0
5	.182	8.5×10^{-4}	1.4×10^{-4}	3.7×10^{-5}	1.3×10^{-5}	5.0×10^{-6}	2.0×10^{-6}	8.4×10^{-7}	3.3×10	1.2×10^{-7}	5.1×10^{-8}
6	.052	1.3×10^{-3}	1.8×10^{-4}	2.6×10^{-5}	2.8×10^{-6}	0	0	0	0	0	0
7	.147	1.7×10^{-3}	3.4×10^{-4}	1.0×10^{-4}	3.9×10^{-5}	1.7×10^{-5}	7.5×10^{-6}	3.4×10^{-6}	1.6×10^{-6}	7.3×10^{-7}	3.9×10^{-7}
8	.086	7.4×10^{-4}	9.7×10^{-5}	1.5×10^{-5}	1.7×10^{-6}	0	0	0	0	0	0
9	.074	2.1×10^{-3}	3.9×10^{-4}	1.0×10^{-4}	3.2×10^{-5}	9.9×10^{-6}	2.6×10^{-6}	3.6×10^{-7}	0	0	0
10	.047	3.5×10^{-3}	7.7×10^{-4}	2.6×10^{-4}	1.0×10^{-4}	4.8×10^{-5}	2.3×10^{-5}	1.1×10^{-5}	6.0×10^{-6}	3.3×10^{-6}	1.9×10^{-6}
	1.0										

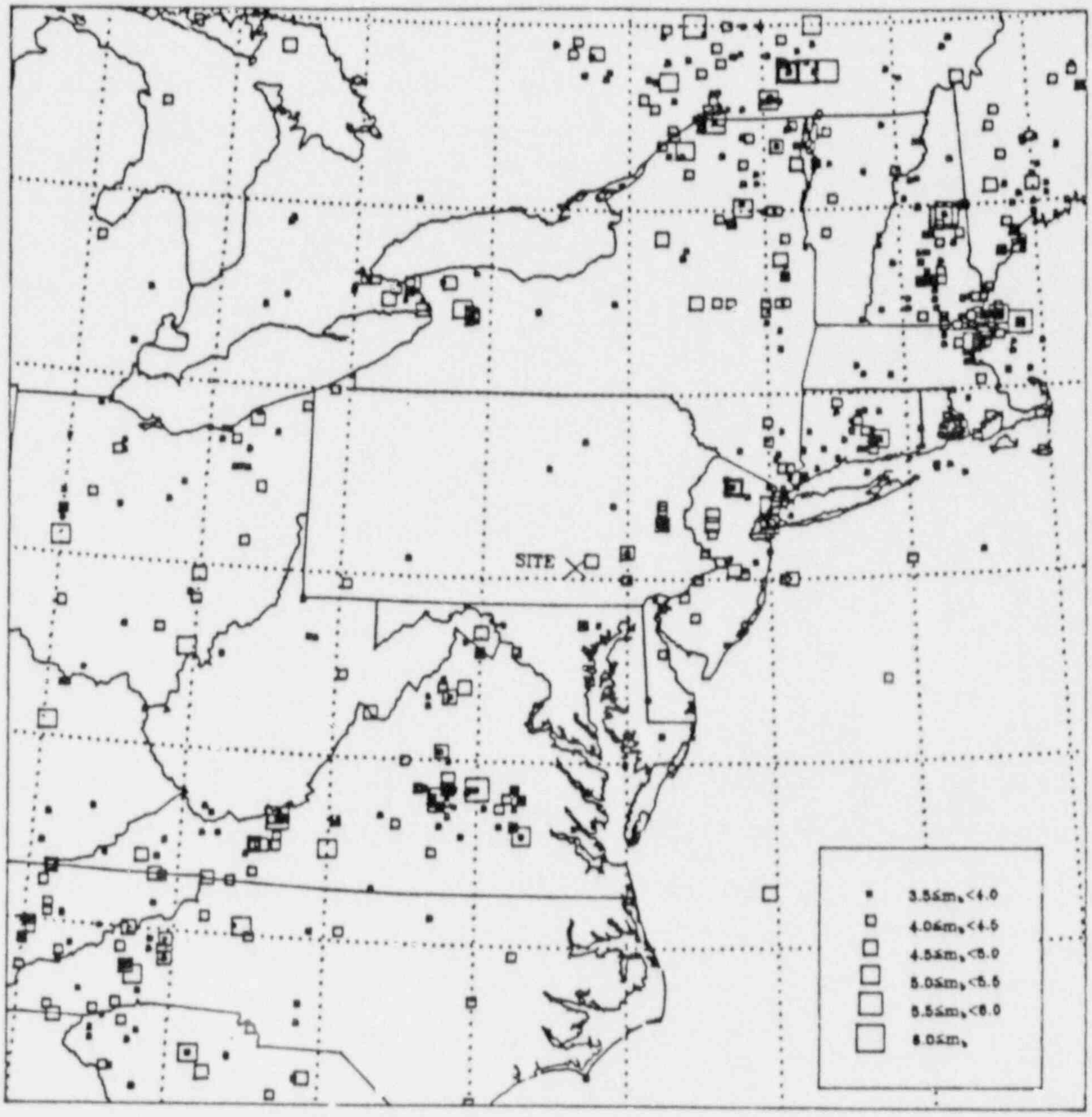

FIGURE 1

SEISMICITY IN THE VICINITY OF TMI-1

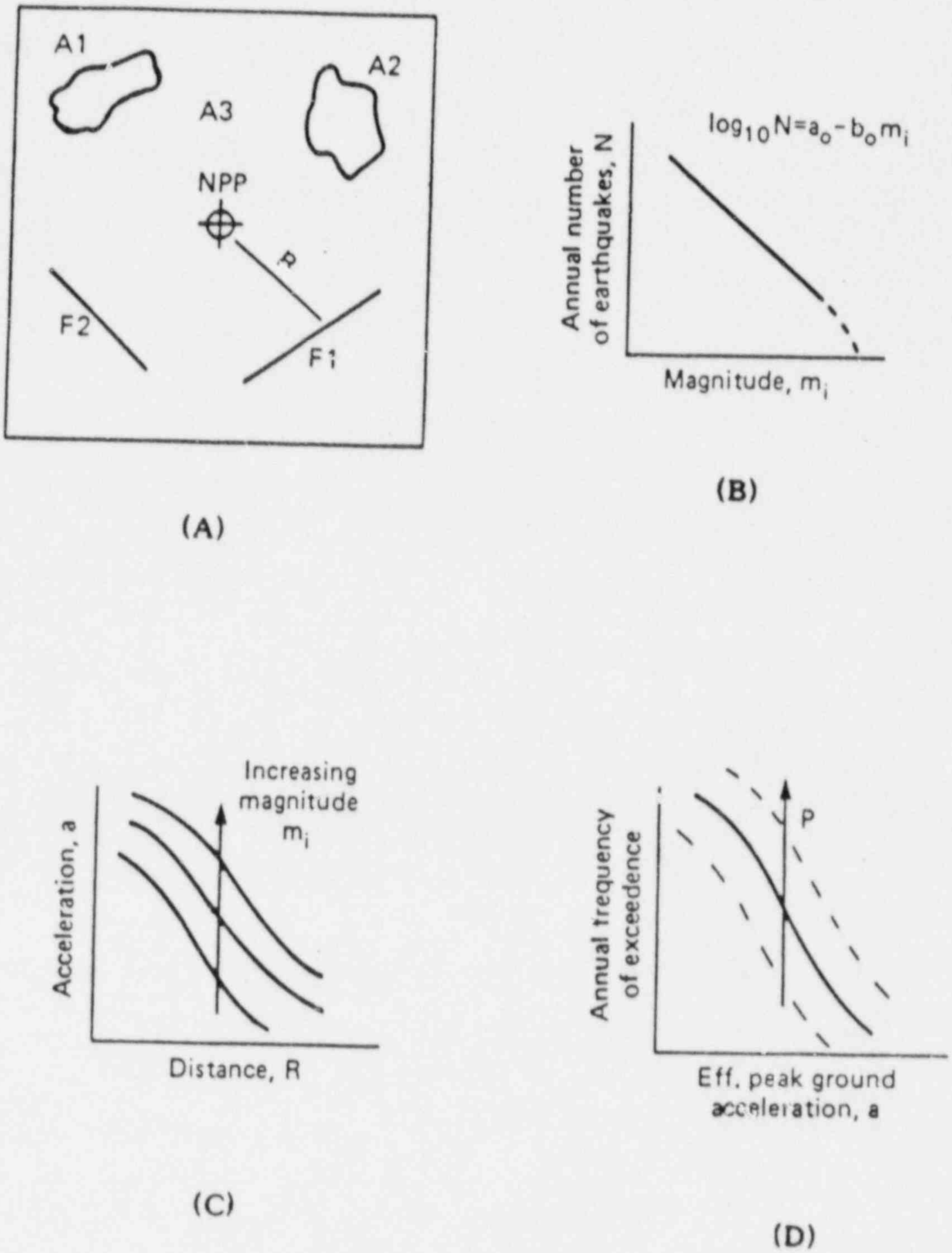

FIGURE 2

CONCEPTUAL REPRESENTATION OF STEPS INVOLVED IN SEISMIC HAZARD EVALUATION
(After American Nuclear Society, 1981)

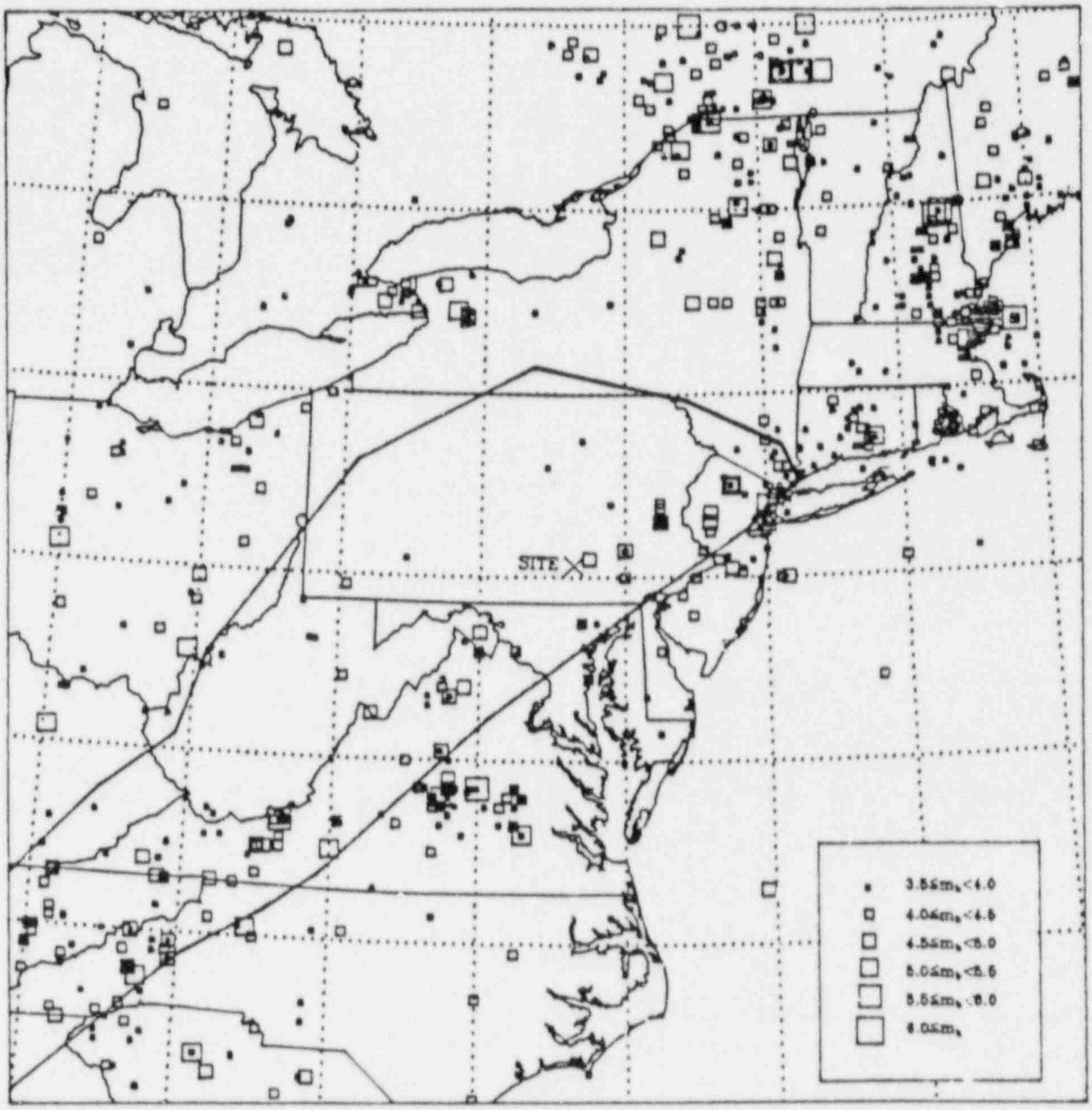

FIGURE 3

SEISMOGENIC ZONATION NO. 1

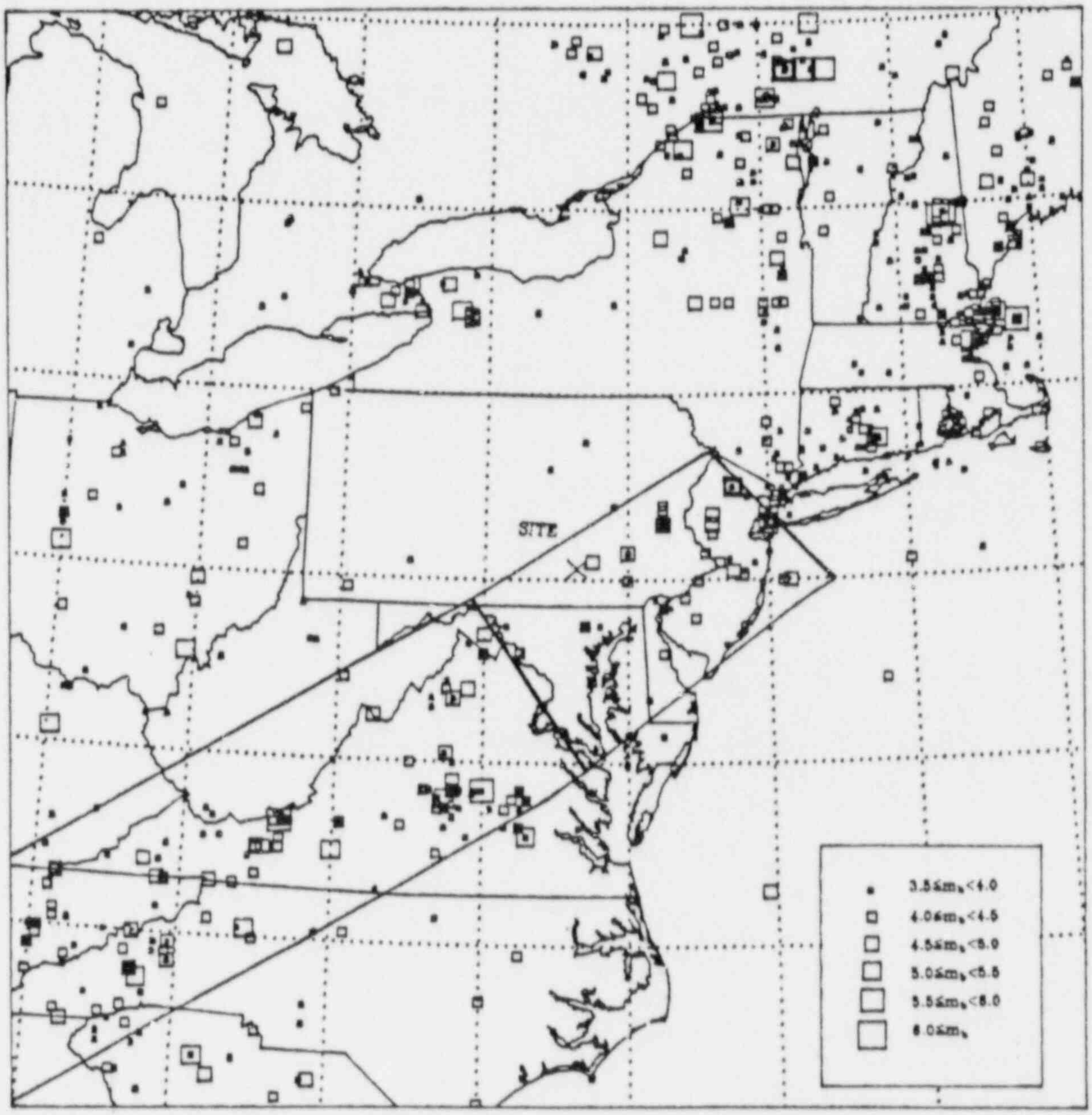

FIGURE 4

SEISMOGENIC ZONATION NO. 2

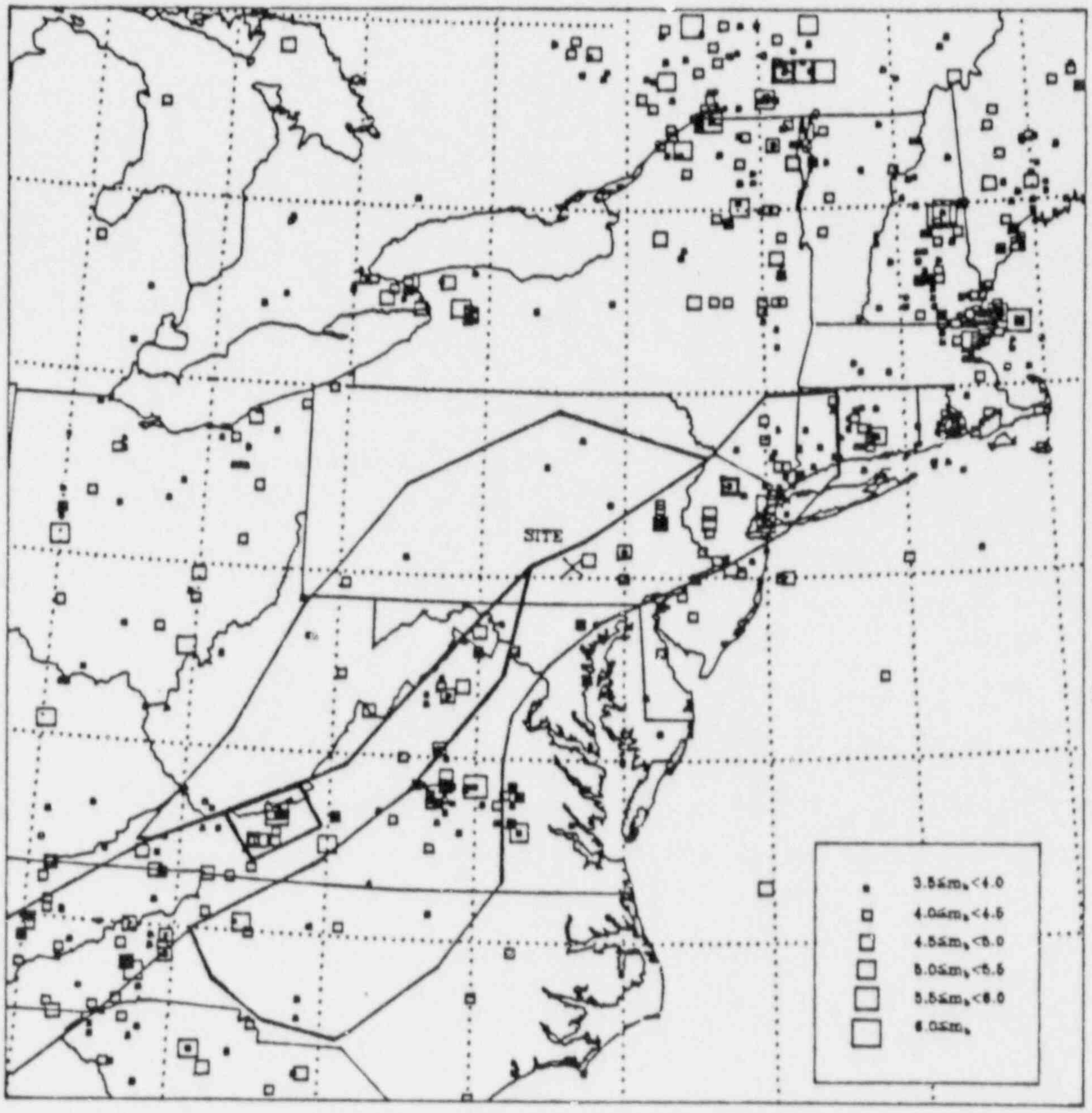

FIGURE 5

SEISMOGENIC ZONATION NO. 3

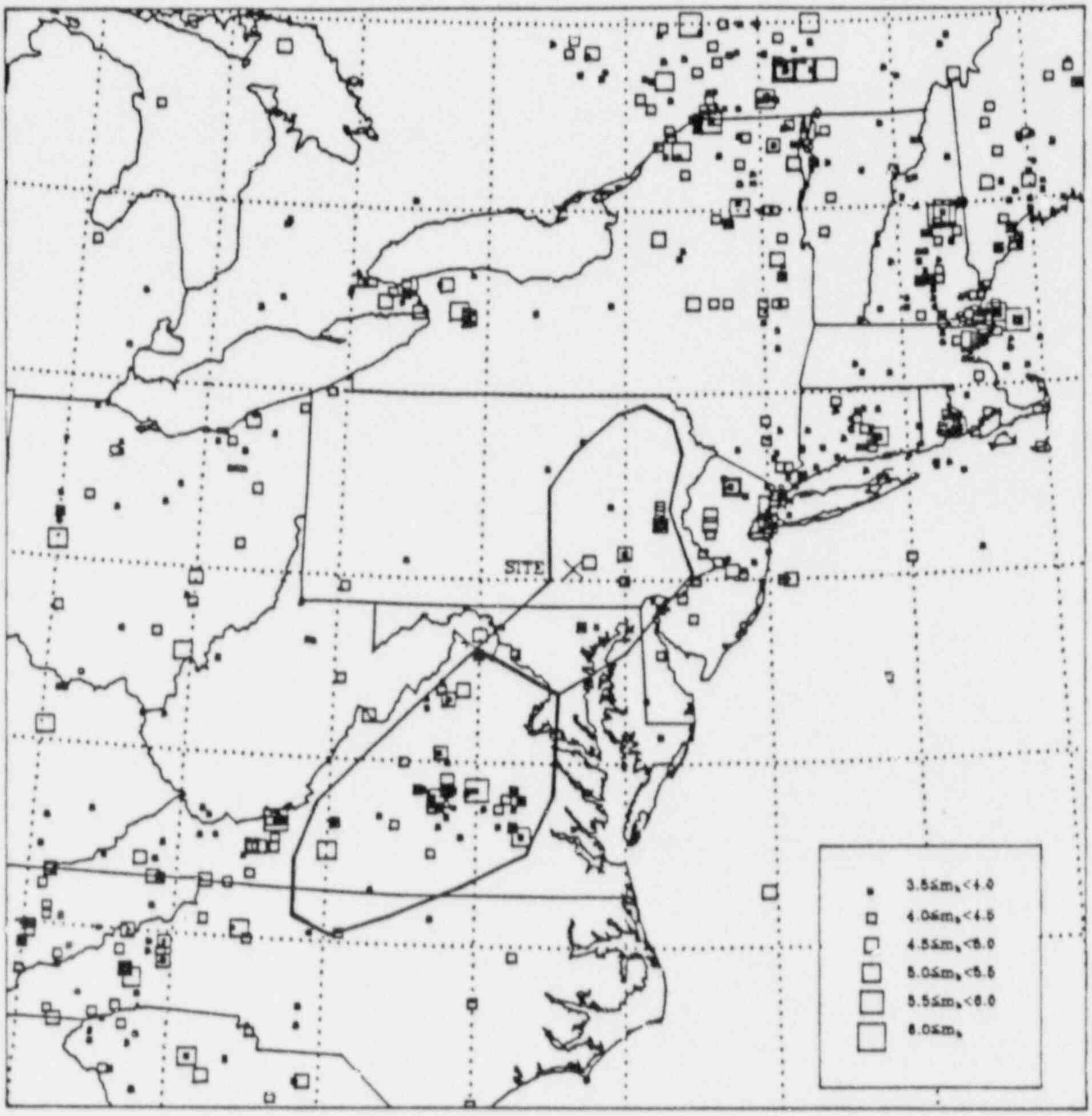

FIGURE 6

SEISMOGENIC ZONATION NO. 4

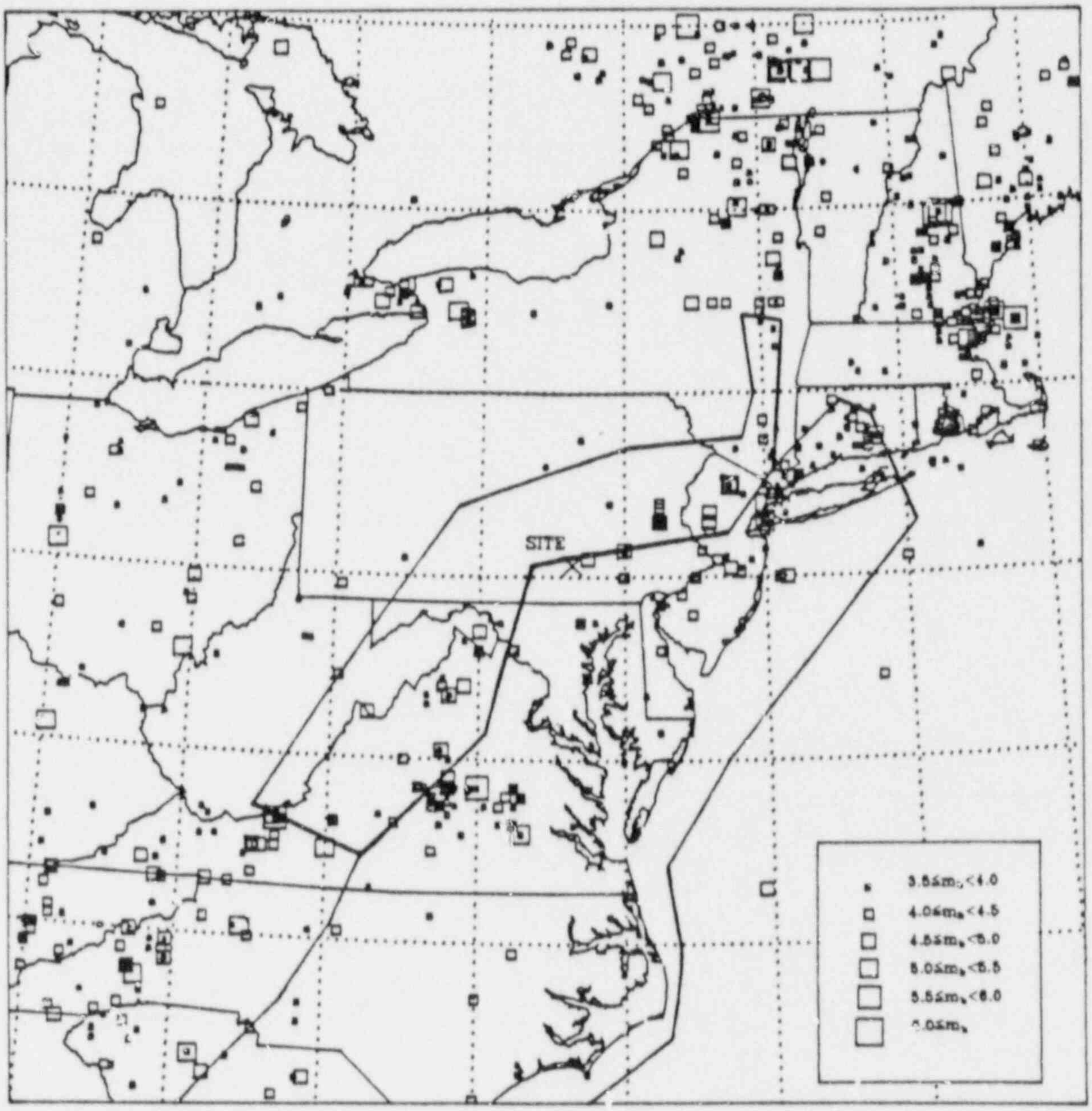

FIGURE 7

SEISMOGENIC ZONATION NO. 5

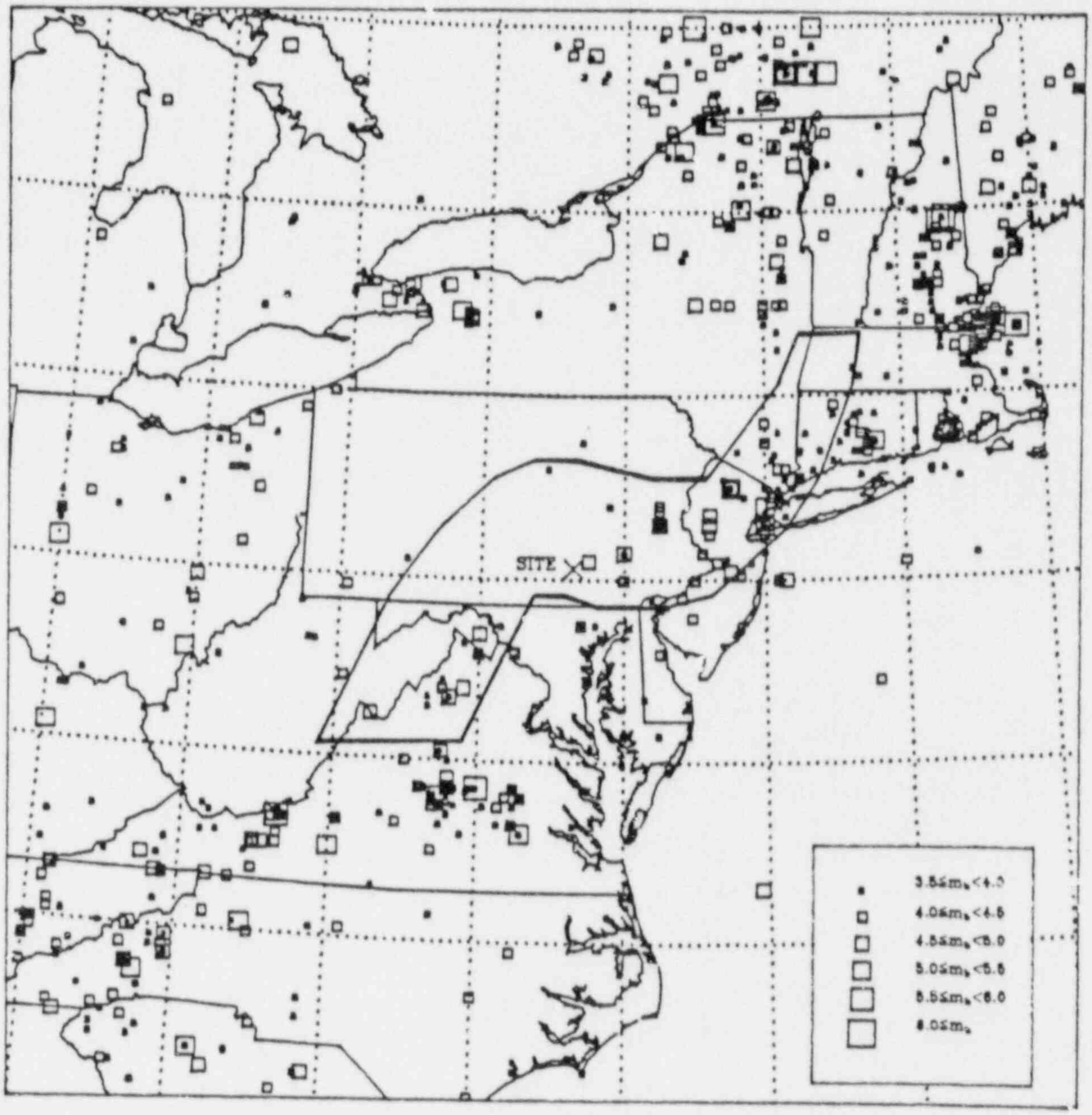

FIGURE 8

SEISMOGENIC ZONATION NO. 6

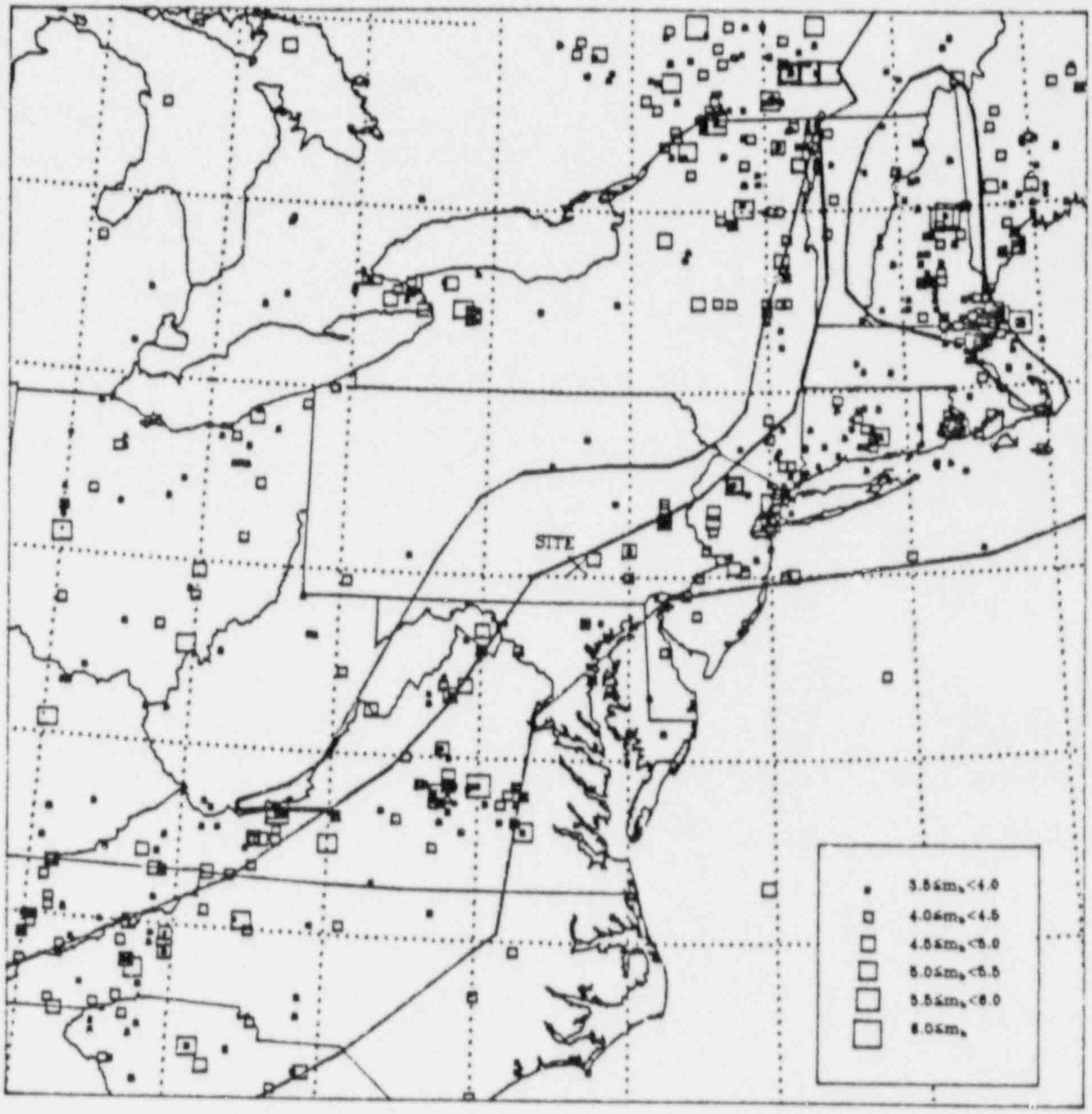

FIGURE 9

SEISMOGENIC ZONATION NO. 7

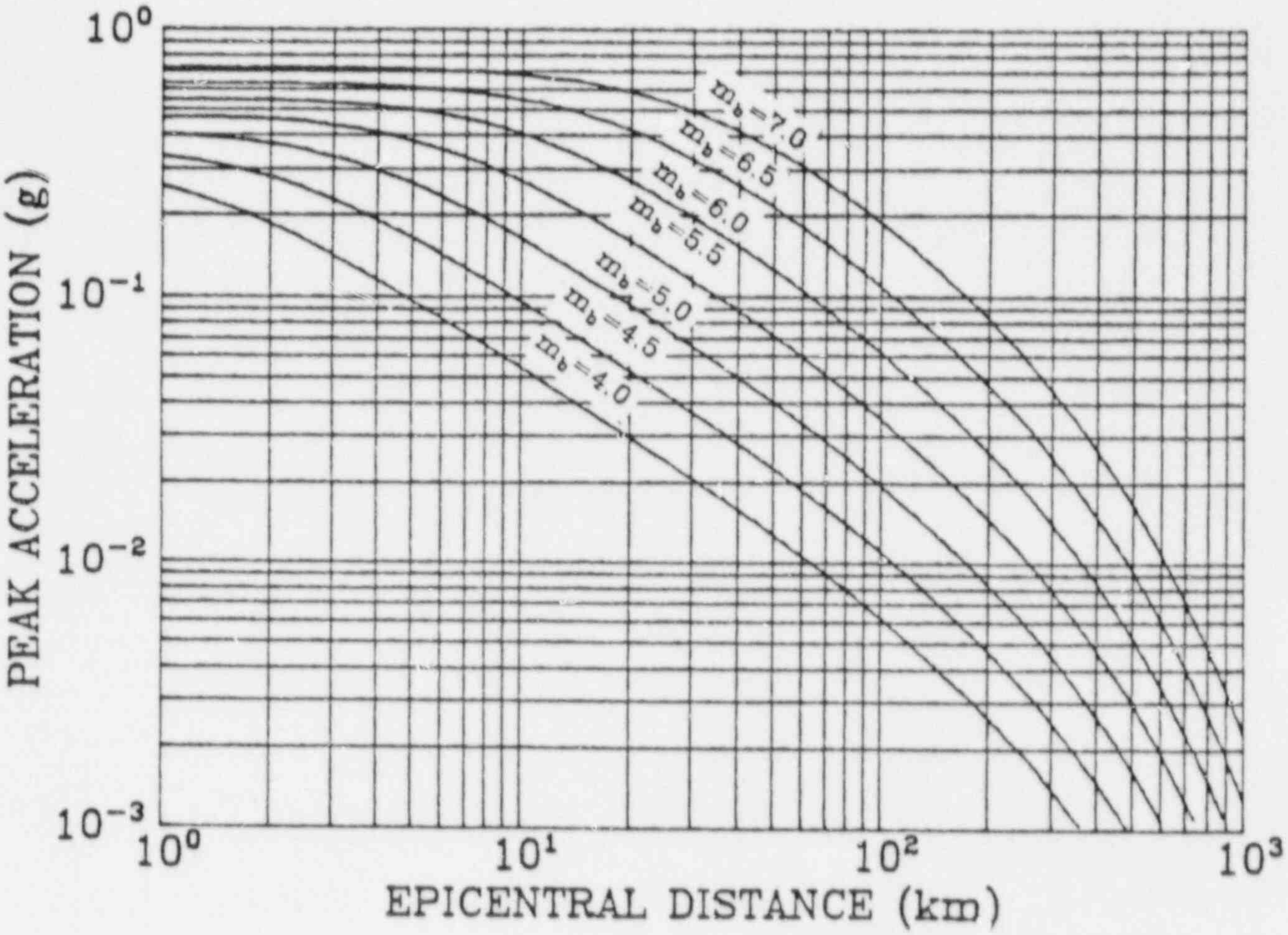

FIGURE 10

PEAK ACCELERATION ESTIMATES FROM NUTTLI (1983), USING VARIABLE FOCAL DEPTH.

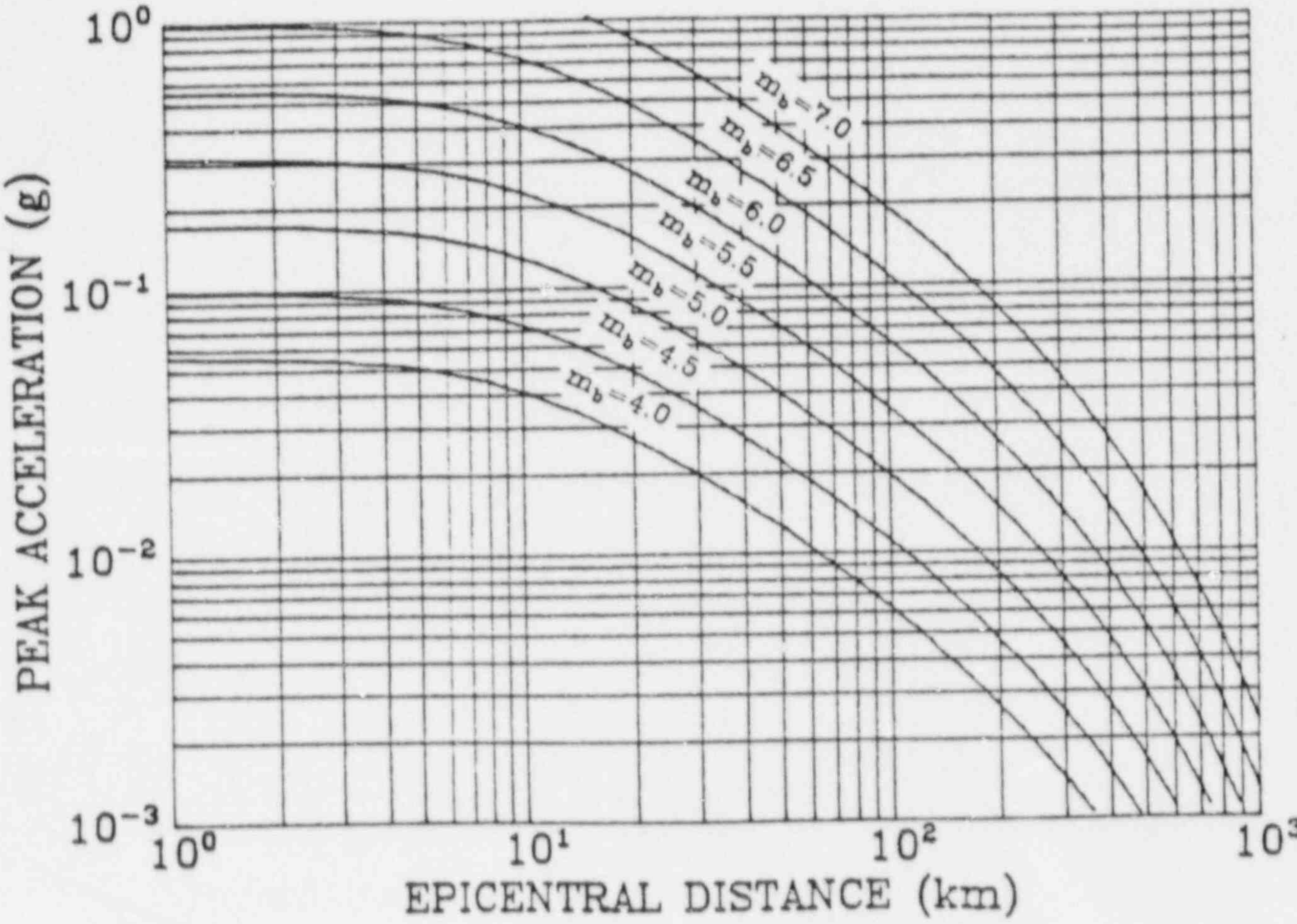

FIGURE 11

PEAK ACCELERATION ESTIMATES FROM NUTTLI (1983), USING FOCAL DEPTH OF 10 km

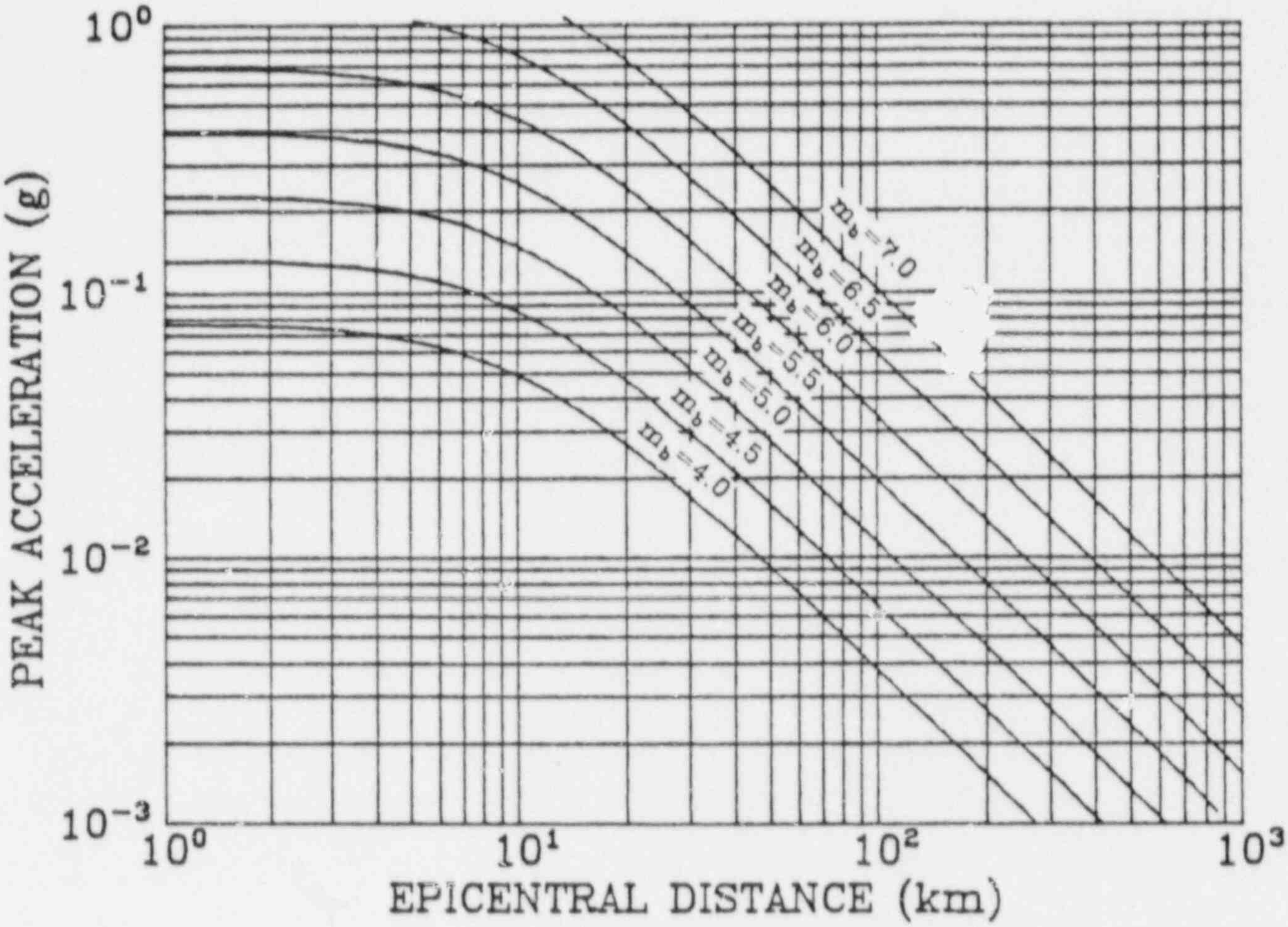

FIGURE 12

PEAK ACCELERATION ESTIMATES FROM
"CORNWALL-MASSENA" ATTENUATION FUNCTION

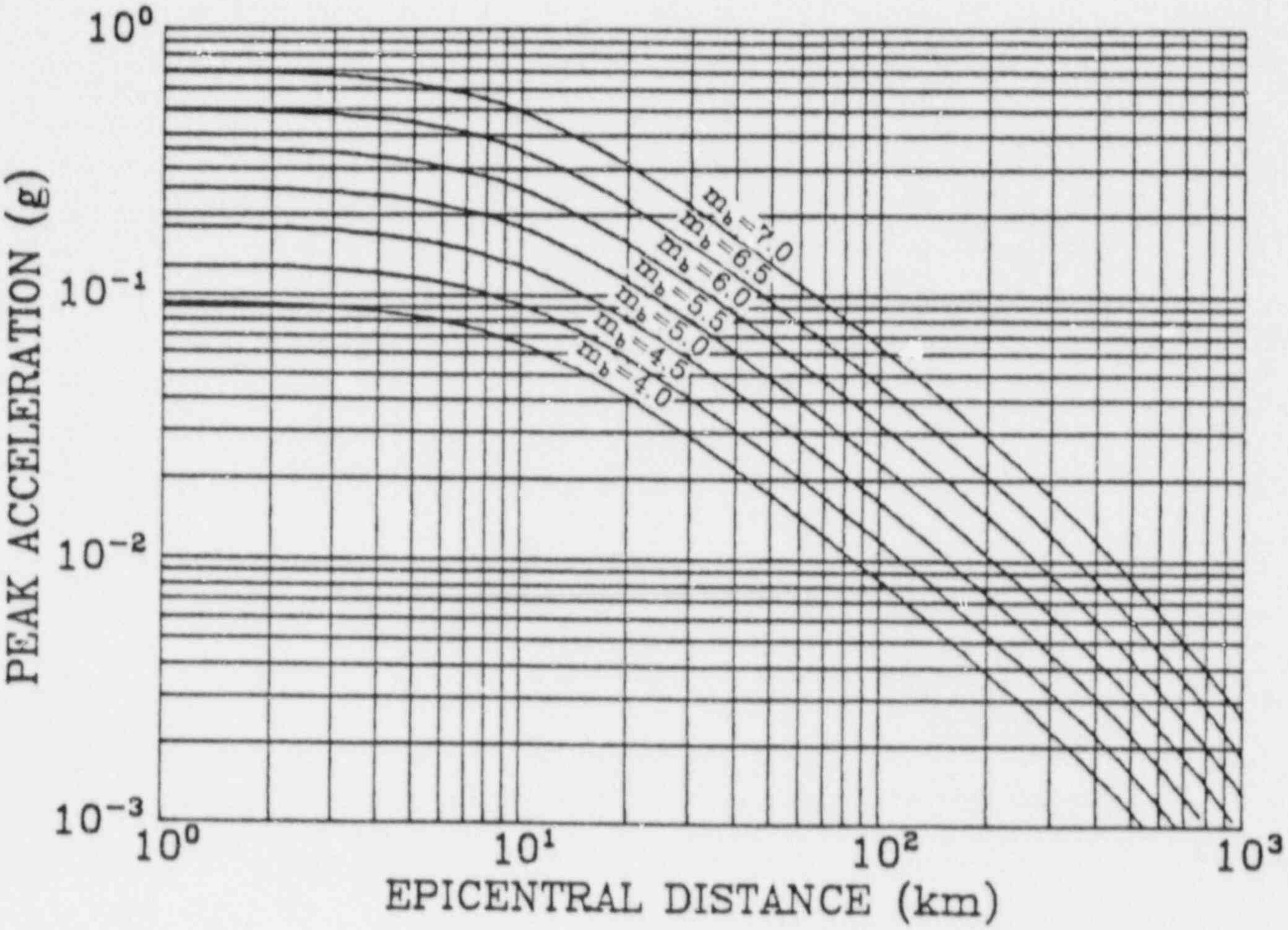

FIGURE 13

PEAK ACCELERATION ESTIMATES FROM ATKINSON (1984)

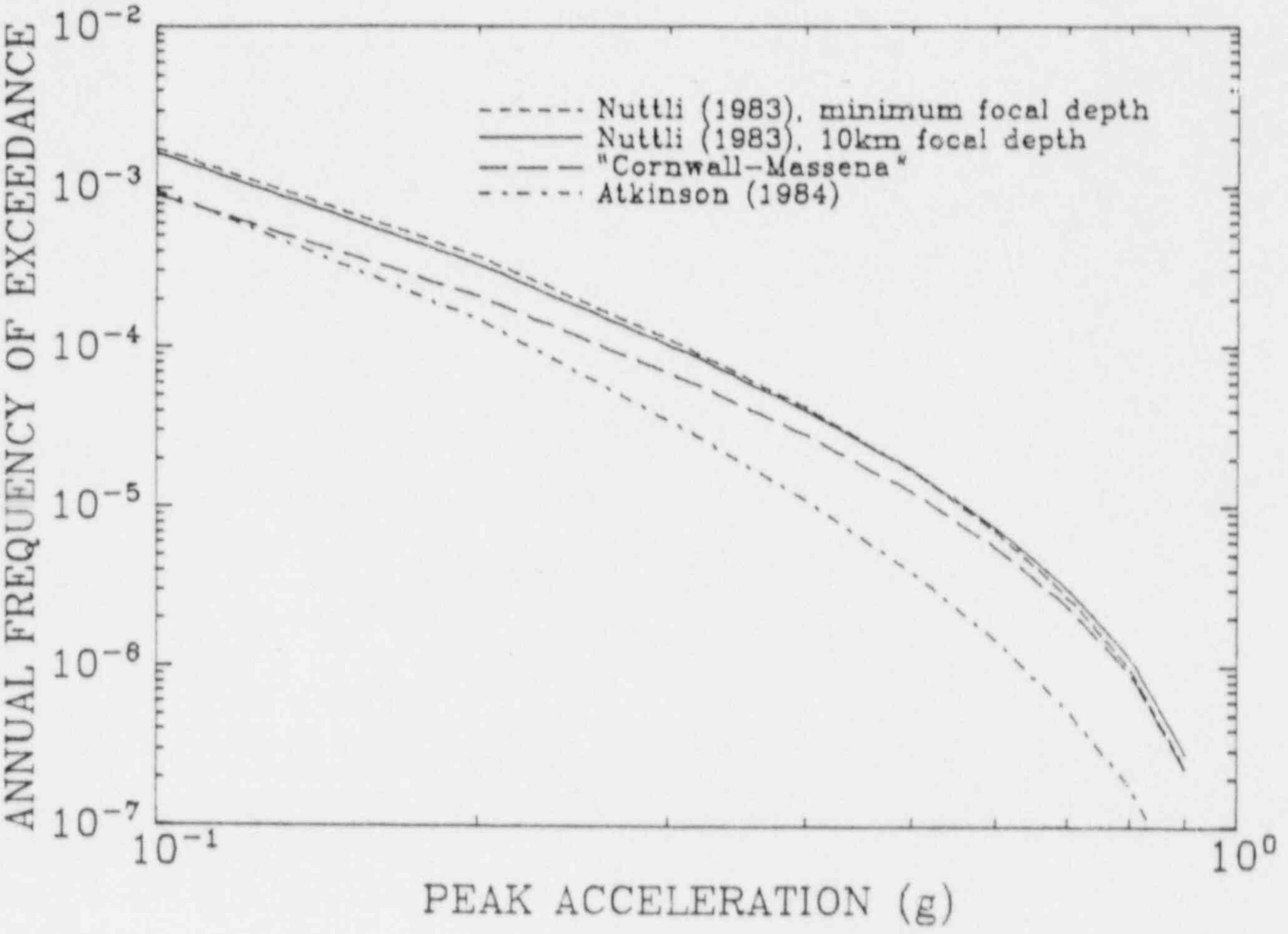

FIGURE 14

SENSITIVITY OF SEISMIC HAZARD RESULTS
TO ATTENUATION FUNCTIONS

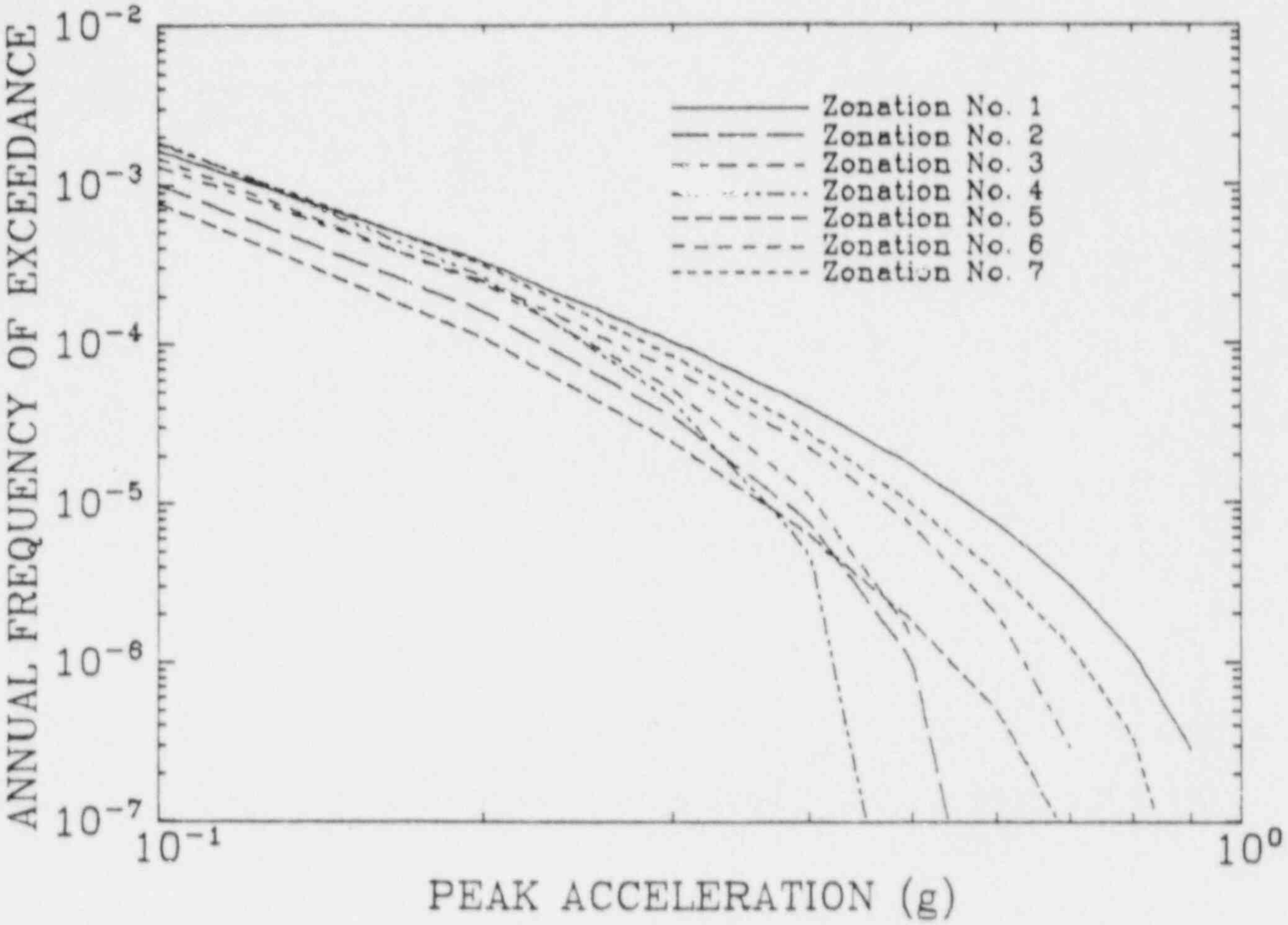

FIGURE 15

SENSITIVITY OF SEISMIC HAZARD RESULTS TO ZONATION

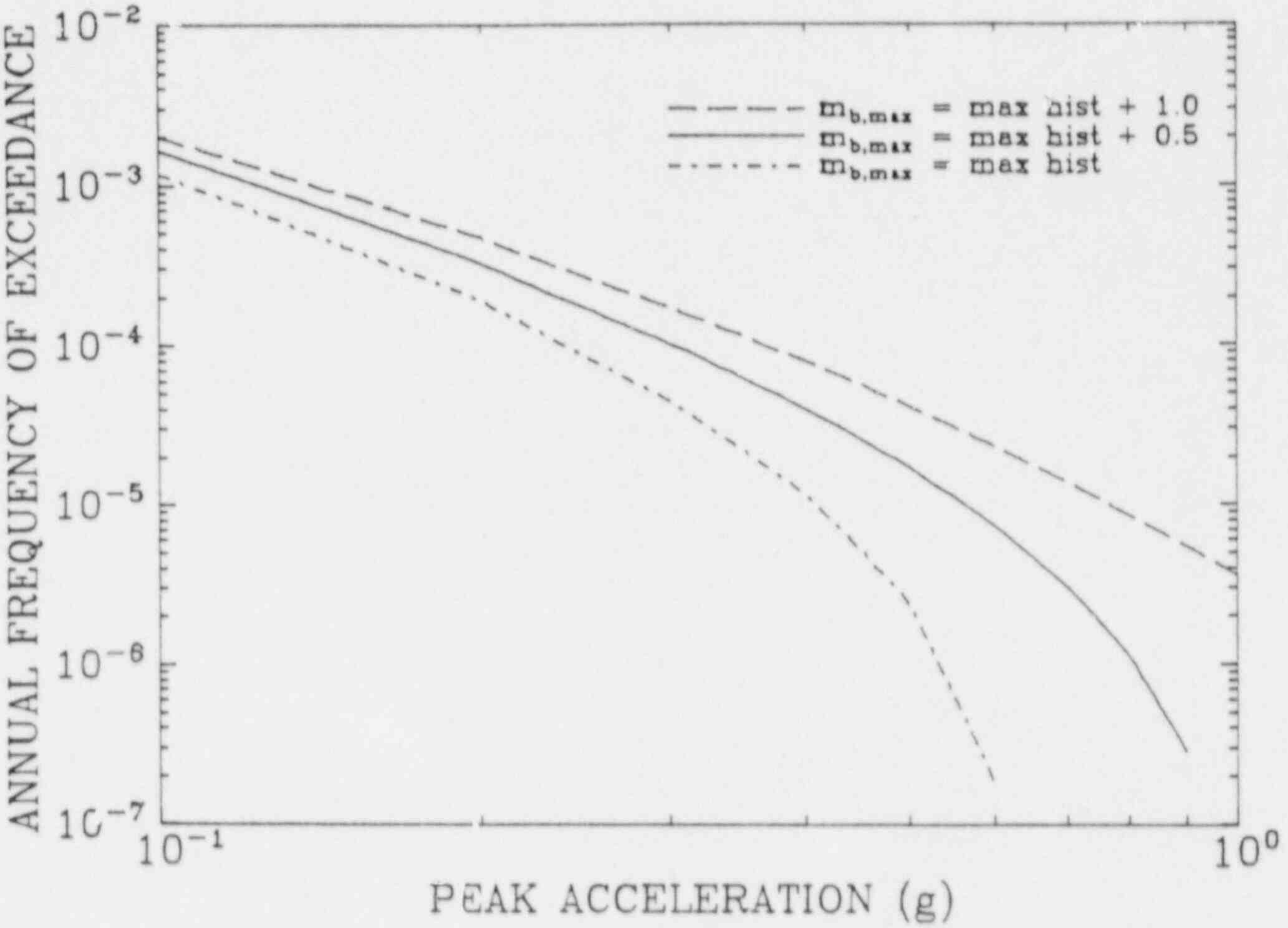

FIGURE 16

SENSITIVITY OF SEISMIC HAZARD RESULTS
TO MAXIMUM MAGNITUDE

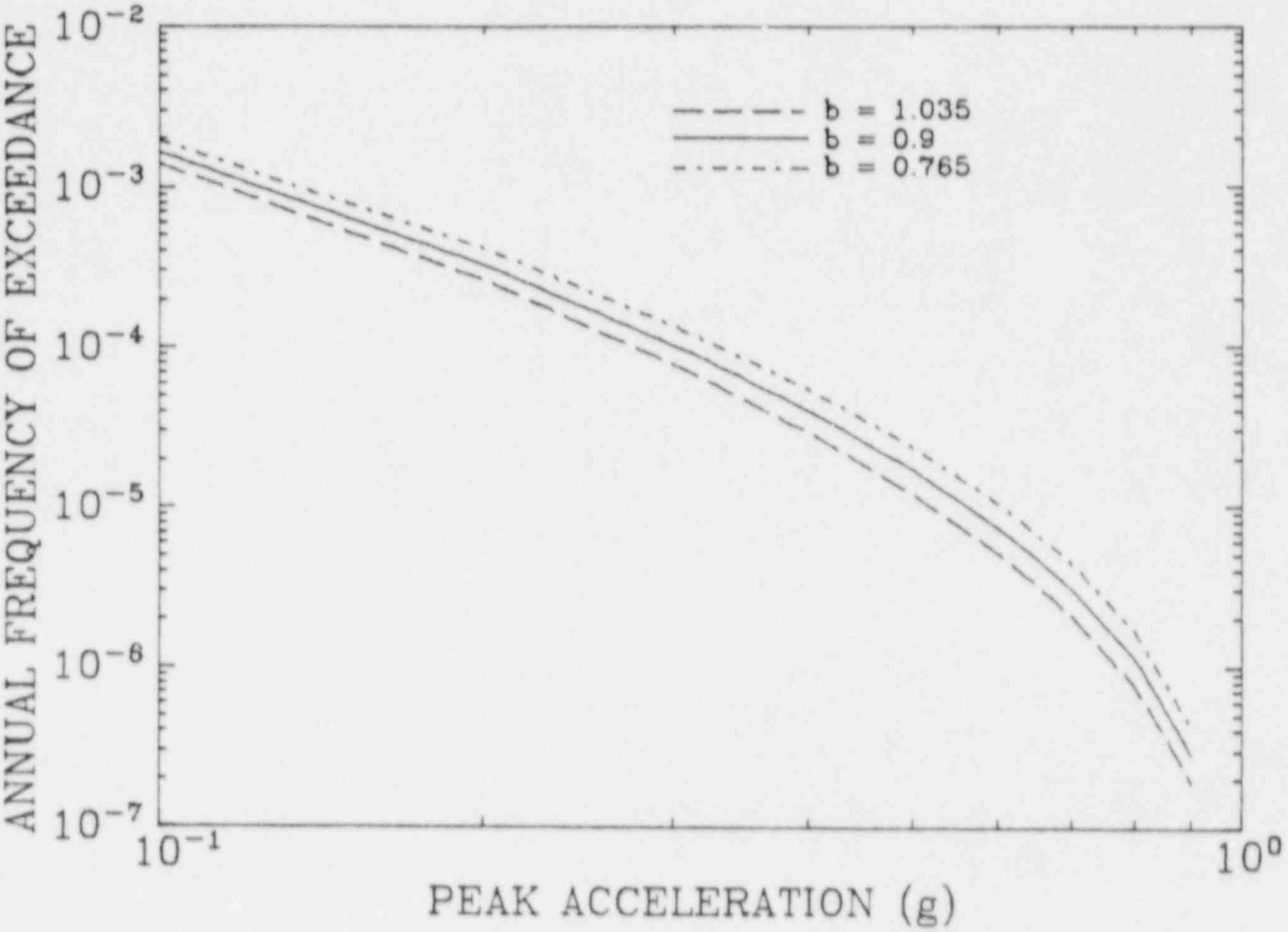

FIGURE 17

SENSITIVITY F SEISMIC HAZARD RESULTS
TO b-VALUE

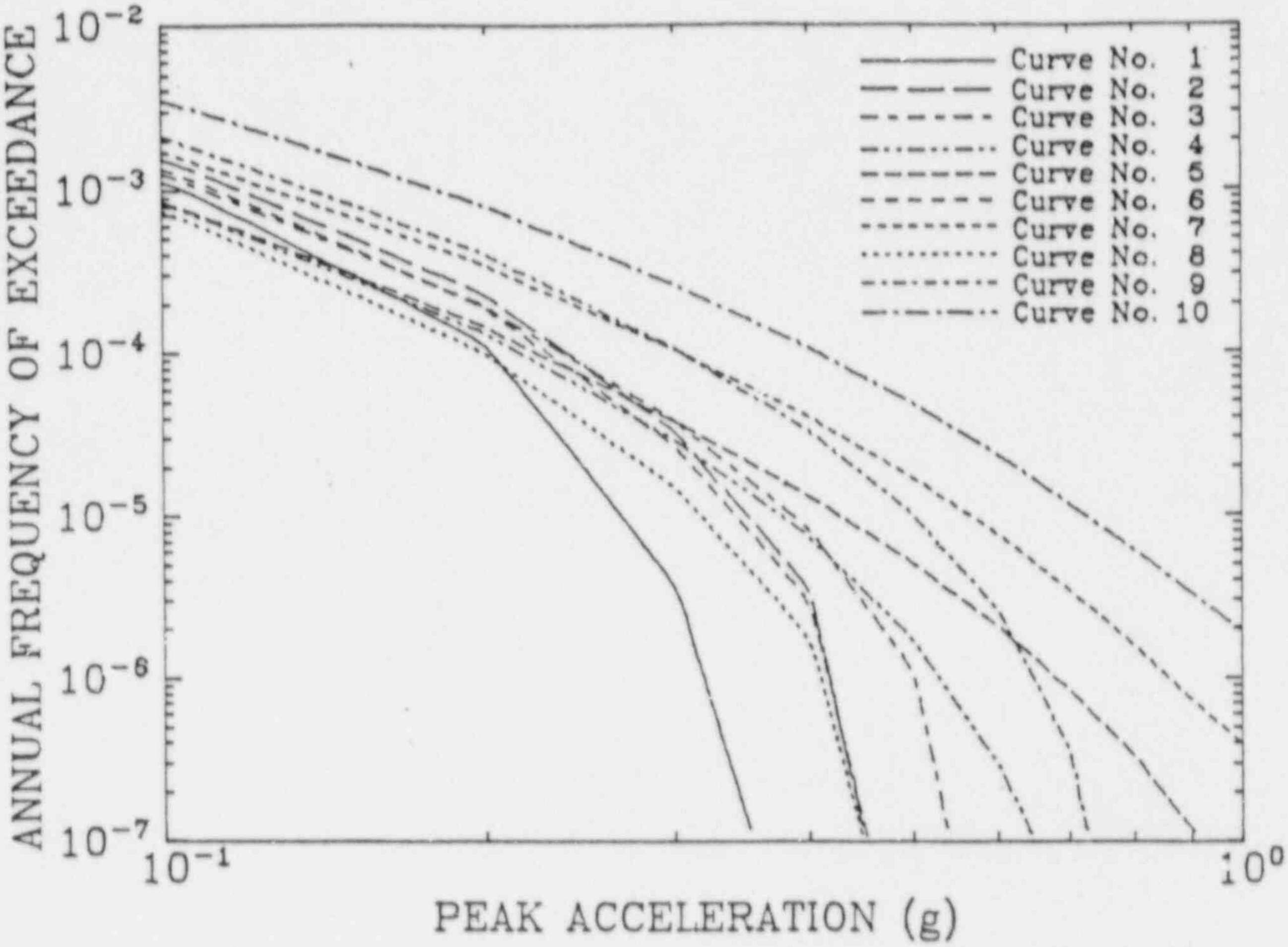

FIGURE 18

AGGREGATE SEISMIC HAZARD CURVES

APPENDIX B

STRUCTURAL MECHANICS ASSOCIATES, INC., REPORT

Seismic Fragilities of Structures and Components at the Three Mile Island, Unit 1, Nuclear Power Plant

SMA 12914.01

SEISMIC FRAGILITIES OF STRUCTURES AND COMPONENTS AT THE THREE MILE ISLAND, UNIT 1, NUCLEAR POWER PLANT

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Rev.	Description	QA	Project Manager
A 2/1985	Draft for Review	Thomas R. Kipp	

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

Section Title Page

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

1. INTRODUCTION

A probabilistic risk assessment (PRA) of the Three Mile Island, Unit 1, (TMI-1) Nuclear Power Station is being conducted by Pickard, Lowe and Garrick, Inc. (PLG). In this evaluation, system models, event trees, and fault trees are utilized to determine the frequency of radioactive release from the site due to random equipment failures and failures initiated by external hazard events. Earthquakes are one of the extreme natural hazards being considered in this PRA. Structural Mechanics Associates, Inc. (SMA) is under subcontract to PLG to provide the required information for earthquake (seismic) capacities of civil structures and mechanical and electrical equipment items that are included in the risk models.

The frequency of seismically-induced failure as a function of effective peak ground acceleration for both safety-related structures and equipment has been developed by SMA for the TMI-1 facility. Also included is the expected variability in the frequency of failure. The determination of the seismic hazard is being conducted by others, and no evaluation of any possible soils-related failures was conducted. The information for both the frequency of occurrence of different levels of effective peak ground acceleration and the frequency of failure of the safety-related systems and components will then be incorporated into the risk models by PLG to determine the frequency of seismic-induced radioactive release from the site.

In order to correctly interpret the fragilities derived in this report, it is necessary to define the effective peak acceleration to which these fragilities are anchored. It is recognized that the damage potential of an earthquake depends on many factors, among which are magnitude, peak acceleration, and duration. For the TMI-1 site, it was initially estimated that the majority of seismic risk results from earthquakes that have magnitudes centered around 6.3 with an approximate range of 5.8 to 6.8. This is the range represented by the ground response spectra used to evaluate the fragilities. The fragilities given in this report are to be anchored to the mean peak acceleration. This acceleration is the average of the peak accelerations from two orthogonal horizontal components.

The Three Mile Island, Unit 1 Nuclear Power Station was designed in the 1960's in accordance with criteria and codes in effect at that time (Reference 1). Table 1-1 lists some of the the more important codes, standards, and specifications used in design of the structures. The TMI-1 systems and components which are essential to the prevention or mitigation of consequences of accidents which could affect the public health and safety were designed to enable the facility to withstand the effects of natural forces including earthquakes. The design criteria included the effects of simultaneous earthquake and loss-of-coolant-accident (LOCA) conditions. The plant was designed to withstand both a design earthquake (equivalent to an Operating Basis Earthquake, OBE) and Maximum Hypothetical Earthquake (equivalent to a Safe Shutdown Earthquake, SSE). The seismic design criteria were based on 0.06g horizontal ground acceleration for the design earthquake and 0.12g for the maximum earthquake for Class I structures and equipment. Vertical accelerations of two-thirds of the corresponding horizontal values were used for both the design and maximum earthquakes.

The plant structures and equipment were originally divided into three classes according to their function and the degree of integrity required to protect the public. Definitions of the three classes used in the seismic design are shown in Table 1-2. The TMI-1 structures, systems and components important to safety, were designed to withstand the effect of design and maximum hypothetical earthquakes and were designated as Class I. Table 1-3 lists the Class I structures and components. Structures, equipment, and components which are important to plant operation, but are not essential for preventing an accident which would result in release of substantial amounts of radioactivity are designated as Class II. All other structures and components are Class III. No Class I equipment is installed in other than Class I buildings.

The site is located on the northern one-third of Three Mile Island in the Susquehanna River approximately 2 1/2 miles south of Middleton, Pennsylvania. The bedrock is a sedimentary sequence of interbedded sandstone, shaley siltstone, and shaley claystone. The bedrock surface is essentially flat and lies at approximately elevation 277 ft. Seismic velocities range from 8500 to 11,500 ft/sec.

Fluvially deposited soil on the island ranges from approximately 6 to 30 foot depths, and consists of stratfied sand and gravel containing various amounts of silt, clay and some lenses of clean sand. Density values range from loose to very dense based on Standard Penetration Tests. The depth of soil is relatively constant at approximately 20 feet in the vicinity of the plant site. All Class I structures are founded on either bedrock or compacted backfill. Table 1-4 shows the type of foundation, base elevation, and foundation medium for the Class I structures. Potential soils-related seismic modes of failure (liquefaction, seismic-induced structure settlement) are not expected to be controlling but were not evaluated as part of the current evaluation.

The ground response spectra used in the design of TMI-1 were developed from smoothed spectra obtained from the 1957 Golden Gate Park, San Francisco earthquake. These spectra were modified to reflect the increased response at the lower frequencies based on the 1940 El Centro spectra (Reference 2). The horizontal ground response spectra are anchored to 0.06g for the design earthquake and 0.12g for the maximum earthquake. The horizontal spectra used for the design earthquake are shown in Figure 1-1.

The modal response spectrum method of analysis was used for design of the TMI Class I structures. A shell of revolution model was developed for the containment building. Two-dimensional lumped mass models were developed for the other civil structures. The seismic response contributions from one horizontal and the vertical component were combined on an absolute sum basis.

For the most part, results of existing analyses and evaluations of structures for the TMI-1 plant were utilized in this study. As part of this evaluation, some limited analysis based on original design analysis loads was conducted to determine the expected seismic capacities of the important structures. The approach adopted in this study was to determine the median factor of safety and its statistical variability which exists for the maximum hypothetical earthquake (SSE) in order to estimate the expected response at failure.

The equipment fragilities for the TMI-1 plant were derived in two phases. The first phase included the development of a set of conservative-generic fragilities for all of the equipment modeled in the seismic event trees and fault trees. The conservative-generic values were based on the results of previous PRA's, a review of the TMI-1 design criteria, results of the TMI-1 site visit and earthquake experience data. These conservative fragility descriptions were run through the plant system models by PLG to determine the governing accident sequences and the components dominating core damage and risk. The second phase of the equipment fragility derivation consisted of the development of plant-specific values for only those components that were identified in Phase 1 to dominate core damage and risk. Chapter 5 contains a more complete description of the equipment fragility methodology along with selected examples.

An evaluation of the individual important structures and of the risk-sensitive equipment was conducted for specific items and failure modes. Although inelastic energy dissipation is included in determining the factors of safety, no nonlinear analyses have been conducted for either the structures or equipment for TMI-1 and all evaluations were based on elastic analysis and load distributions.

These results can be used together with the estimated annual frequency of occurrence of various ground motion levels to determine the frequency of seismic-induced failure for each safety-related structure or component in the plant. In the total study, these conditional component failure frequencies are used with systems models to determine the probability of core melt frequencies and radioactive release frequencies. These results are then combined with the results of the consequence analysis to determine the risk induced by earthquakes.

TABLE 1-1

TMI-1 STRUCTURES

DESIGN CODES AND STANDARDS

Building Code Requirement for Reinforced Concrete, ACI 318-63

Specifications for Structural Concrete for Buildings, ACI 301-66 (except as modified).

AISC Manual of Steel Construction

ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessel; Section VIII, Unfired Pressure Vessels; Section IX, Welding Qualifications (applicable portions).

AEC Publication TID-7024

Regulation for Protection From Fire and Panic - Commonwealth of Pennsylvania

TABLE 1-2

SEISMIC DESIGN CLASS DEFINITIONS

Class I

Those structures, components, and systems, including instruments and controls, whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of radioactivity, and those structures and components which are vital to safe shutdown and isolation of the reactor are designated Class I. When a system as a whole is referred to as Class I, certain portions not associated with loss of function of the systems may have been designated under Class II or III as appropriate. A listing of Class I structures, components, and systems is given in Table 1-3.

Class II

Those structures, components, and systems which are important to reactor operation but not essential to safe shutdown and isolation of the reactor and whose failure could not result in the release of substantial amounts of radioactivity are designated Class II.

Class III

Those structures, components, and systems which are not related to reactor operation or containment are designated Class III.

TABLE 1-3

SEISMIC DESIGN CLASS I STRUCTURES, COMPONENTS AND SYSTEMS

a. Buildings and Structures

Reactor Building including all penetrations, equipment hatch and air locks, concrete shell, liner, and interior structures.

Auxiliary Building

Fuel Handling Building and fuel storage pools

Control Building

Diesel Generator Building

Intermediate Building (portions)

Intake screen and pump house

Heat exchanger vault and access tunnel-vault to Auxiliary Building

Air intake structure (portion belowground)

b. NSSS Components

Reactor vessel

Reactor internals (including fuel elements and control rods)

Control rod drive mechanisms (and support)

Pressure parts of In-Core Monitoring System

Steam generators (and supports)

Pressurizer (and supports)

Reactor coolant Pumps and motors and supports

TABLE 1-3 (Continued)

SEISMIC DESIGN CLASS I STRUCTURES, COMPONENTS AND SYSTEMS

c. Engineered Safeguards Systems

Makeup and Purification System (high-pressure injection system) including: makeup pumps, makeup tank, letdown coolers, letdown filters, seal return cooler, process and instrument piping and valves.

Core flooding tanks including process and instrument piping and valves.

Decay Heat System (low-pressure injection system) Including: decay heat pumps, decay heat coolers, process and instrument piping and valves.

Borated water storage tank.

Sodium hydroxide storage tank.

Reactor Building Spray System including: spray pumps, spray headers and nozzles, process and instrument piping and valves.

Reactor Building emergency air cooling units including: fans and motors, demisters, cooling coils and connecting air handling duct.

Combustible Gas Control System Including Hydrogen Recombiner System, and Hydrogen Monitoring System.

Reactor Protection Systems.

Engineered Safeguards Actuation System.

All piping penetrations and associated isolation valves.

d. Vital Cooling Water Systems.

Decay Heat Services Cooling Water Systems A and B including: surge tank, pumps, heat exchangers, process and instrument piping and valves.

Decay Heat River Cooling Water Systems including: pumps, heat exchangers, process and instrument piping and valves.

Nuclear Services Closed Cooling Water System including: pumps, heat exchangers, process and instrument piping and valves.

Nuclear Services River Cooling Water System including: pumps, heat exchangers, process and instrument piping and valves.

Reactor Building Emergency Cooling Water System including: pumps process and instrument piping and valves and cooling coils.

TABLE 1-3 (Continued)

SEISMIC DESIGN CLASS I STRUCTURES, COMPONENTS AND SYSTEMS

e. Emergency Power Supply System

Diesel generators and fuel oil storage tanks

DC power supply system and inverters

Power distribution lines to equipment required for emergency

Switchgear and power centers supplying the engineered safety features

Control console

Motor control centers

f. Spent Fuel Cooling System including: spent fuel pumps, heat exchangers, all process and instrument piping and valves, etc.

g. Vital Ventilation Systems, Ventilation system for pump of Spent Fuel Cooling System

Ventilation system for intake screen and pump house

Ventilation system for diesel generators

Ventilation system for nuclear service cooling system pumps

Ventilation system for Emergency Feedwater System

Ventilation system for Control Building

h. Miscellaneous Vital Systems and Components

Those portions of the Emergency Feedwater System required for decay heat removal including: pumps, condensate storage tanks (excluding hotwell), steam generator pressure and level indications, auxiliary (emergency) feedwater control valves, and atmospheric relief valves.

Underground diesel fuel tank

Instrument and Control Air System

TABLE 1-3 (Continued)

SEISMIC DESIGN CLASS I STRUCTURES, COMPONENTS AND SYSTEMS

Hydrogen and Nitrogen Supply System Including: nitrogen manifold (portions supplying core flood tanks, vital to penetration pressurization system and fluid block system).

New and spent fuel storage racks

Reactor building polar crane

Fuel Handling crane

River pump service crane

Water gates in fuel storage pools

i. Waste Disposal System

Reactor coolant bleed tanks

Miscellaneous waste storage tank

Reactor coolant drain tanks

Spent resin storage tank

Used filter precoat tank

Concentrated radioactive waste storage tank

Reclaimed boric acid tanks

Neutralizer tank

Neutralized waste storage tank

Laundry waste tank

Reactor coolant drain tank cooler

Reactor coolant drain tank pump

Liquid outlet piping to second isolation valve downstream from each of the above tanks and the process piping associated with the reactor coolant drain tank.

TABLE 1-4

FOUNDATIONS FOR CLASS I STRUCTURES

Structure	Type of Foundation	Base Elevation	Foundation Medium
Reactor Building	Mat	270 ft	Bedrock
Auxiliary Building	Mat	278 ft	Bedrock
Fuel Handling Building	Mat	276 ft-6 in	Bedrock
Control Building	Continuous footings under walls. Square footings under columns	278 ft	Bedrock
Diesel Generator Building	Mat	303 ft	Compacted backfill
Intermediate Building	Continuous footings under walls	277 ft	Bedrock
Intake Building	Mat	259 ft 6 in 262 ft 6 in	Bedrock
Heat exchanger vault	Mat	267 ft 6 in	Bedrock
Access tunnel vault to Auxiliary Building	Mat	279 ft	Bedrock
Air intake structure	Mat	279 ft 278 ft	Bedrock
Borated water tank Sodium thiosulfate Sodium hydroxide	Mat	300 ft	Compacted backfill
Emergency Feedwater tank System condensate	Mat	300 ft 11 in	Compacted backfill
Underground diesel fuel tank	Mat	283 ft 6 in	Compacted backfill

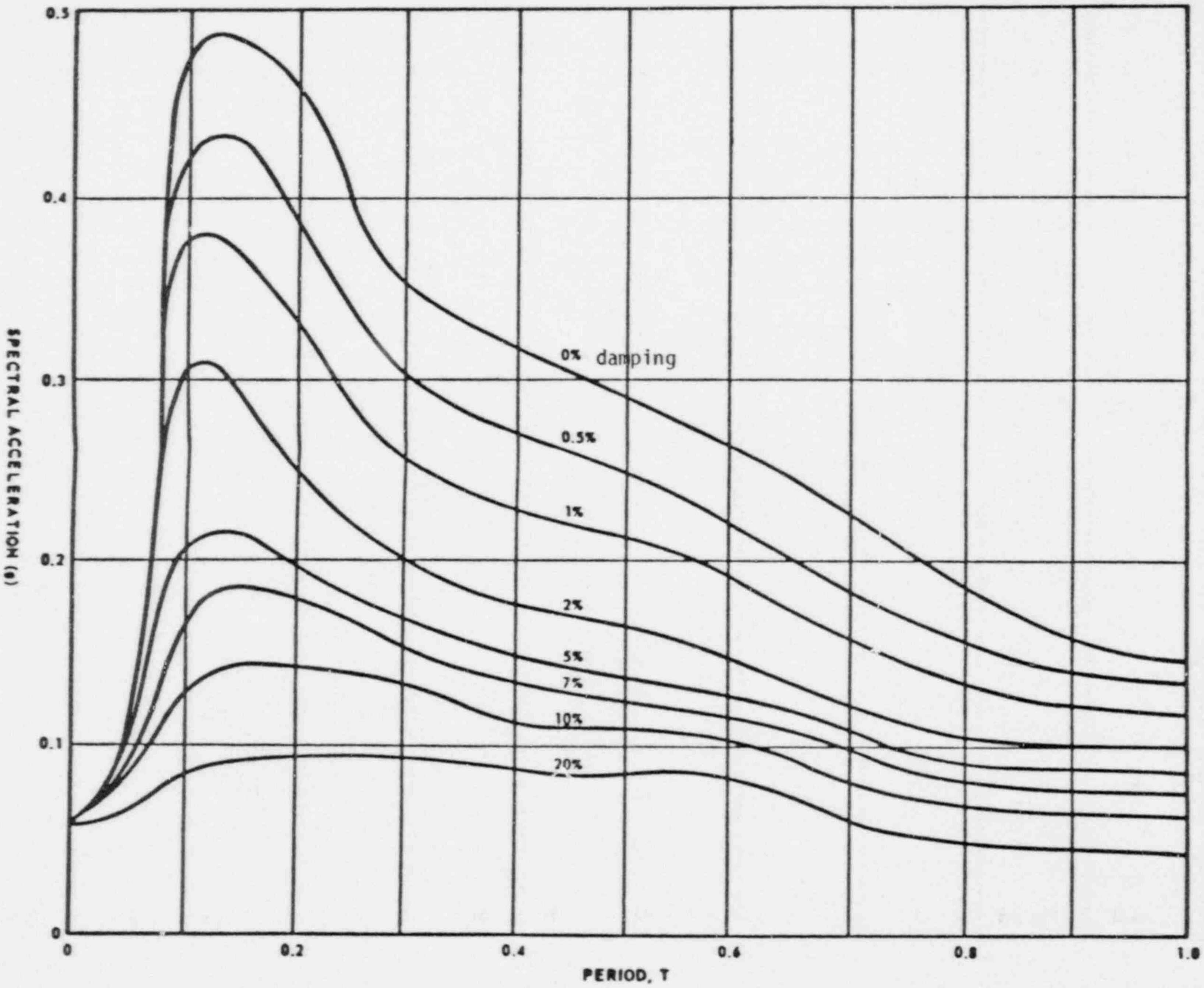


FIGURE 1-1. DESIGN EARTHQUAKE GROUND RESPONSE SPECTRA

2. GENERAL CRITERIA FOR DEVELOPMENT OF MEDIAN SEISMIC SAFETY FACTORS

The factor of safety of a structure or component is defined herein as the resistance capacity divided by the response associated with the maximum hypothetical earthquake (SSE) of 0.12g effective peak acceleration used in the design of the structures. The development of seismic safety factors associated with the maximum earthquake is based on consideration of several variables. The variability of dynamic response to the specified acceleration and the strength capacity of the structure or equipment component are the two basic considerations in determining the variability in the factor of safety. Several variables are involved in determining both the structural response and the structural capacity, and each such variable, in turn, has a median factor of safety and variability associated with it. The overall factor of safety is the product of the factors of safety for each variable. The median of the overall factor of safety is the product of the median safety factors of all the variables. The variabilities of the individual variables also combine to determine that of the overall safety factor.

Variables influencing the factor of safety on structural capacity to withstand seismic-induced vibration include the strength of the equipment or structure compared to the design stress level and the inelastic energy absorption capacity (ductility) of a structure or its ability to carry load beyond yield. The variability in computed structural response for a given effective peak free-field ground acceleration is made up of many factors. The more significant factors include variability in (1) ground motion and the associated ground response spectra for a given peak free-field ground acceleration, (2) energy dissipation (damping), (3) structural modeling, (4) method of analysis, (5) combination of modes, (6) combination of earthquake components, and (7) soil-structure interaction for structures founded on soil. For structures which may be susceptible to sliding, the variability in the amount of sliding is also significant. Equipment located inside a building acts as a secondary system and requires the previously mentioned structural response factors together with a similar set of equipment response factors which are specific to the equipment itself (see Chapter 5). The ratio between the median value of each of these factors and the value used in design of the TMI-1 plant and the variability of each factor are quantitatively estimated in Chapters 4 and 5 for various structures and components. These estimates are based on available test data for TMI-1 structures and equipment, limited analysis, and engineering judgment and experience in the analysis of nuclear power plants and components.

2.1 DEFINITION OF FAILURE

In order to estimate the median factor of safety against the structure or component failure for the maximum hypothetical earthquake (SSE) effective peak acceleration (0.12g), it is necessary to define what constitutes failure.

2.1.1 Structures

For purposes of this study, structures are considered to fail functionally when inelastic deformations of the structure under seismic load are estimated to be sufficient to potentially interfere with the operability of safety-related equipment attached to the structure. These limits on inelastic energy absorption capability (ductility limits) chosen for structures are estimated to correspond to the onset of significant structural damage. For many potential modes of failure, this is believed to represent a conservative bound on the level of inelastic structural deformation which might interfere with the operability of components housed within the structure. It is important to note that considerably greater margins of safety against structural collapse are believed to exist for these structures than many cases reported within this study. Thus, the conditional probabilities of failure for a given free-field ground acceleration reported herein for structures are considered appropriate for equipment operability limits and should not necessarily be inferred as corresponding to structure collapse. Structures which are susceptible to sliding are considered to have failed when sufficient sliding deformation is incurred to fail piping or electrical duct banks or to cause sufficient damage resulting from structure-to-structure impact to interfere with equipment.

2.1.2 Equipment and Piping

Piping, electrical, mechanical and electro-mechanical equipment vital to safe shutdown of the plant or mitigation of an accident are considered to fail when they will no longer perform their designated functions. Rupture of the pressure boundary on mechanical equipment is also considered a failure. Therefore, for mechanical equipment, a dual failure definition exists: failure to function and pressure boundary rupture. Depending upon the equipment type, one or the other definition will govern. For active equipment, the functional failure definition usually governs as equipment pressure boundaries are generally very conservatively designed for equipment such as pumps and valves. For piping, failure of the support system or plastic collapse of the pressure boundary are considered to represent failure. The inelastic energy absorption limits (ductility limits) associated with these failure modes have been conservatively estimated in order to define the margins of safety.

2.2 BASIS FOR SAFETY FACTORS DERIVED IN STUDY

There was a general lack of detailed information available for this study on seismic fragility of specific TMI-1 structures and equipment. This condition exists for all plants and occurs because existing codes and standards do not require determination of ultimate seismic capacities, either for structures or equipment qualified by analysis, or for equipment or components qualified by testing. Therefore, most median safety factors, estimates of variability, and conditional frequencies of failure estimated in this study are based on existing analyses and tests together with qualified engineering judgment and assumptions. Limited additional analyses were conducted to evaluate the expected failure capacities of the important structures and of selected equipment. The additional analyses on the structures were conducted to develop structural loads and load distributions and were derived from the available design information.

2.2.1 Structural Response and Capacity

Information available from the seismic design analyses of the important structures was extensively used in this study. This was supplemented as required to provide estimates of load distributions through the structures, etc. Levels of conservatism associated with the method of analysis used in the design were estimated such that safety factors reflecting this analysis could be estimated for the building structures and for the seismic excitation of equipment mounted within the building. Some ultimate load capacity analyses were conducted which served as a basis for estimating the median factor of safety on structural resistance to the maximum earthquake used for design.

2.2.2 Equipment Response and Capacity

As described in Chapter 1 of this report, a combination of both generic and plant specific information was utilized in developing equipment fragilities. Conservative generic fragilities (Phase 1) were determined primarily from values for similar equipment from previous PRA's. The specific fragilities (Phase 2) for equipment critical to the plant risk were based primarily on available vendor seismic qualification reports or design calculations for specific components. Safety factors for response and structural or functional capacity were estimated from existing information. No new analyses were conducted.

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

Capacity factors were derived from several sources of information; plant-specific design reports, test reports, generic fragility test data from military test programs and generic analytical derivations of capacity based on governing codes and standards. Two failure modes were considered in developing capacity factors for piping and equipment: structural and functional. Equipment design reports delineate stress levels for the specified seismic loading plus normal operating conditions. Where the equipment fails in a structural mode (i.e., pressure boundary rupture or loss of support), the median capacity factor and its variability were derived in the same manner as for structures considering strength and energy absorption (ductility). In cases where equipment must function, the capacity factor was derived by comparing the equipment functional failure (or fragility) level to the design level of seismic loading. Some fragility test data are available on generic classes of equipment that have been utilized in hardened military installations. Such equipment was off-the-shelf without special shock-resistant design but is similar to nuclear power plant equipment. These data provide estimates of the fragility levels, and thus, safety factors could be developed for the specified design earthquake. Fragility levels were not normally determinable from equipment qualification reports, but the achieved test levels could be utilized to update generic fragilities derived from the military data.

2.3 FORMULATION USED FOR FRAGILITY CURVES

Seismic-induced fragility data are generally unavailable for specific plant components and are certainly unavailable for the specific TMI-1 structures. Thus, fragility curves must be developed primarily from analysis combined heavily with engineering judgment supported by very limited test data. Such fragility curves will contain a great deal of uncertainty, and it is imperative that this uncertainty be recognized in all subsequent analyses. Because of this uncertainty, great precision in attempting to define the shape of these curves is unwarranted. Thus, a procedure which requires a minimum amount of information, incorporates uncertainty into the fragility curves, and easily enables the use of engineering judgment was used in this study.

The entire fragility curve for any mode of failure and its uncertainty can be expressed in terms of the best estimate of the median ground acceleration capacity, $\check{A}$, times the product of random variables. Thus, the ground acceleration, A, corresponding to failure is given by:

$$A = \check{A}\, \varepsilon_R\, \varepsilon_U \qquad (2\text{-}1)$$

in which ε_R and ε_U are random variables with unit median representing the inherent randomness (failure fraction) about the median and the uncertainty (probability) in the median value, respectively. Equation 2-1 enables the fragility curve and its uncertainty to be represented as shown in Figure 2-1; i.e., as a set of shifted curves with attached uncertainty levels. Thus, it is assumed that all uncertainty in the fragility curves can be expressed through uncertainty in the median alone.

Next, it is assumed that both ε_R and ε_U are lognormally distributed with logarithmic standard deviations of β_R and β_U, respectively. The advantages of this formulation are:

1. The entire fragility curve and its uncertainty can be expressed by three parameters - $\check{A}$, β_R, and β_U. With the very limited available data on fragility, it is much easier to only estimate three parameters rather than the entire shape of the fragility curve and its uncertainty.
2. The formulation in Equation 2-1 and the lognormal distribution are very tractable mathematically.

Another advantage of the lognormal distribution is that it is easy to convert Equation 2-1 to a deterministic composite fragility curve (i.e., one which does not separate out uncertainty from underlying randomness) defined by:

$$A = \check{A}\, \varepsilon_C \qquad (2\text{-}2)$$

where ε_C is a lognormal random variable with unity median and logarithmic standard deviation β_C given by:

$$\beta_C = \sqrt{\beta_R^2 + \beta_U^2} \qquad (2\text{-}3)$$

This composite fragility curve (shown in Figure 2-1) can be used in preliminary deterministic safety analyses if one only needs a preliminary estimate on failure fraction and does not desire an estimate of uncertainty.

In this study, the guidelines used to estimate the values of β_R and β_U for each variable affecting A were based on considering the inherent randomness, β_R, to be associated with the earthquake characteristics themselves, and β_U to be associated with other lack of knowledge. Thus, such variability as resulting from earthquake response spectra shapes and amplification, earthquake duration, numbers and phasing of peak excitation cycles, etc., together with their contributions to structure ductility and response characteristics is attributed to randomness. In general, it is not considered possible to significantly reduce randomness by additional analysis or test based on current state-of-the-art techniques. Uncertainty, on the other hand, is considered to result primarily from analytical modeling assumptions and other lack of knowledge concerning variables such as material strength, damping, etc., which could in many cases be reduced by additional study or test.

The lognormal distribution can be justified as a reasonable distribution since the statistical variation of many material properties (References 3 and 4) and seismic response variables may reasonably be represented by this distribution (Reference 5). In addition, the central limit theorem states that a distribution consisting of products and quotients of distributions of several variables tends to be lognormal even if the individual distributions are not lognormal. Use of this distribution for estimating failure fractions on the order of one percent or greater is considered to be quite reasonable. Lower fraction estimates which are associated with the extreme tails of the distributions must be considered less accurate.

Use of the lognormal distribution for estimating very low failure fractions of components or structures associated with the tails of the distribution is considered to be conservative because the low-frequency tails of the lognormal distribution generally extend farther from the median than actual structural resistance or response data might indicate. Such data generally show cut-off limits beyond which there is essentially zero failure fraction. The degree of conservatism introduced into the probability of release is dependent not only on the conservatism in the fragility description, but also on the seismic hazard description at low seismic levels. If the seismic hazard for low seismic input levels is large enough, it is apparent that very low level earthquakes can govern the seismic-induced release. This is considered unrealistic for engineered structures and equipment found in nuclear power plants. Structures and equipment are subjected to low level dynamic loads from a number of sources including wind on a repetitive basis which have never been known to produce nuclear power plant structural failures. Similarly, for low level earthquakes, it is expected that below some threshold, there is virtually no chance of failure due to seismic excitation. Material strength data, for instance, normally does not fall to very low values compared to the median value but instead normally exhibits some lower bound (References 3 and 4). Other variables, such as damping, also indicate both lower and upper bounds which are not zero or infinite. Extensive studies have been conducted to develop response spectra from available earthquake records and while dispersion exists about the median values, spectra with essentially zero or infinite response do not occur (Reference 5). For these as well as other variables contributing to the seismic fragility of a given structure or component, it is apparent that some lower and upper bound cutoffs on the tails of the dispersion exist. Since the overall fragility curves are based on a combination of these variables, it is expected that a threshold exists below which no failures will occur. This is supported by experience. Although quantitative data are lacking, this threshold value is expected to be at approximately minus two lognormal standard deviations for the median curves using the composite fragility variability. The composite lognormal standard deviation, β_C, is used for the basis of the cut-off rather than randomness or uncertainty since the composite value combines the effects of both dispersions.

However, it is also apparent that some variability should be associated with the cut-off. Essentially no data are available to establish the distribution of this variability or its range. A lognormal distribution is, therefore, assumed consistent with the majority of the other variables encountered in the PRA. The following approximation is recommended for establishing the cut-offs for the various fragility curves:

The cut-off on the lower tails of the median (50 percentile) fragility curve should be:

$$\check{A}_{co} = \check{A}\ [\exp\ (-2\beta_C)]$$

where $\check{A}_{co}$ is the cut-off on the median curve, $\check{A}$ is the median effective peak ground acceleration for failure, and β_C is the composite lognormal standard deviation.

The cut-off for the lower tails of the other fragility curves should be:

$$A_{co} = \check{A}_{co}\ [\exp\ (\ x\beta_C/1.65)\ ]$$

where x is the ratio of the deviation divided by the standard deviation. For instance, for the median curve, $x = 0$; for the 25 percentile curve, $x = -0.67$; for the 5 percentile curve and below, $x = -1.65$; and for the 95 percentile curve and above, $x = 1.65$.

It is recommended that the cut-off on the upper tails be established as $+3\beta_C$ for all fragility curves. Similarly, for fragility curves involving only uncertainty, it is recommended that the cut-offs be set at $-3\beta_U$ for the lower bound and $+3\beta_U$ for the upper bound, respectively.

Some characteristics of the lognormal distribution as applied to seismic capacities are discussed in Appendix A of this report.

2.4 DESIGN AND CONSTRUCTION ERRORS

An inadequate data base exists upon which to determine explicitly the contributions of design and construction errors for the TMI-1 structures and equipment seismic capacities. Although some discrepencies may have been identified and others may be in the future, these items have been or will be modified as necessary or shown to have no safety implications. Thus, these items are not expected to significantly affect the seismic capacity of the equipment or structures after they have been identified. However, there is a possibility that unidentified design and construction errors may exist which can affect the seismic capacity.

It should be recognized that design and construction errors do not necessarily always result in a decrease in capacity. It is possible to install higher strength bolts than specified, larger reinforcing bars or more closely-spaced bars than required, or slip a decimal point in the conservative as well as in the unconservative direction in the analysis. Some additional confidence exists in that structures and equipment are subjected to normal operating loads continually. In many cases, these loads may be large; as for instance, in the case of pressure, water hammer, and thermal loads in fluid systems when compared to seismic loads. In other cases, as for instance the wind forces on structures, the loads may be less than seismic loads but occur on a much more frequent basis. Pressure tests of containment vessels, while producing different types of response than seismic, would likely provide an indication if significant construction errors exist in these structures. Thus, although data on which to quantify accurate estimates of the effects of major design and construction errors are not available, these are expected to affect a minimal number of components.

2.5 CORRELATION BETWEEN FAILURE MODES

Many of the potential failure modes discussed in the following sections are not considered to be completely independent. The most obvious examples involve failure of one item caused by failure of a separate component. For instance, if a potential mode of failure is the collapse of a structure, failure of the equipment and piping located in that structure is also expected. Similarly, failure of relatively heavy equipment may often be expected to fail lighter equipment in the immediate vicinity. Some degree of correlation exists for all items and for all modes of failure since they are all excited by the same earthquake. An example of very high dependency of failure modes of components and structures include two identical items located very close to each other in the same structure. For two components which are identical but located in different structures or different locations in the same structure, some degree of correlation is expected but less than 100%.

For different modes of failure in a given structure, or in similar structures, some degree of correlation between modes is also expected. For instance, if the capacity of the lateral force resisting system (i.e., the shear walls) is actually higher or lower than the value used in the analysis, the acceleration capacities of all failure modes (including different structures) governed by the shear walls would be expected to be proportionately higher or lower. The actual capacity of the force resisting system may be different from that used in the evaluation due to differences in strength or modeling assumptions. These effects are of course included in the variabilities associated with each mode of failure for a given structure or component. However, different degrees of correlation may exist from mode-to-mode. For instance, for a given structure with given concrete and reinforcing steel strengths, the variability on strength from mode-to-mode may be strongly correlated, while different modeling assumptions may result in little correlation for different failure modes.

There is also a certain degree of interdependency between structural and sliding modes of failure that could be considered. In Section 4, fragilities are presented for failure modes associated with structure sliding or failure of the structure itself (i.e., shear wall failure). These fragilities were developed assuming that the sliding and structural failure modes were completely independent. That is, structural failure acceleration capacities were based upon seismic loads with the structure bonded to the supporting rock or soil, even if sliding was expected at lesser acceleration levels. In reality, the occurrence of sliding will limit the structure inertial loads since accelerations in excess of those corresponding to sliding cannot be transmitted through the structure/rock or soil interface. Treatment of the structure fragilities which incorporate the probability that sliding does occur would likely result in higher structure capacities.

For failure modes with little contribution to risk, consideration of correlation between modes is probably unimportant. However, consideration should be given to possible correlation between controlling seismically-induced failure modes.

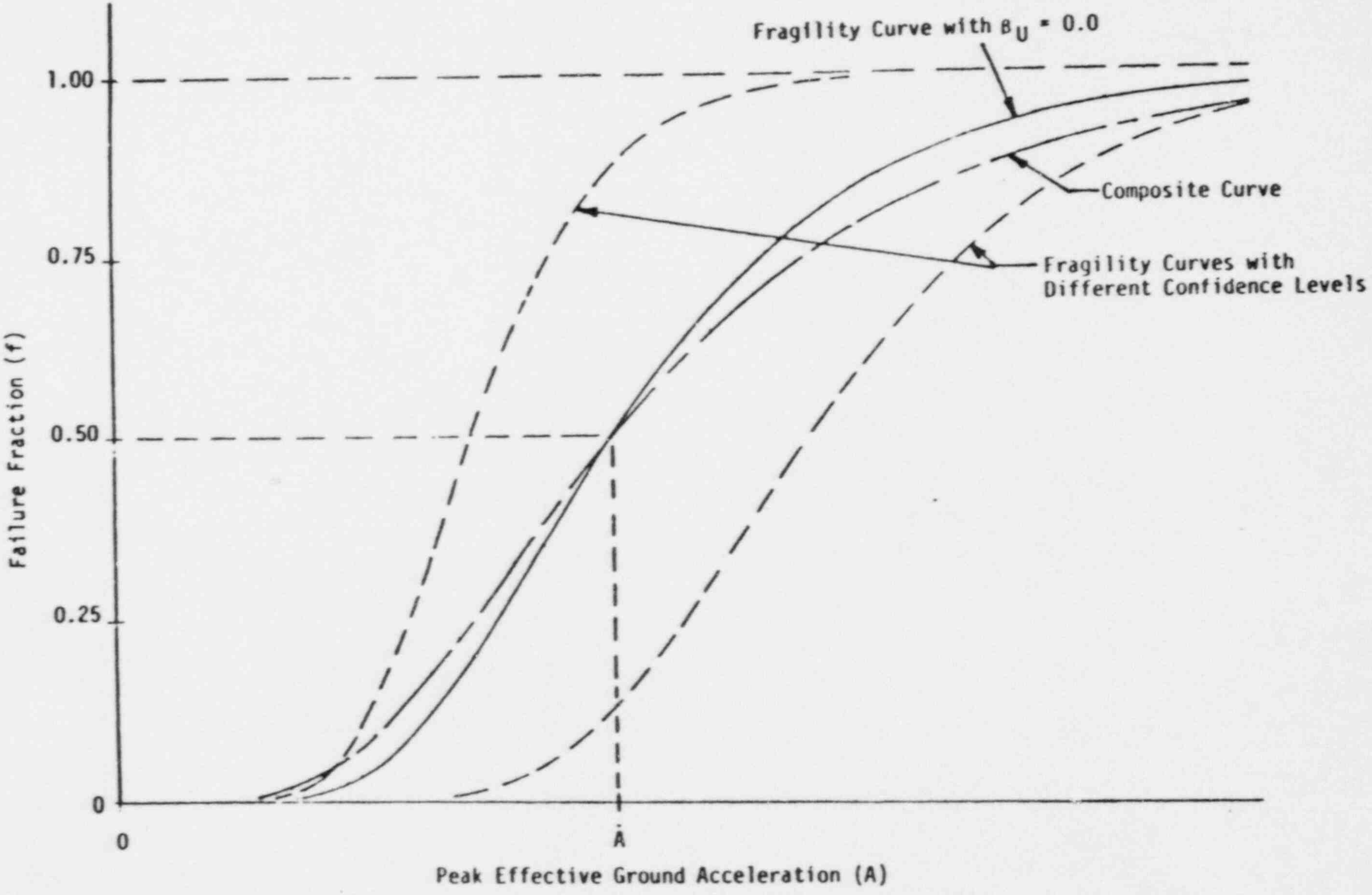


FIGURE 2-1. FRAGILITY CURVE REPRESENTATIONS

3. DIFFERENCES BETWEEN CRITERIA USED FOR ORIGINAL DESIGN OF TMI-1 AND PARAMETERS USED IN THE EVALUATION OF THE SEISMIC CAPACITY

The original seismic design of the TMI-1 structures and equipment was based upon criteria accepted at the time the design work was conducted. The evolution of the nuclear power industry has brought about changes in seismic design and qualification criteria for structures and equipment since that time. These changes do not necessarily imply that the old seismic qualifications were unsafe; merely that the criteria are now much better defined with less interpretation by the designer.

The original design criteria and methods together with the design codes form conservative design bases and ensure that substantial factors of safety were introduced at various stages in the design procedure. The exact magnitude of many of these safety factors is still a matter of considerable discussion. Nevertheless, in order to establish a realistic value of the actual seismic capacity of a structure or equipment component, the amount of conservatism along with its variability must be established as accurately as possible. In this chapter, the original design bases of the most important parameters affecting seismic capacity are identified, and the general methods used in obtaining more realistic values associated with very high seismic response levels are discussed. The detailed determination of these parameters is described in Chapters 4 and 5 for structures and equipment, respectively. The estimated seismic capacities of the most probable failure modes are also developed in Chapters 4 and 5.

The general approach used in the evaluation of the TMI-1 seismic capacities is to develop the overall factor of safety associated with each important potential failure mode. Based on the governing design parameters, a median seismic capacity is then obtained in terms of some representative seismic input such as free-field acceleration. The overall factor of safety is typically composed of several important contributions such as strength, allowance for inelastic energy dissipation (ductility), and differences in median structure response compared to design values resulting from such parameters as earthquake characteristics, damping, and directional load components.

3.1 STRENGTH

The design strength of a structure or an equipment component is typically determined from applicable codes and standards such as the ACI building code for concrete or the ASME boiler and pressure vessel code for mechanical equipment. Inherent in these design codes is a factor of safety on material strength. Sometimes this factor is known reasonably accurately, such as the design allowable being one-half the minimum yield strength or some similar relationship. At other times, it is less well defined or may be a function of the geometry or other physical characteristics of the component such as for reinforced concrete shear walls. For metal structures and components, the safety factor included in the codes is usually fairly accurately known as are the relationships between minimum and mean or median strengths. For concrete structures, the factor of safety is normally less accurately known. In this case, the strength of the element is a function of the concrete strength, the amount and strength of the reinforcing steel, and the configuration of the element including the element geometry and reinforcing steel details. In establishing the strength and seismic capacity of concrete components, the results of concrete compression tests and reinforcing steel strength and elongation tests provide a valuable basis for establishing the element strength capacity. However, the increase in concrete strength with age together with the specific details of the element must also be considered. These effects are discussed in more detail in Chapter 4 for structures and Chapter 5 for the piping and equipment.

3.2 DUCTILITY

In order to establish realistic seismic capacity levels for most structures and components, an assessment of the inelastic energy absorption must usually be considered. Exceptions to this are some modes involving brittle failure, functional failure or elastic buckling. However, most failures due to seismic response involve at least some degree of yielding. This is true for reinforced concrete as well as the normally somewhat more ductile metal structures and components.

Consideration of structure ductility typically results in the ability of the structure to withstand greater seismic excitation than would be predicted using linear elastic techniques. In the original design of the TMI-1 structures, all analyses were based on linear elastic behavior of structures. No nonlinear seismic analyses of the structures were conducted. Although inelastic analysis would be desirable in order to more accurately quantify the inelastic effects, the dissipation of inelastic energy may be adequately accounted for without the time and expense of performing nonlinear analyses. This can be accomplished by the use of the ductility-modified response spectrum approach (References 6 and 7) together with a knowledge of the elastic model results and the expected ductility ratios of the critical elements of the structure or component. This approach is based on a series of nonlinear time-history analyses using single-degree-of-freedom models with various nonlinear resistance functions and levels of damping. For different levels of ductility, the reduction in seismic response for the nonlinear system compared to the equivalent elastic system response is calculated. This reduction has been shown to be a function of the frequency and damping of the system as well as the ductility. However, a reasonably accurate assessment of the reduction in response of a structure or component can be made provided the results of the elastic analysis are available and a realistic evaluation of the system ductility can be made. In the current evaluation, the effective ductility was also considered to be a function of the earthquake magnitude.

3.3 SYSTEM RESPONSE

A number of parameters must be evaluated when considering the expected system response near failure compared to the design conditions. Among these are the expected compared to the design earthquake characteristics, directional combinations, system damping, load combinations, and system modeling approaches and assumptions. In addition, the duration of the earthquake must be considered since short duration earthquakes do not possess sufficient energy to fully excite the structural systems. Some of these parameters may be essentially median-centered and introduce little change in the expected seismic capacity while other design criteria may be quite conservative. Several of the more important parameters required in evaluating the system seismic response are discussed below. The factors of safety associated with these parameters are developed in the following chapters for the specific failure modes identified.

3.3.1 Earthquake Characteristics

The TMI-1 Class I structures are founded on bedrock or on compacted backfill and overburden overlying the rock. The Class I structures and equipment were originally designed to the modified 1957 Golden Gate Park earthquake spectra described in Section 1. These spectra, shown in Figure 1-1, were normalized to a peak horizontal ground acceleration of 0.06g for the design earthquake and 0.12g for the maximum earthquake. They were applied to structures founded both on the rock and the backfill.

Recommendations for median-centered peak ground motion parameters and spectral amplification factors for the TMI-1 site are available from Reference 8. This information was used to develop the median-centered site-specific ground response spectra. The same spectra were used for structures founded on rock and structural backfill since the shallow layer of soil is not expected to cause a significant shift in the ground response spectra for the TMI-1 site. The median spectra and the original design spectra scaled to 0.06g are compared in Figure 3-1. It is more informative to compare the spectra giving consideration to the damping values used in the different analyses. The original design spectra from 0% to 20% of critical damping are shown in Figure 3-1. Two percent damping was used in the original design of the reactor building and concrete internal structure for the maximum earthquake while five percent damping was used for the other concrete structures. One-half percent damping was used in the design of the vital piping systems and one and two and one-half percent damping, respectively, were used in the design of the welded and bolted assemblies. These are very conservative values for structures and equipment at response levels approaching failure. The five and ten percent damped median spectra are also shown in Figure 3-1. These values are representative of the range of damping expected at failure. As shown in Figure 3-1, the five percent and ten percent damped design spectra exceed the corresponding five and ten percent damped median spectra at all frequencies. When compared with the original design spectra at the conservative damping values used for design, the median spectra with more realistic damping values show considerable factors of safety were introduced in this phase of the design.

Determination of fragility parameters for structure sliding-induced failure is, in part, dependent on the ratio of the peak ground velocity to the peak effective ground acceleration. A median velocity to acceleration ratio (v/a ratio) of 28 in/sec/g was selected for use in the TMI structural fragility evaluation based upon the recommendations of References 5, 9, and 10 for rock sites. These three sources are based to considerable degree on data from the San Fernando earthquake which had a local magnitude, M_L, of 6.4. Reference 8 recommends median broadbanded response spectra for moderate earthquakes having body wave magnitudes, M_b, of about 6.3 (equivalent to local magnitudes of about 6.5). These events are expected to contribute to most of the seismic risk for the TMI site. A comparison of earthquake magnitude ranges considered in References 5, 9, and 10 indicates these sources should provide appropriate median v/a ratios for TMI-1. The median v/a ratios recommended by these sources range from 24 to 28 in/sec/g. Thus, the 28 in/sec/g is considered to be slightly conservative, but is consistent with recommendations from all three sources as well as the choice of ground response spectra recommended in Reference 8.

In order to develop the lognormal standard deviations appropriate for TMI, v/a ratios for rock sites for earthquakes having magnitude ranges from 5.3 to 6.3 from Reference 11 were used to compute a β_R of 0.37. Because the median values suggested by the various sources are quite close, it is expected the uncertainty should be relatively small. It is estimated that there should be a 95% confidence the median value of v/a will be less than 34 in/sec/g which results in a β_U of 0.12.

3.3.2 System Damping

Damping values used for the design analysis of the TMI-1 plant are shown in Table 3-1. At response levels of structures and equipment near failure levels, the damping ratios used for design are considered conservative when used in conjunction with the ductility factors used in this evaluation. Very little actual test data for damping ratios exist at failure levels, particularly for structures. However, the damping values used for design, even at the higher stress levels, are generally lower compared with median-centered values recommended in References 6, 13 , 14 and 28. These damping values for structures and equipment at or near yield are shown in Table 3-1 in comparison with those used for the design analysis. The median damping values which have been taken from Reference 13 have a range of levels shown in Table 3-1. The lower levels of the pairs of values are considered to be lower bounds while the upper levels are considered to be essentially average values. The values of damping used for the TMI-1 PRA were taken from Table 3-1 assuming the upper level to be a median value. Review of piping damping values derived from experiments supports the use of 5 percent of critical (Reference 28). Electrical and mechanical equipment assemblies have also been shown to have a median damping value of 5 percent of critical (Reference 14).

Damping values used in the TMI-1 fragility evaluation are considered appropriate for structures, equipment, and piping at seismic stress levels at or just below the yield point. Dissipation of inelastic energy at higher response levels is included by consideration of the system ductility as described in Section 3.2. In order to avoid a possible unconservative combination of the two sources of energy dissipation, the structural damping values are not increased as the system response levels rise above the yield point.

3.3.3 Load Combinations

The load combinations on which the original design of the TMI-1 reactor building were based are shown in Table 3-2 (Reference 1). Similar load combinations were used for the other civil structures. For the reactor building structure and much of the equipment contained within the reactor building, these load combinations and those specified by current licensing criteria include a combination of a loss-of-coolant-accident (LOCA) and the SSE loads. Random LOCA events have an extremely low-frequency of occurrence as do seismic events such that the frequency of both events occurring simultaneously is so small that their inclusion is judged to be not important to the risk analysis results.

3.3.4 Modal Combination

Seismic responses of the civil structures represented as multi-degree-of-freedom systems in the original design analysis were determined by the absolute sum of the modal responses (Reference 1). The analysis of piping was conducted using the square-root-of-the-sum-of-the-squares (SRSS) method with the response of closely-spaced modes combined on an absolute sum basis. Current licensing criteria specified in USNRC Regulatory Guide 1.92 (Reference 15) permits the use of the square-root-of-the-sum-of-the-squares (SRSS) method for combining modal responses. For systems whose response is dominated primarily by a single mode, the absolute sum and SRSS methods lead to essentially the same seismic loads. However, the absolute sum method predicts greater seismic loads than the SRSS method for systems whose response is strongly influenced by two or more modes.

SRSS methods are considered to give approximately median-centered results. Although some frequency shifts are expected as structures approach failure, these shifts in frequency are normally not large unless very high ductility ratios exist. Also, the relationship between loads developed from individual modes may be expected to change once nonlinear response levels are reached. In the absence of a nonlinear analysis, the changes in the modal ratios are unknown. For the seismic evaluation of TMI-1, it is assumed that the load response relationships between modes does not change significantly once the structures reach the yield point. For systems where most of the response results from one mode, this assumption introduces negligible possibility for error. For systems with a large number of modes with significant response levels, some additional uncertainty is introduced. The resulting assumed dispersion is discussed in Chapter 4 for structures.

3.3.5 Combination of Responses for Earthquake Directional Components

The horizontal and vertical seismic loadings were assumed to act simultaneously in the TMI-1 original design analyses (Reference 1). Two-dimensional response was considered for the civil structures except that the containment building was treated as a shell of revolution.

Depending on the degree of coupling in the structures, the absolute sum of one horizontal component and the vertical component may be unconservative. Current design procedures are specified in Regulatory Guide 1.92 (Reference 15). One approach permits the effects of two horizontal directional responses to be combined with the vertical response by the SRSS. Other methods of combining directional components such as delineated in Newmark and Hall (Reference 13) also yield realistic results. This approach recommends adding 100% of one-directional component to 40% of the remaining components. This method has the advantage of being easy to use and retains a consistent relationship between loads and stresses. The SRSS, simultaneous time-history, and the 100%, 40%, 40% methods yield similar results and are considered to be essentially median-centered.

3.3.6 Structure Modeling Considerations

The original seismic design analysis of the civil structures was conducted utilizing two-dimensional, lumped mass representations of the buildings. No soil-structure interaction effects were included in the design analyses, either for the structures founded on rock or for the few structures founded on overburden and compacted backfill. The effects of earthquake amplification through the soil layer at TMI for the few structures not founded on bedrock were estimated based on analyses conducted for other sites with soil layers with similar characteristics.

Some aspects of the analysis procedure yield variations which can be quantifiably assessed compared to the design results. For instance, the increase in the actual concrete strength compared to the design values may be used to evaluate the change in stiffness and, hence, the change in frequencies of the concrete structures compared to the design values. The modified frequencies may, in turn, be used to reevaluate the modal responses. Another area where modified responses are considered is in the load distribution for structures where local yielding occurs in some elements before others or through diaphragms containing relatively large cut-outs. The details of these and similar evaluations necessary to account for change between parameter design values and values more representative of seismic response levels near failure are discussed in the following chapters.

TABLE 3-1

COMPARISON OF CRITICAL DAMPING FOR DESIGN AND FAILURE

Component or Structure	Percent of Critical Damping	
	TMI-1 Design (Ref. 1)	Fragility Evaluation* (Refs. 6 & 13)
1. Reactor Building	2.0	7 to 10
2. Concrete Support Structure inside Reactor Building	2.0	7 to 10
3. Structures		
a. Bolted or Riveted	2.5	7 to 15
b. Welded	1.0	5 to 7
4. Vital Piping Stystems	0.5	5
5. Mechanical and Electrical Equipment Assemblies	1.0-2.5	5
6. Other Concrete Stuctures Aboveground	5.0	7 to 10

* Lower values are considered to be approximately lower bounds; upper values are considered to be essentially median-centered.

TABLE 3-2

LOAD COMBINATIONS USED FOR THE TMI-1 REACTOR BUILDING DESIGN

---

a. $C = (1.0 \pm 0.05)\ D + 1.5\ P + 1.0\ T$

b. $C = (1.0 \pm 0.05)\ D + 1.25\ P + 1.0\ T' + 1.25\ E$

c. $C = (1.0 \pm 0.05)\ D + 1.0\ P + 1.0\ \underline{T} + 1.0\ E'$

d. $C = (1.0 \pm 0.05)\ D + 1.0\ W_t + 1.0\ P_t$

Symbols used in the above equations are defined as follows:

C: Required load capacity of section

D: Dead load of structure

P: Accident pressure load

$\underline{T}$: Thermal loads based upon temperature transient associated with 1.5 times accident pressure.

T': Thermal loads based upon temperature transient associated with 1.25 accident pressure.

E: Design earthquake (OBE) (0.06g ground motion)

E': Maximum Hypothetical Earthquake (SSE) (0.12g ground motion)

W_t: Wind loads based on a 390-mph (300 mph times a gust factor of 1.3) tornado. See Subsection 5.2.1.2.6.

P_t: Pressure load based on an external pressure drop of 3 psig between inside and outside of the Reactor Building.

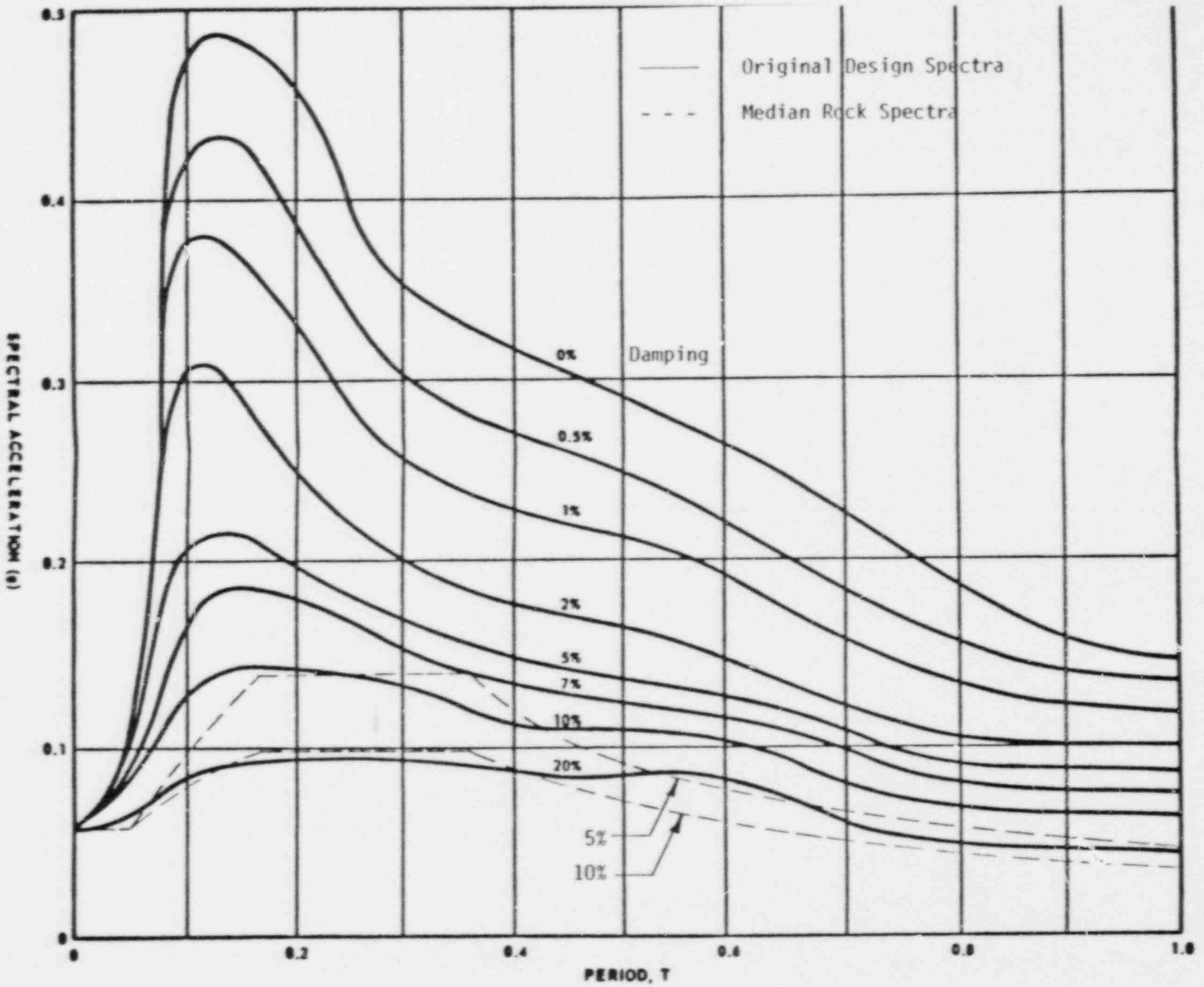


FIGURE 3-1. COMPARISON BETWEEN DESIGN SPECTRA AND MEDIAN SPECTRA

4. STRUCTURES

In this chapter, the median factors of safety and logarithmic standard deviations for the important civil structures are developed. Based on these factors of safety, median acceleration levels associated with seismic failure are presented. For most of these structures, available dynamic models were used to generate seismic response characteristics in order to determine the median factors of safety and logarithmic standard deviations for each of the variables associated with structure response. All seismic analyses were based on linear response model results, but some seismic design loads were modified to more closely approximate the expected inelastic response at the high acceleration levels expected for failure.

4.1 MEDIAN SAFETY FACTORS AND LOGARITHMIC STANDARD DEVIATIONS

As discussed in Section 2.3, the seismic fragilities of structures and components are described in terms of the median ground acceleration, $\check{A}$, and random and uncertainty logarithmic standard deviations, β_R and β_U. In estimating these fragility parameters, it is computationally attractive to work in terms of an intermediate random variable called the factor of safety, F. The factor of safety is defined as the ratio of the ground acceleration capacity, to the Maximum Hypothetical (or Safe Shutdown Earthquake, SSE) acceleration used in the seismic qualification. For equipment and structures qualified by analysis, it is easier to estimate the median factor of safety, $\check{F}$, and variability parameters, β_R and β_U, based upon the original SSE stress analysis than it is to directly estimate the fragility parameters. Thus,

$$\check{A} = \check{F} \cdot A_{SSE} \tag{4-1}$$

From the existing analyses of the important structures together with a knowledge of the deterministic design criteria utilized, median factors of safety associated with the maximum hypothetical earthquake (SSE) ground acceleration of 0.12g can be estimated. These are most conveniently separated into those factors associated with the seismic strength capacity and inelastic energy absorption capability of the structure and those factors associated with the expected building response.

The factor of safety for the structure seismic capacity consists of the following parts:

1. The strength factor, F_s, based on the ratio of actual member strength to the design forces.
2. The inelastic energy absorption factor, F_μ, related to the ductility of the structure and to the magnitude range that is believed to contribute to most of the seismic risk.

Associated with the median strength factor, $\check{F}_s$, and the median ductility factor, $\check{F}_\mu$, are the corresponding logarithmic standard deviations, β_s and β_μ. The structure strength factors of safety and logarithmic standard deviations vary from structure-to-structure and according to the different failure modes of a given structure. Factors of safety for the most important modes of failure are summarized in subsequent sections.

The factor of safety, F_{RS}, related to building response is determined from a number of variables which include:

1. The response spectra used for design compared to the median-centered spectra for the site from multiple seismic events.
2. Damping used in the analysis compared with damping expected at failure.
3. Modal combination methods.

4. Combination of earthquake components.
5. Modeling accuracy.
6. Soil-structure interaction effects.

Based on the characteristics of the lognormal distribution, median factors of safety and logarithmic standard deviations for the various contributing effects can be combined to yield the overall estimates. For instance, the capacity factor of safety of a structure, F_{cap}, is obtained from the product of the strength and inelastic energy absorption factors of safety which, in turn, may include effects of more than one variable.

$$F_{cap} = F_S \times F_\mu \qquad (4\text{-}2)$$

The methods of determining these safety factors are discussed in the following sections. The logarithmic standard deviation on capacity, β_{cap}, is found by:

$$\beta_{cap} = \sqrt{\beta_S^2 + \beta_\mu^2} \qquad (4\text{-}3)$$

As discussed in Section 2.3, the logarithmic standard deviations are composed of both an inherent randomness and uncertainty in the median value.

Median factor of safety, $\check{F}$, and variability, β_R and β_U, estimates are made for each of the parameters affecting capacity and response. These median and variability estimates are then combined using the properties of the lognormal distribution (described in Section 2) in the same manner as Equations 4-2 and 4-3 to obtain the overall median factor of safety and variability estimates required to define the fragility curve for the structure.

For each variable affecting the factor of safety, the random variability, β_R, and the uncertainty, β_U, must be estimated separately. The random variability, β_R, represents those sources of dispersion in the factor of safety which cannot be reduced by more detailed evaluation or by gathering more data. Thus, β_R is due primarily to the variability of an earthquake time-history and, therefore, to a structure's response when the earthquake is only defined in terms of the peak effective ground acceleration. The uncertainty, β_U, represents those sources of dispersion which could be reduced only through better understanding or more knowledge. β_U is associated with such items as our lack of ability to predict the exact strength of materials (concrete and steel) and of structural elements (shear walls and diaphragms); errors in calculated response due to inaccuracies in mass and stiffness representations as well as load distributions; and use of engineering judgment in the absence of plant-specific data on fragility levels.

Each of the factors presented in Chapter 3 will be discussed in more detail in the following sections. Examples are included to assist in the understanding of the application of the methodology.

4.1.1 Structure Capacity

The primary lateral load-carrying systems of the structures that were analyzed are of reinforced concrete construction with the exception of the prestressed concrete reactor building and the field-erected water storage tanks which are fabricated of steel. For lateral load-carrying systems which are composed of concrete, the structure strength is a function of material strengths associated with the concrete and the reinforcing (and prestressing) steel. The determinations of these strengths are presented in the following two sections.

4.1.1.1 Concrete Compressive Strength

The evaluation of the strength of most concrete elements, whether loaded in compression or shear, is based on the concrete compressive strength, f'_c. Concrete compressive strength used for design is normally specified as some value at a specific time from mixing (for example, 28 or 90 days). This value is verified by laboratory testing of mix samples. The strength must meet specified values allowing a finite number of failures per number of trials. As previously stated, there are two major factors which justify the selection of a median value of concrete strength above the design strength.

1. To meet the design specifications, the contractor attempts to create a mix that has an "average" strength above the design strength.
2. As concrete ages, it increases in strength.

The concrete utilized in the construction of the TMI-1 Class I structures was specified to have a minimum compressive strength of 5000 psi or 3000 psi at 28 days. Testing to verify the attainment of this minimum strength was conducted on cylinder specimens. Results of concrete compression testing were available for the TMI-1 Class I structures (Reference 16). Table 4-1 summarizes these results.

As concrete ages, its strength increases. This must also be accounted for in determining the median strength compared to the design strength. Figure 4-1 from Reference 17 shows the increase of the concrete compressive strength with time assuming the concrete poured-in-the-field is adequately represented by the curve designated as "air-cured, dry-at-test." At 28 days, the concrete has a relative strength of 50 percent which approaches 60 percent asymptotically. The median factor relating the strength of aged concrete to the 28-day strength is, therefore, 1.2. No information is available on the standard deviation expected for aging. A logarithmic standard deviation associated with the 28-day aging factors was estimated to be 0.10. Median concrete compressive strengths and variabilities used in the fragility evaluations of the TMI-1 structures are listed in Table 4-2.

Other effects which could conceivably be included in the concrete strength evaluation include some decrease in strength in the in-place condition as opposed to the test cylinder strength, and some increase in strength resulting from rate of loading at the seismic response frequencies of the structure. The variation in the strength of in-place concrete compared with the test cylinder strength is accounted for to a large degree in the use of empirical representations of shear wall capacities. These empirical capacities are typically developed by comparing actual wall strengths to the cylinder test strengths of the wall's concrete. Although experimental data on the in-place and rate effects are limited, that which is available would tend to indicate these effects are relatively small and of the same order magnitude. Since the two effects are opposite, they were neglected.

4.1.1.2 Reinforcing Steel and Post-Tensioning Tendon Yield Strengths

The reinforcement used in the construction of the TMI-1 reactor building and concrete internal structure was specified to be Grade 40. For the other Class I structures, Grade 60 reinforcing steel was specified for 14S and 18S bars with Grade 40 specified for smaller bar sizes. The results of tensile testing conducted on the reinforcement were reported in Reference 16. Based on this data, the median yield strengths, $\check{f}_y$, and logarithmic standard deviations, β, for the reinforcing steel are:

	$\check{f}_y$(ksi)	β
Grade 40	47	0.08
Grade 60	69	0.07

The wire forming the tendons used to post-tension the containment wall and dome was required to conform to ASTM A421-65T, Type BA. This material has a minimum specified tensile strength of 240,000 psi and a minimum yield strength not less than 80 percent of this minimum tensile strength. Only very limited strength test data for the wire used in the construction of TMI-1 reactor building was available. A review of median yield strengths for tendons used in other nuclear plants was conducted. Based on this survey, a median yield strength of 225,000 psi and a logarithmic standard deviation of 0.05 were estimated to be appropriate for use in the TMI-1 fragility evaluation. The very limited TMI-1 plant-specific test data provides reasonable conformance to these values.

Two other effects must be considered when evaluating the yield strength of reinforcing steel. These are the variations in the cross-sectional areas of the bars and the effects of the rate of loading. A survey of information (Reference 18) determined that the ratio of actual to nominal bar area has a mean value of 0.99 and a coefficient of variation of 0.024. The same reference notes that the standard test rate of loading is 34 psi/sec. Accounting for the rate of loading anticipated in seismic response of structures results in a slight decrease in yield strength of reinforcing steel in tension. This effect is neglected in concrete compression.

4.1.1.3 Shear Strength of Concrete Walls

Recent studies have shown that the shear strength of low-rise concrete shear walls with boundary elements are conservatively predicted by the ACI 318-71 code provisions (Reference 19). This is particularly true for walls with height to length ratios in the order of 1 or less. Barda (Reference 20) determined that the ultimate shear strength of low-rise walls tested could be represented by the following relationship:

$$
\begin{aligned}
v_u &= v_c + v_s \\
&= 8.3 \sqrt{f'_c} \quad -3.4 \sqrt{f'_c} \left(\frac{h_w}{\ell_w} -0.5 \right) + \rho_u f_y \qquad (4\text{-}4)
\end{aligned}
$$

where:

v_u = Ultimate shear strength, psi

v_c = Contribution from concrete, psi

v_s = Contribution from steel reinforcement, psi

f'_c = Concrete compressive strength, psi

h_w = Wall height, in

ℓ_w = Wall length, in

ρ_u = Vertical steel reinforcement ratio

f_y = Steel yield strength, psi

The contribution of the concrete to the ultimate shear strength of the wall as a function of h_w/ℓ_w is shown in Figure 4-2. Also shown in Figure 4-2 are the available test values (References 21 through 23) and the corresponding ACI 318-71 formulation. The tests included load reversals and varying reinforcement ratios and h_w/ℓ_w ratios. Web crushing generally controlled the failure of the test specimens. Testing was performed with no axial loads, but an increase in shear capacity of $N/4\ell_w h$ was recommended, where N is the axial load in pounds, and h is the wall thickness in inches.

The contribution of the steel to the ultimate shear strength according to ACI 318-71 is:

$$v_s = \rho_h f_y \qquad (4\text{-}5)$$

where ρ_h = horizontal steel reinforcement ratio.

Furthermore, one of the conclusions reached by Oesterle (Reference 23) is that for low-rise shear walls (specifically, $h_w/\ell_w = 1$), vertical steel has no effect, and the entire contribution to shear strength is due to the horizontal steel.

In order to estimate the effects that the horizontal and vertical steel have, the steel contribution to wall shear strength was determined from test values for the range of $0.5 < h_w/\ell_w < 2$. Test data from the above references were used. The effective steel shear strength was assumed to be in the form:

$$v_{se} = Av_{su} + Bv_{sh} \tag{4-6}$$

where A, B are constants and

$v_{su} = \rho_u f_y$ = vertical steel contribution to shear strength

$v_{sh} = \rho_h f_y$ = horizontal steel contribution to shear strength

The constants A and B were then calculated assuming the concrete contribution to the ultimate strength is given as shown in Equation 4-4. Based on the results of this evaluation, the constants A and B can be shown to be:

$A = 1$	$B = 0$	$h_w/\ell_w \leq 0.5$
$= -2.0\,(h_w/\ell_w) + 2.0$	$= 2.0\,(h_w/\ell_w) - 1.0$	$0.5 \leq h_w/\ell_w \leq 1.0$
$= 0$	$= 1$	$1.0 \leq h_w/\ell_w$

and the median ultimate shear strength is given by:

$$\begin{aligned} v_u &= v_c + v_{se} \\ &= 8.3\sqrt{f'_c} - 3.4\sqrt{f'_c}\left(\frac{h_w}{\ell_w} - 0.5\right) + \frac{N}{4\ell_w h} + \rho_{se} f_y \qquad (4\text{-}7) \end{aligned}$$

where $\rho_{se} = A\rho_u + B\rho_h$ with A and B determined as shown above. Based on an evaluation of the same experimental data, the logarithmic standard deviation was estimated to be 0.15.

The data used to substantiate the median shear strength equations presented above were derived from tests conducted on cantilever walls. The height h_w for these walls is known. However, the walls evaluated in this study typically span more than one story. For these walls, the equivalent cantilever wall height, h_{we} was taken as the ratio of the in-plane moment to the in-plane shear at the section under consideration. The equivalent height h_{we} was used to determine the median wall shear strength and provides a more accurate representation of the moment-shear interaction.

4.1.1.4 Example of Shear Wall Failure in Shear

The determination of the median shear strength of the east-west interior wall on Column Line 11 of the control building for the story from EL. 282 ft to EL. 306 ft is selected as an example. This wall is 3 feet thick and 68 feet long. It is reinforced by No. 7 bars spaced 18 inches apart at both faces in the vertical direction and No. 8 bars spaced 18 inches apart at both faces in the horizontal direction. The median concrete compressive strength is 5900 psi and the median reinforcement yield strength is 47 ksi. The equivalent cantilever wall height was estimated to be approximately 81.6 ft. The effect of the axial load acting on the wall was neglected with slight conservatism resulting. The median concrete shear strength was found to be:

$$v_c = 8.3\sqrt{5900} - 3.4\sqrt{5900}\ (81.6/68 - 0.5)$$

$$= 455 \text{ psi}$$

The effective wall height-to-length ratio is greater than 1.0 so the steel shear strength was based upon the horizontal reinforcement.

$$\rho_{se} = \rho_h$$

$$= \frac{2(0.79)}{36(18)}$$

$$= 0.0024$$

The steel shear strength was found to be:

$$v_{se} = 0.0024\ (47{,}000)$$

$$= 113 \text{ psi}$$

For rectangular walls with uniformly distributed vertical reinforcement, the effective depth, d, from the extreme compressive fiber to the resultant of the tension force was taken to be 0.8 ℓ_w from Reference 19. The median wall shear strength was found to be:

$$V_u = (455 + 113)(0.8)(68)(3)(144)(10^{-3})$$

$$= 13{,}300 \text{ k}$$

The applied shear load based on an elastic load distribution was found to be 930 k. The median strength factor corresponding to shear failure of this wall was then determined to be 14.

4.1.1.5 <u>Strength of Shear Walls in Flexure Under In-Plane Forces</u>

Equations to predict the overturning (in-plane) moment capacity of rectangular shear walls containing uniformly distributed vertical reinforcement are found in Reference 22. These equations were derived from the basic ultimate strength design provisions for reinforced concrete members subjected to flexure and axial loads contained in Section 10.2 of ACI 318-71 (Reference 19). These provisions are based upon the satisfaction of force equilibrium and strain compatibility.

Equation 1 of Reference 22 can be used to predict the flexural strength of rectangular walls having uniformly distributed reinforcement. The accuracy of this equation has been verified by testing. Equation 2 of Reference 22, shown as Equation 4-8 below, was presented as an adequate approximation to Equation 1.

$$M_u = 0.5\, A_s f_y \ell_w \left(1 + \frac{N_u}{A_s f_y}\right)\left(1 - \frac{c}{\ell_w}\right) \text{in-lb} \qquad (4\text{-}8)$$

where

A_s = Total area of vertical reinforcement at section, sq. in.

f_y = Yield strength of vertical reinforcement, psi

ℓ_w = Horizontal length of wall, in.

c = Distance from extreme compressive fiber to neutral axis, n

N_u = Axial load, positive in compression, lb.

f'_c = Compressive strength of concrete, psi

Inspection of Equation 4-8 reveals that the overturning moment capacity of a rectangular wall can be adequately represented by lumping the total area of the uniformly distributed vertical reinforcement at midlength of the wall and applying the basic design provisions in Section 10.2 of ACI 318-71.

$$M_u = (A_s f_y + N_u)\left(\frac{\ell_w}{2} - \frac{\beta_1 c}{2}\right) \qquad (4\text{-}9)$$

where β_1 is the ratio of the depth of the equivalent rectangular concrete stress block to the distance to the neutral axis (c).

This approach was typically used to predict the median flexural strength for walls without concentrated reinforcement. Concentrated reinforcement can be vertical wall reinforcement bars within the effective flanges of the cross walls cast integrally with the wall evaluated. The compression flange steel is typically neglected since it is near the neutral axis, and its effect on the moment capacity is small. The total moment capacity of reinforced concrete shear walls including tensile flange steel is then:

$$M_u = (A_s f_y + N_u)\left(\frac{\ell_w}{2} - \frac{\beta_1 c}{2}\right) + A_f f_y\left(d - \frac{\beta_1 c}{2}\right) \qquad (4\text{-}10)$$

where

A_f = Area of flange steel

d = Distance from the extreme compressive fiber to the centroid of tensile flange steel.

For the fragility evaluation of TMI-1 structures, flanges were for the most part neglected with slight conservatism resulting.

4.1.1.6 Example of Shear Wall Failure in Flexure

The same wall that was analyzed for shear in Section 4.1.1.4 will be analyzed for flexure. This east-west interior wall along Column Line 11 has a 5-foot thick cross wall along Column Line F and a 2'-6" thick cross wall along Column Line H1. The vertical wall reinforcement bars are #18 at 18" each face for the 5-foot cross wall and #7 at 18" each face for the 2'-6" cross wall. It is expected that the flexural strength of this wall will be lower for in-plane overturning moment causing tension in the 2.5-foot cross wall. As discussed in Section 4.1.1.5, the compression flange is neglected here with slight conservatism resulting. Thus, the compressive stress block is assumed to be contained within the width of the wall web which is 3-foot thick.

The effective tensile flange width of the 2'-6" cross wall along Column Line H1 was estimated from the recommendations in Reference 19, 41, and 42. An effective width of 29.5 feet was selected. The distance from the extreme compressive fiber to the centroid of the flange reinforcement bars was estimated to be 66.8 feet.

The total axial load acting on the 3-foot E-W direction interior wall was found to be 6962 kips based on the tributary gravity load of the floors and walls above the section evaluated. The line of action of the axial load was found to be at a distance 41.1 feet from the extreme compressive fiber. This axial load was reduced by the vertical ground acceleration acting upward. The vertical ground acceleration was taken to be 2/3 of the horizontal ground acceleration, A. The effect of this vertical direction response was then combined with the concurrent response in the horizontal direction using the 100%, 40%, 40% method discussed in Section 3.3.5.

The values to be used to evaluate the flexural strength of this wall are as follows:

$f_y = 47$ ksi

$f'_c = 5900$ psi

$$\beta_1 = 0.85 - 0.05\left(\frac{5900-4000}{1000}\right)$$

$$= 0.76$$

$\ell_w = 68$ ft

$A_s = 48\ \text{in}^2$ 80 - #7 vertical wall reinforcement bars

$$A_f = \left(\frac{29.5\text{x}12}{18}\right) \text{ x } 2\text{x}0.6$$ #7 vertical wall reinforcement bars spaced at 18 inches on both faces

$$= 23.6 \text{ in}^2$$

$$d = 66.8 \text{ ft}$$

$$N_u = 6962 \left[1 - 0.4 \left(\frac{2}{3} A\right)\right]$$

$$d_w = 41.1 \text{ ft}$$ distance from the extreme compressive fiber to the line of action of the axial load

By solving for force equilibrium

$$c = 6.3 - 1.1 A \quad \text{ft}$$

$$\frac{\beta_1 c}{2} = 2.4 - 0.42 A \quad \text{ft}$$

$$M_u = 48(47)[34-(2.4-0.42A)] + 6962 \left[1 - 0.4\left(\frac{2}{3}A\right)\right][41.1 -$$

$$(2.4-0.42A)] + 23.6(47) [66.8 -(2.4-0.42A)]$$

$$= 412150 - 67510A - 780 A^2$$

The overturning moment acting on this wall due to the 0.12g peak ground acceleration was found to be 76,200 k-ft from an elastic load distribution. The elastic applied load, M, due to some peak ground acceleration A is then:

$$M = 76{,}200 \left(\frac{A}{0.12}\right)$$

$$= 635{,}000A$$

The acceleration, A_y, at which the wall's capacity against elastic loads is mobilized can be found by equating the available overturning moment resistance, M_u, to the applied elastic load, M, and solving for A.

$$M_u = M$$

$$412{,}150 - 67{,}510\, A_y - 780\, A_y^2 = 635{,}000\, A_y$$

$$A_y = 0.59 \quad (g)$$

$$M_u = 375{,}000 \text{ k-ft}$$

$$\check{F}_s = \frac{A_y}{0.12g}$$

$$= \frac{0.59}{0.12}$$

$$= 4.9$$

This value is less than the median strength factor against shear failure of 14 calculated in Section 4.1.1.4. Consequently, flexure failure is the controlling failure mode for this wall. It is to be noted that after this wall yields, load will redistribute to other walls of the structure. These other walls have substantially higher strength than the wall evaluated here. Not accounting for this load redistribution results in some conservatism in the evaluation of flexure strength of this wall.

4.1.1.7 Structure Sliding

Resistance to structure sliding is provided by static friction between the structure foundation and the rock or soil below, lateral earth pressures from backfill placed against exterior walls, and any shear keys embedded into rock or soil. Gross structure sliding initiates when the base shear acting at the foundation-rock or soil interface equals the available resistance. Initiation of sliding does not constitute structure or equipment failure. As a structure slides as a rigid body, its accelerations and relative story drifts cannot exceed those values occurring at the initiation of sliding. Failure modes resulting from structure sliding are displacement-dependent. For example, piping attached at one end to the structure that is sliding and at the other end to some adjacent structure may fail under relative end displacement. Also, impact with adjacent structures may cause concrete spalling and subsequent damage to equipment or piping mounted near the localized spalling regions. However, the sliding displacements necessary to cause these failure modes are substantial and can occur only under peak ground accelerations well in excess of acceleration levels initiating sliding.

An approach recommended by Newmark (Reference 24) was used to predict structure sliding displacements. This approach is simple and results in conservative estimates of the sliding displacement for single acceleration pulses. Figure 4-3 summarizes the features of Newmark's approach. The ground beneath the structure experiences a single horizontal acceleration pulse, Ag, that lasts for a time duration t_1, and results in a velocity V. The structure is represented as a rigid body that begins to slide relative to the ground when its rigid body acceleration reaches Ng, where N is a coefficient relating the net sliding resistance to the total structure weight. Since the ground acceleration is conservatively assumed to be a square pulse, sliding initiates instantaneously. Structure sliding ends at time t_m when the structure has achieved the ground velocity, V. The relative displacement between the ground and the structure is determined by integrating the relative velocity between the ground and the structure from time $t = 0$ to time $t = t_m$.

With estimates of the net sliding resistance coefficient, N, and the ground velocity, V, as a function of the peak ground acceleration, Equation 4-11 (see Figure 4-3) can be used to determine the ground acceleration resulting in sufficient relative sliding displacement, u_m, to cause the failure mode under consideration. Since the ground acceleration is actually a reversing function rather than a single pulse, this would tend to reduce the sliding time duration and thus the relative sliding displacement. However, for earthquakes in the Magnitude 5.8 to 6.8 range, several strong motion cycles can occur. Depending on the phasing of the input motion and structure response, some preferential motion (or ratcheting) may possibly occur. This could only be determined analytically for a given structure by conducting a series of time-history analyses using actual earthquake records scaled to different accelerations as inputs. This was considered to be beyond the scope of the current investigation.

Due to the highly uncertain nature of structure behavior past the initation of sliding, the logarithmic standard deviation associated with Newmark's approach was estimated to be 0.4. Because no relative displacement can occur until sliding initiates, the acceleration capacity corresponding to the initiation of sliding can be treated as a cutoff on the fragility curve for sliding-induced failure in a manner similar to that described in Section 2.3.

Many of the TMI-1 structures are not expected to slide. For example, the reactor building containment is embedded in bedrock and, in addition, has a sump keyed into the rock. With the exception of the diesel generator building, the capacities of the other structures are controlled by failure modes other than sliding. The capacity of the diesel generator building was found to be controlled by sliding towards and subsequent impact with the intermediate building.

4.1.1.8 Example of Sliding-Induced Failure

Determination of the ground acceleration causing impact between the intermediate building and the diesel generator building due to sliding of the latter structure will be presented as an example of the application of Newmark's approach. Accounting for the sliding resistance provided by the shear strength of the compacted backfill confined by the shear keys below the mat foundation, the sliding resistance coefficient was found to be:

$$N = 0.839 - 0.267\ A$$

The second term above accounts for the reduction in sliding resistance associated with a vertical seismic acceleration acting upward. As discussed in Section 3.3.1, the peak bedrock velocity corresponding to a peak ground acceleration of 1g was estimated to be 28 in/sec. The peak ground acceleration at the top of the compacted backfill was estimated to be about 1.2 times the bedrock peak ground acceleration due to amplification of the ground motion by the overburden. However, relatively little amplification was expected for the ground velocity. More detailed discussion of the soil amplification of the diesel generator building overburden is presented in Section 4.1.7.1. There is a 1.5 inch gap between the diesel generator building and the intermediate building that is filled with styrofoam. After the diesel generator building slides and closes this gap, any additional sliding displacement will cause crushing or spalling of the concrete. It was estimated that a total of 2.5 inches of sliding displacement will correspond to failure of the 1'-3" thick south wall of the diesel generator building and damage to the safety-related ducting and piping supported by this wall. The acceleration capacity was found by solving for A using Equation 4-11:

$$
\begin{aligned}
u_m &= 2.5 \text{ inches} \\
V &= 28 \text{ in/sec/g} \\
&= 28\,A \\
&= 28\,A_r \\
A &= 1.2\,A_r
\end{aligned}
$$

where A_r and A are the bedrock peak ground acceleration and the free surface ground acceleration, respectively.

$$u_m = \frac{V^2}{2gN}\left(1 - \frac{N}{A}\right) \qquad (4\text{-}11)$$

$$0 = \frac{V^2}{2gN} - \frac{V^2}{2gA} - u_m$$

$$= \frac{(28\ A_r)^2}{2(386.4)[0.839-0.267(1.2\ A_r)]} - \frac{(28\ A_r)^2}{2(386.4)(1.2\ A_r)} - 2.5$$

$$A_r = 1.3$$

$$F_S = \frac{1.3}{0.12}$$

$$\approx 11$$

4.1.2 Structure Ductility

A much more accurate assessment of the seismic capacity of a structure can be obtained if the inelastic energy absorption of the structure is considered in addition to the strength capacity. One tractable method involves the use of ductility modified response spectra to determine the deamplification effect resulting from the inelastic energy dissipation. Early studies indicated the deamplification factor was primarily a function of the ductility ratio, μ, defined as the ratio of maximum displacement to displacement at yield. More recent analytic studies (Reference 7) have shown that for single-degree-of-freedom systems with resistance functions characterized by elastic-perfectly plastic, bilinear, or stiffness-degrading models, the shape of the resistance function is, on the average, not particularly important. However, as opposed to the earlier studies, more recent analyses have shown the deamplification factor is also a function of the system damping.

The Riddell-Newmark ductility modified response spectra approach can be used to predict the inelastic energy absorption factor, F_μ, corresponding to some ductility ratio, μ, in the following manner:

$$F_\mu = [p\mu - q]^r \qquad (4\text{-}12)$$

where $p = q+1$

$q = 3.0\gamma^{-0.30}$ in the amplified acceleration region.

$= 2.7\gamma^{-0.40}$ in the amplified velocity region.

$r = 0.48\gamma^{-0.08}$ in the amplified acceleration region.

$= 0.66\gamma^{-0.04}$ in the amplified acceleration region.

γ = percent of critical damping.

For systems in the amplified acceleration region of the spectrum (i.e., between about 2 Hz and 10 Hz), Figure 4-4 from Reference 7, shows the deamplification function for several damping values as a function of the ductility ratio.

One drawback of the ductility modified response spectra approach is that it does not reflect the relationship between earthquake magnitude and ductility. It is well known that lower magnitude earthquakes are not as damaging to structures and equipment as higher magnitude earthquakes with the same peak ground accelerations. The reason for this is that the lower magnitude response spectra have lower energy content and shorter durations which develop fewer strong response cycles. Structures and equipment are able to withstand larger deformations (i.e., higher ductility) for a few cycles compared to the larger number of cycles resulting from longer duration events.

The method used in the TMI-1 fragilities evaluation to account for this effect was based on the use of an effective ductility, μ^*, in conjunction with the Riddell-Newmark ductility modified spectra approach. The following formulation was developed to calculate the effective ductility.

$$\mu^* = 1.0 + C_D(\mu - 1.0) \qquad (4\text{-}13)$$

where the duration correction factor, C_D, is a function of the earthquake magnitude.

A limited amount of research is available for use in developing C_D factors. In Reference 26, structures with elastic frequencies of approximately 2, 3, 5 and 8 Hz were subjected to 12 earthquake records scaled to sufficient intensity to produce ductility ratios of approximately 1.9 and 4.3. Included was one artificial record which developed response spectra which envelope the US NRC Reg. Guide 1.60 spectra. The C_D factors used in the TMI-1 fragilities evaluation were based on the results from Reference 26. C_D is considered to be frequency-independent based on these limited data.

The factor of safety resulting from ductility effects, F_μ, is dependent on both duration and spectral shape. Figure 4-5 is reproduced from Reference 26 and clearly shows the effect of strong motion duration for a ductility ratio of approximately 4.3. However, F_μ is most strongly influenced by the spectral shape and the frequency of the structure. Tables 4-3 and 4-4 also reproduced from Reference 26, show the F_μ factors for the various earthquake records and structure frequencies for the 1.9 and 4.3 ductility ratios, respectively. It is inappropriate to include results from Reference 26 for frequencies which lie in a steeply rising or falling portion of a sharply peaked region of the response spectra. As a structure reaches significant levels of inelastic response, there is a decrease in the resonant frequency of the structure. If the elastic frequency of the structure is in a portion of the response spectrum where the frequency shift results in lower response, a relatively higher F_μ will be developed. Conversely, if the elastic frequency of the structure lies in a region of the response spectrum where the frequency shift results in increased response, a relatively lower F_μ will be predicted. A review of the data from Reference 26 indicates that many of the F_μ factors shown in Tables 4-3 and 4-4 do, in fact, lie in steeply rising or falling regions of the response spectra.

The TMI-1 median ground response spectra, however, are relatively broadband and contain significant energy throughout the frequency range from approximately 2.5 Hz to over 10 Hz. Thus, even though a number of structures at TMI-1 have relatively high fundamental elastic frequencies, it is incorrect to use all the F_μ factors directly from the results from Reference 26 together with the Riddell-Newmark method and the TMI-1 median spectra. For earthquakes in the magnitude 6.5 to 7.5 range from Reference 26 considered appropriate for eastern U.S. sites such as TMI, an average value of F_μ of approximately 2.2 is indicated. For the appropriate earthquakes in the magnitude 4.5 to 6.0 range, an average value of F_μ of about 2.9 results for ductilities of about 4.3.

Using the Riddell-Newmark formulation for F_μ given above together with the 4.27 ductility ratio and 7 percent of critical damping used in Reference 26, a value of about 2.55 was calculated. For earthquakes in the 4.5 to 6.0 range, an effective ductility of about 5.6 results using the Reference 26 results which yields an "effective" ductility of about 5.6, or a duration correction coefficient of 1.4. Similarly, for earthquakes in the magnitude 6.5 to 7.5 range, an effective ductility of about 3.2 with duration coefficient of 0.7 is indicated. The majority of seismic risk for the TMI-1 plant is expected to result from earthquakes centered around the magnitude 6.3 range. Linear interpolation was used for the 1.4 and 0.7 factors to yield an effective duration coefficient considered appropriate for TMI-1 of about 1.0.

The following definition of the inelastic energy absorption factor was used for the TMI-1 structures whose fundamental frequencies are within the amplified acceleration region:

$$F_\mu = \frac{S_{a_e}}{S_{a_\mu}} \qquad (4\text{-}14)$$

S_{a_e} = Spectral acceleration from the elastic response spectrum for the fundamental structure mode having a frequency in the amplified acceleration region.

S_{a_μ} = Deamplified spectral acceleration accounting for nonlinear structure response.

= Greater of $S_{a_{\mu,A}}$ or $S_{a_{\mu,RIG}}$

$$S_{a_{\mu,A}} = (p\mu^{*} - q)^{-r} S_{a_e} \qquad (4\text{-}15)$$

$$S_{a_{\mu,RIG}} = (\mu^{*})^{-0.13} (PGA) \qquad (4\text{-}16)$$

p,q,r - Equation 4-12

PGA = Peak ground acceleration

Equation 4-16 is also presented in Reference 7.

4.1.2.1 Example of the Inelastic Energy Absorption Factor

As an example, the derivation of the inelastic energy absorption factor for E-W interior shear wall failure of the control building will be shown. This failure mode is expected to have a median system ductility of approximately 3.5. Response of the structure in the E-W direction is dominated by the fundamental mode which has a frequency of about 11 Hz. The median damping ratio was estimated to be 10 percent of the critical damping.

S_{a_e} = 0.176g from the 10% damped median site-specific spectrum

$\check{\mu} = 3.5$

$\check{\mu}^{*} = 1.0 + 1.0(3.5-1)$

$= 3.5$

$q = 3.0(10)^{-0.30}$

$= 1.5$

$p = 1.5 + 1$

$= 2.5$

$$r = 0.48(10)^{-0.08}$$

$$= 0.40$$

$$S_{a_{\mu,A}} = [2.5(3.5)-1.5]^{-0.4} \quad (0.176g)$$

$$= (0.453)(0.176g)$$

$$= 0.08g$$

$$S_{a_{\mu,RIG}} = (3.5)^{-0.13} \quad (0.12g)$$

$$= (0.850)(0.12g)$$

$$= 0.102g$$

$$S_{a_{\mu}} = S_{a_{\mu,RIG}} = 0.102g$$

$$\tilde{F}_{\mu} = \frac{0.176}{0.102}$$

$$= 1.7$$

4.1.3 Structure Response Used for Structure Fragility Evaluations

Determination of the structure response factors and their variabilities in fragility evaluations is typically performed using structure responses predicted by the original design dynamic analyses. No design information regarding the TMI-1 Class I structure loads was available to permit assessment of the structural fragilities except for the reactor building containment. As an alternative, details of the original design dynamic models and eigensolutions were obtained. Eigensolutions predicted using the model information supplied were generated and compared to the original design eigensolutions to verify that the dynamic model information was correctly interpreted. Median-centered overall structure loads were then developed using the median-centered methods described in Section 3. These structure loads were then used to determine the median strength factors in the structures fragilities evaluations.

Because median structure responses were used directly, median response factors were taken to be unity in the structures fragility evaluations. An exception is the reactor building containment. The original design response reported in the FSAR was used in the fragility evaluation of this structure. Also, equipment fragility evaluations were typically performed on the basis of the original design in-structure response spectra since generation of median-centered, in-structure spectra would require greater effort than warranted.

The following discussion describes the determination of the median structure response factors based upon comparison of the median versus design responses. This convention is retained for the benefit of understanding the structure response factors used in the reactor building containment and equipment fragility evaluations.

4.1.4 Spectral Shape, Damping, and Modeling Factors

As previously discussed, the important TMI-1 structures were designed using the ground response spectra shown in Figure 1-1. For the SSE, five percent of critical damping was used for the reinforced concrete structures except for the reactor building and the concrete internal structure where two percent of critical damping was used. For the reinforced concrete comprising the lateral load-carrying structures for TMI-1, ten percent of critical damping is considered to be the median value expected at response levels near failure (Reference 13). As shown in Figure 3-1, the ten percent damped median-centered response spectrum scaled to 0.06g is below the five percent damped original design OBE spectrum at all frequencies. The frequencies predicted by the TMI-1 original design dynamic models were available. The spectral shape factor for each structure was based on the mode or modes contributing to most of the seismic response. The spectral shape factor at the frequency under consideration is given by:

$$F_{SS} = \frac{S_{D_\zeta}}{S_{M_{\zeta=10\%}}} \qquad (4\text{-}17)$$

where S_{D_ζ} represents the design spectral acceleration at the design damping value used for the structure evaluated and $S_{M_{\zeta=10\%}}$ represents the estimated spectral acceleration associated with the median site-specific response spectrum for 10 percent damping. As noted in Section 4.1.3, structure loads used in the structure fragility evaluations of all structures except the reactor building containment were derived from the median-centered response spectrum. The median spectral shape factors for these structures are therefore unity.

In computing the spectral shape factor of safety, it is convenient to combine the damping and ground response spectrum effects. In the development of logarithmic standard deviations on spectral shape, however, it is informative to consider the damping effects separately. This implies a factor of safety of unity on damping alone since it has already been included in the factor of safety on spectral shape.

The logarithmic standard deviation on spectral acceleration, β_{SA}, may be estimated from Reference 13.

$$\beta_{SA} = \ln \frac{S_{M+1\sigma}}{S_M} \qquad (4\text{-}18)$$

where $S_{M+1\sigma}$ is the spectral acceleration from the ten percent damped mean plus one standard deviation (84 percentile) site-specific spectrum, and S_M is the spectral acceleration from the ten percent damped median site-specific spectrum.

The deviation on spectral acceleration resulting from damping, β_ζ, can be estimated from:

$$\beta_\zeta \simeq \ln \frac{S_{M_{\zeta=7\%}}}{S_{M_{\zeta=10\%}}} \qquad (4\text{-}19)$$

where $S_{M_{\zeta=7\%}}$ is the spectral acceleration from the median site-specific spectrum at seven percent damping, and $S_{M_{\zeta=10\%}}$ is the spectral acceleration from the ten percent damped median site-specific spectrum. Seven percent damping is estimated by Reference 13 to be one standard deviation below the median damping value of ten percent for reinforced concrete structures at yield. The randomness and uncertainty components of β_ζ are judged to be approximately equal. Thus,

$$(\beta_R)_\zeta = (\beta_U)_\zeta = \frac{\beta_\zeta}{\sqrt{2}}$$

The original design dynamic models of the TMI-1 structures were typically determined to be adequate to predict the seismic response. In generating loads for the structures fragility evaluations, model modifications were incorporated if necessary. Modeling factors of unity typically were used.

Variability in modeling predominantly influences the calculated mode shapes and modal frequencies. Since the concrete strength and, consequently, the stiffness of the structures is above the design values, calculated frequencies would be expected to be somewhat less than actual values, at least for low-to-moderate levels of response. At response levels approaching failure, softening of the structures due to concrete cracking occurs, and for structures analyzed using uncracked section properties, some decrease in the actual frequencies compared to the calculated values is expected. Calculated frequencies were generally assumed to be median-centered unless material properties used in the original analyses differ from the material properties calculated from test data enough to significantly change the calculated frequencies. The mode shapes were assumed to stay the same regardless of whether or not frequencies changed.

Modeling uncertainties from both the mode shapes and modal frequencies enter into the uncertainty on calculated modal response as defined by β_M. Thus,

$$\beta_M = \sqrt{\beta_{MS}^2 + \beta_{MF}^2} \qquad (4\text{-}20)$$

where β_{MS} and β_{MF} are estimated logarithmic standard deviations on structural response of a given point in the structure due to uncertainties in mode shape and due to uncertainties in modal frequency, respectively. Based upon experience in performing similar analyses, β_{MS} was estimated to be typically about 0.15. The modal frequency variability shifts the frequency at which spectral accelerations are to be determined, so that:

$$\beta_{MF} \approx \ell n \left(\frac{S_{M_{f=f_\beta}}}{S_{M_{f=f_M}}} \right) \qquad (4\text{-}21)$$

where f_M is the median frequency estimate, and f_β is the 84 percent exceedance probability frequency estimate. The logarithmic standard deviation on frequency was typically estimated to be approximately 0.30 for the structures evaluated.

4.1.4.1 Example of Spectral Shape, Damping, and Modeling Factors

As an example, determination of the spectral shape, damping, and modeling factors and variabilities appropriate for failure modes associated with the containment wall of the reactor building will be presented. Review of the modal responses indicated that nearly all of the response quantities are associated with the fundamental mode. This mode was found to have a median elastic frequency of 4.2 Hz compared to the original design frequency of 3.8 Hz. This frequency shift can be attributed to the increase in structure stiffness associated with the median rather than design concrete compressive strength.

As noted in Section 4.1.3, evaluations of the structural failure modes for the reactor building were based on the design structure loads at 2 percent damping. For failure modes whose fragilities were derived from the original design basis, the median spectral acceleration of 0.42g at the original design frequency of 3.8 Hz for two percent design damping with the median spectral acceleration of 0.20g at the median frequency of 4.2 Hz for ten percent median damping:

$$\check{F}_{SA} = \frac{0.42}{0.20} = 2.1$$

From information defining the median-centered spectrum for the TMI-1 site, the variability associated with randomness was estimated as shown below:

$$\beta_R = \ln \frac{0.24}{0.20} = 0.18$$

where 0.24g and 0.20g are the spectral accelerations at the median frequency obtained from the ten percent damped mean plus one sigma and median site-specific spectra, respectively.

$$\beta_U \approx \frac{2}{3} \beta_R = 0.12$$

The composite variability associated with damping was based on a comparison of the median spectral acceleration of 0.23g for seven percent damping at the median frequency of 4.2 Hz with the median spectral acceleration of 0.20g for ten percent damping.

$$\beta_\zeta = \ln\left(\frac{0.23}{0.20}\right)$$

$$= 0.14$$

$$(\beta_R)_\zeta = (\beta_U)_\zeta = \frac{\beta_\zeta}{\sqrt{2}} = 0.10$$

For this structure, uncertainty on frequency was estimated to be 0.30. The $+1\beta$ and -1β frequencies were found to be:

$$F_{+1\beta} = 4.2\ e^{0.30}$$

$$= 5.7\ \text{Hz}$$

and

$$F_{-1\beta} = 4.2\ e^{-0.30}$$

$$= 3.1\ \text{Hz}$$

The $\pm 1\beta$ frequency range was found to be within the amplified acceleration region of the median site-specific spectrum where the spectral acceleration is constant. Consequently, modeling uncertainty due to frequency was estimated to be essentially zero.

$$\beta_{mf} = 0$$

This value was combined with the estimated modeling uncertainty associated with mode shape of 0.15 to give the total modeling uncertainty:

$$\beta_M = \sqrt{0^2 + 0.15^2}$$

$$= 0.15$$

4.1.5 Modal Combination

The seismic design analysis of TMI-1 structures was performed by response spectrum analysis; therefore, phasing of the individual modal responses was unknown. Most current design analyses are normally conducted using response spectra techniques. The current recommended practice of the USNRC as given in Regulatory Guide 1.92 (Reference 15) is to combine modes by the square-root-of-the-sum-of-the-squares (SRSS). For the TMI-1 structures, the absolute sum method was used to combine the modal responses whereas for the equipment, the SRSS method was used as discussed in Section 3.3.4. Many studies have been conducted to determine the degree of conservatism or unconservatism obtained by use of SRSS combination of modes. Except for very low damping ratios, these studies have shown that SRSS combination of modal responses tends to be median centered. The coefficient of variation (approximate logarithmic standard deviation) tends to increase with increasing damping ratios. Figure 4-6 (taken from Reference 27) shows the actual time-history calculated peak response versus SRSS combined modal responses for structural models with four predominant modes. Based upon these and other similar results, it is estimated that for ten percent structural damping, the SRSS response is median-centered. The median modal combination factor of safety was therefore taken to be 1.0 for equipment fragilities based on the original design information. For the reactor building for which the original design loads were available, it was observed that the response was dominated primarily by the fundamental mode such that the absolute sum and SRSS methods lead to essentially the same seismic loads. Consequently, the median modal combination factor of safety was taken to be 1.0. The SRSS method of modal combination was used to develop the median structure loads for other Class I structures as described in Section 4.1.3. The median modal combination factor for structural fragilities based on these loads was also taken to be 1.0. Where individual modal responses were known, the absolute sum of these responses was used to estimate the coefficient of variation. The absolute sum is an upper bound estimated to be three standard deviations above the median SRSS response.

4.1.6 Combination of Earthquake Components

The design of the essential TMI-1 structures was based on loads developed from the absolute sum of one horizontal component and vertical component of ground motion. Current licensing requirements consist of the SRSS combination of responses from three principal directions (Reference 15). Alternatively, it is recommended (Reference 13) that directional effects be combined by taking 100 percent of the effects due to motion in one direction and 40 percent of the effects from the two remaining principal directions of motion. This was considered the median condition for the current evaluation.

Depending on the geometry of the particular structure under consideration together with the relative magnitude of the individual load or stress components, the expected stresses due to the 100%, 40%, 40% method of load combinations are decreased when compared with those calculated using the original design method. For shear wall structures where the shear walls in the two principal directions act essentially independently and are the controlling elements, the two horizontal loads do not combine to a significant degree except for the torsional coupling. Thus, only the vertical component affects the individual shear wall stress. A moderate amount of vertical load slightly increases the ultimate shear load carrying capacity of reinforced concrete walls, while the overturning moment capacity may be more significantly affected. Typically, the effect of the vertical dead load on the wall capacities was conservatively neglected. In these cases, the effect of the vertical seismic component on the capacities and the earthquake component combination variabilities was not included since these capacities already contain conservatism due to not including the dead load. In other cases, such as the control building interior wall described in Section 4.1.1.6, the increase in capacity due to the dead load was included. For these cases, the effect of the vertical seismic response on the capacity and the earthquake component combination variability was also included.

The coefficient of variation is calculated in the same manner as it was for the modal combination factor. The absolute sum of the three components is an upper bound, estimated to be three standard deviations above the median.

4.1.7 Soil-Structure Interaction Effects

Two types of soil-structure effects are considered in the analysis of nuclear power stations. The first involves the variation in frequency and response of the structure due to the flexibility of the soil and the dissipation of energy into the soil by geometric damping. For structures founded on competent bedrock such as most of the TMI-1 Category I structures, these effects are usually small and are typically neglected in current design analyses. A second effect is the amplification of the bedrock motion through the soil. Again, for structures founded directly on the bedrock, essentially no amplification occurs, and the motion is normally specified at the foundation level as was done in the design of the TMI-1 structures. Thus, the design of the TMI-1 structures founded on rock was conducted using current state-of-the-art assumptions and methods of analysis in regard to the soil-structure interaction effects. The median seismic loads acting on the diesel generator building and the field-erected tanks were determined using fixed-base models. The effects of soil-structure interaction on the seismic response of these structures were accounted for as described in Sections 4.1.7.1 and 4.1.7.2.

4.1.7.1 Soil Amplification

The free-field peak ground acceleration to which the seismic hazard curves are anchored is assumed to be at the bedrock where most of the TMI-1 Class I structures are founded. The backfill and overburden upon which the diesel generator building and field-erected tanks are founded is about 30 feet deep overlying the bedrock. Some limited amplification of the bedrock acceleration through this soil layer to the free ground surface is expected. However, an evaluation to determine the increase of the peak acceleration through the depth of the soil layer was not performed for the original design analysis.

A review of other nuclear plant sites that have soil overburden with similar characteristics to that at the TMI-1 site was conducted as part of the fragility evaluation. For these sites, the computer program SHAKE was used to determine the amplification of the rock motion by the soil layer, including the effects of strain degradation. Based on this survey, a median soil amplification factor of 1.2 was estimated for the TMI-1 structures founded on overburden. For the diesel generator building, this effect of soil amplification was included in the evaluation of sliding failure mode capacity as shown in Section 4.1.1.8. For field-erected tanks, seismic forces were obtained from dynamic analysis of stick model representing the tank and contained fluid mass with a fixed-base at the top of the overburden. However, the seismic input was the median site-specific ground response spectrum anchored to 0.12g peak bedrock ground acceleration. To account for this amplification of the bedrock ground acceleration to the free ground surface by the overburden, the median soil amplification factor of safety used in the fragility evaluation of these tanks is:

$$F_{SA} = \frac{1}{1.2} = 0.83$$

The logarithmic standard deviation associated with uncertainty of the soil amplification factor of safety was found by estimating that there is a 95 percent confidence that the amplification factor is less than 1.5. The randomness was estimated to be about one-fourth of the uncertainty.

$$(\beta_U)_{SA} = \frac{1}{1.65} \ln\left(\frac{1.5}{1.2}\right)$$

$$= 0.13$$

$$(\beta_R)_{SA} = 0.03$$

4.1.7.2 Soil-Structure Interaction Method of Analysis

Flexibility of the soil and dissipation of energy into the soil by geometric damping will affect the frequency and response of the the structures founded on overburden. The seismic responses of the diesel generator building and the field-erected tanks used in the fragilities evaluation were determined using fixed-base models. For the diesel generator building, the governing failure mode results from impact with the intermediate building due to structure sliding. As shown in Section 4.1.1.8, the capacity for this failure mode is a function of permissible maximum sliding displacement, the available resistance, and the ratio of the peak ground velocity to the peak ground acceleration. As noted in Sections 4.1.1.8 and 4.1.7.1, the amplification of the bedrock ground motion by the soil was accounted for in the sliding fragility. The method of analysis used to represent soil-structure interaction would affect the acceleration at which the structure would be expected to begin to slide. However, soil-structure interaction is expected to have relatively little influence on the capacity against structural damage due to sliding. Thus, a method of analysis factor of unity is used for failure of the diesel generator building due to sliding. The shear walls and diaphragm of the diesel generator building have capacities much greater than that for sliding so any increase in response due to soil flexibility is not expected to change the controlling fragility values for this structure.

For the borated water storage tank, the soil impedances presented in Reference 43 were used to assess the effect of soil-structure interaction method of analysis on the seismic response. The fundamental frequency of the tank was found to shift from the 5.2 Hz predicted by the fixed-base model to about 3 Hz by considering the soil flexibility. Since both frequencies are in the amplified acceleration region of the median ground response spectrum, no change in the spectral acceleration would be expected. Although the response may increase due to a change in the predicted mode shape, the overall system damping would also be expected to increase due to contributions from soil material and geometric damping. These two effects tend to cancel each other. The median soil-structure interaction method of analysis factor was estimated to be unity for the borated water storage tank. Potential uncertainties include accuracy of the equivalent stiffness and geometric damping, strain degradation effects, soil properties, and the layering effect of underlying bedrock. Logarithmic standard deviations associated with randomness and uncertainty were estimated to be approximately 0.02 and 0.10, respectively.

One other possible area of concern is the slab uplift of the structures at high input acceleration levels. For structures founded on competent rock, there is insufficient energy in the low frequency earthquake waves to sustain overturning motion of the structure at the very long response periods required to overturn an auxiliary building or containment structure. At the frequencies of maximum input energy content, although a very small amount of uplift may occur, the direction of input motion is reversed before any significant rocking motion can occur. So long as significant rock or concrete crushing does not occur, relative motion sufficient to cause piping or electrical conduit failure is not considered a possible failure mode. The bedrock at the TMI-1 site is considered to be of adequate strength to preclude failures resulting from base slab uplift.

4.2 STRUCTURE FRAGILITIES

The significant failure modes for each of the essential TMI-1 structures included in this study were evaluated. The resulting fragilities for each of these structures are discussed in the following sections.

4.2.1 Containment and Internal Structures

The containment structure is a post-tensioned reinforced concrete structure consisting of a circular cylindrical wall capped by a shallow dome. The containment wall is supported by a base mat bearing on rock. Principal dimensions of the containment structure are:

Mat	Radius	77'-3/8"
	Thickness	9'-0"
	Liner plate thickness	1/4"
Cylinder	Inside radius	65'-0"
	Wall thickness	3'-6"
	Liner plate thickness	3/8"
	Height to springline	157'
Dome	Inside radius	110'
	Thickness	3'
	Liner plate thickness	3/8"

The controlling mode of failure for the containment structure was found to be shear failure of the wall near the base. Concrete with a design compressive strength of 5000 psi at 28 days was used to construct the wall. Reinforcement in the meridional and hoop directions was provided with additional reinforcement included at the discontinuities to resist increased stresses imposed by LOCA loading.

Horizontal shear forces due to seismic response of the containment structure primarily introduce tangential shear stresses in the wall. The results of scale model testing conducted to determine the strength of reinforced and prestresed containment structures subjected to seismic loads with and without internal pressure are summarized in Reference 44. The median shear strength of the containment wall was determined using empirical relationships derived from these test results. Resistance to horizontal seismic shear is provided by the concrete, the two-way reinforcement pattern, and the hoop and meridional prestressing tendons. This failure mode was found to have a median acceleration capacity of approximately 5.5g. Median factors of safety and variabilities for this failure mode are listed in Table 4-6. This mode of failure results in a loss-of-liner integrity and potential failure of the safety systems and components supported by the containment wall. Other potential seismic failure modes were evaluated and found to have higher seismic capacities than the wall shear failure.

The concrete internal structure consists of a primary shield wall enclosing the reactor cavity, the secondary shield wall supporting the steam generators, pressurizer, and reactor coolant pumps, and various floor slabs and other walls. The internal structure also provides biological shielding and missile protection. Both the primary and secondary shield walls are founded on the base mat common with the containment structure. Dimensions of the concrete internal structure are:

Primary Shield Wall	Inner radius	11'-6"
	Thickness	5'-0" increased to 10'-6" at the bottom 10 feet
	Height	41'-6"
Secondary Shield Wall	Thickness	4'-0"
	Height	86'-0"

Review of the internal structure indicated that failure due to seismic response is expected to occur towards the base of the secondary shield or at the 5-foot thick portion of the primary shield wall. Capacity of the internal structure was found to be controlled by seismic loads acting primarily in the N-S direction. Structural failure of the secondary shield wall is expected to result from the overall overturning moment. The median acceleration capacity was found to be approximately 2.4g. Median factors of safety and variabilities for this failure mode are listed in Table 4-7. The controlling mode of failure for the primary shield wall was found to be shear failure near the base of the 5-foot wall. The median acceleration capacity of this failure mode was found to be approximately 2.6g. Median factors of safety and variabilities for this failure mode are listed in Table 4-8. Structural failure of either the primary shield wall or the secondary shield wall is expected to result in failure of the reactor coolant pressure boundary.

4.2.2 Control Building and Auxiliary Building

The control building, fuel handling building, and auxiliary building of TMI-1 are structurally connected by various walls and floor slabs such that it is essentially an integral structure. All three buildings are located to the north of the reactor building and are founded on sound bedrock. The primary lateral force resisting systems of all three structures are the reinforced concrete shear walls. Seismic forces were obtained from the dynamic analysis of a single dynamic model representing all three structures with seismic input consisting of the ten percent damped median site-specific ground response spectrum anchored to 0.12g peak ground acceleration.

The control building is located to the east of the fuel handling building and is constructed of reinforced concrete floor slabs poured on steel decking and shear walls, with structural steel framing provided for additional vertical load support. The structure is founded on continuous wall footings and column spread footings at EL. 278 ft. It spans six stories up to the 5-foot thick roof slab at EL. 400 ft. The control building is structurally tied to the fuel handling building by the roof slab and the concrete floor slab at EL. 306 ft. The E-W direction 5-foot thick exterior shear walls of the control building are also tied to the N-S direction 5-foot thick west wall of the fuel handling building. Nuclear instrumentation and reactor protection panels are contained in the control building. The control room is located on the floor at EL. 355 ft.

The fuel handling building is a rectangular box-type reinforced concrete structure with partial floor slabs at various elevations. It is located between the control building and the auxiliary building and is structurally tied together by the roof slabs of these two structures at EL. 400 ft and 329 ft, respectively. All three buildings are tied together by the concrete slab at EL. 306 ft. The fuel handling building houses the spent fuel pool, and is not important for safe shutdown except to the extent it influences the remaining structure. The auxiliary building is a two-story reinforced concrete structure housing equipment related to the chemical and volume control, component cooling water, and reactor protection system. It is founded on a base mat which bears on bedrock at EL. 258 ft and 278 ft. The roof slab is at EL 331 ft. Numerous exterior and interior reinforced concrete shear walls oriented in both the N-S and E-W directions are present in the auxiliary building.

The concrete shear walls in the control building consist of 5-foot thick exterior walls at the north, south and east sides, a 2'-6" interior wall in the N-S direction, and a 3-foot interior wall in the E-W direction. Except at the roof level and the floor slab at El. 306', two-inch gaps filled with compressible material separate each major floor slab from the 5-foot exterior shear walls (see Figure 4-7) such that floor inertia forces are to be resisted by the 2'-6" and 3-foot thick interior shear walls. Review of the structural responses, wall dimensions, and the available resistances against seismic loads indicated that the 3-foot interior wall oriented in the E-W direction will govern the capacity of the control building.

The original design dynamic model of the control building, the fuel handling building and the auxiliary building was a single stick model with tributary masses lumped at the major floor elevations. Overall story stiffnesses were modeled by vertical beam elements between the mass points. This relatively simple dynamic model was judged to be adequate for the prediction of overall seismic responses of these three buildings. However, for the evaluation of the E-W direction 3-foot interior wall on Column Line 11 of the control building, a more refined load distribution model was employed to obtain more realistic seismic loads in this wall. This load distribution model, shown in Figure 4-8, reflects the fact that all control building and fuel handling building shear walls are tied together by the 5-foot thick concrete roof slab at EL. 400 ft and that the concrete shear walls of all three buildings are tied together by the floor slab at El. 306 ft. To account for the elastic interaction between the 3-foot interior wall of the control building and the other shear walls of all three buildings, the load distribution model consisted of two separate vertical sticks which were connected rigidly at the roof level of the control building and at EL. 306 ft.

One vertical stick modeled the lateral stiffness of the 3-foot control building wall on Column Line 11 and was subjected to its tributary seismic inertia forces. The second vertical stick modeled the lateral stiffness of the rest of the structure and its corresponding seismic inertia forces. The seismic inertia forces acting on the first stick were estimated by factoring the tributary floor and wall masses by the SRSS E-W acceleration at each floor. The inertia forces acting on the second stick were then estimated by factoring the total lumped masses at each floor, reduced by the tributary masses included in the first stick, by the SRSS E-W floor accelerations. The overturning moment in the 3-foot wall at EL. 306' was obtained from the static analysis performed on this load distribution model. This overturning moment was transferred to the story below. Additional loads acting on the lower story (from EL. 282' to EL. 306') wall were found from a load distribution model representing the connectivity of the entire structure.

The controlling failure mode of the control building was found to be failure of this wall at the bottom story (EL. 282' to EL. 306') due to in-plane overturning moment. The capacity of this wall was determined as described in Section 4.1.1.6. The median acceleration capacity for this failure mode was estimated to be approximately 1.0g. Median factors of safety and variabilities are listed in Table 4-9. Failure of this wall is expected to lead to damage to the critical equipment located in the control building. It must be noted that when the 3-foot thick interior wall yields, loads will be redistributed to the other shear walls which have substantially more capacity than this wall. This load redistribution was not accounted for in this study and the estimated median acceleration capacity of 1.0g is therefore considered to be conservative. If this failure mode is found to be dominant, an evaluation of the load redistribution would be warranted. However, this evaluation is beyond the scope of this study.

Inspection of available drawings indicated that the control room ceiling was safety-wired and the light fixtures above the control room were braced. Failures of either of these systems are expected at acceleration levels in excess of the controlling failure modes. Diaphragm capacities were evaluated and found not to be controlling.

The controlling failure mode of the auxiliary building is also expected to be shear wall failure. Based upon an elastic load distribution of the overall median structural loads, yielding due to in-plane overturning moment is expected to initiate at the west wall between the auxiliary building and the heat exchanger vault between EL. 281' and EL. 305'. The median acceleration capacity was found to be approximately 1.7g. Median factors of safety and variabilities for this failure mode are listed in Table 4-10. Shear wall failure is expected to lead to damage of the critical equipment located in the auxiliary building. Other potential failure modes investigated include the diaphragm failure of the roof slab. This failure mode was found not to be controlling.

4.2.3 Intake Screen House

The intake screen house is a reinforced concrete box-type structure housing the safety-related river water pumps. The structure is founded on a base mat bearing on sound bedrock. The main lateral force resisting system consists of concrete slabs at the roof and the operating floor at EL. 308' and the 5-foot thick exterior shear walls. Two 15-ft by 10-ft openings for drawing river water are located near the base of the west wall facing the Susquehanna River. Twelve-foot thick transverse guide walls are present in the structure to channel the water flow.

Capacity of the intake screen house is expected to be governed by flexural failure at the base of the west wall due to in-plane overturning moment. The capacity of this wall against overturning moment was found by using the approach described in Section 4.1.1.5. Any additional capacity provided by load redistribution and flanges formed by the intersecting walls was conservatively neglected. The median acceleration capacity was estimated to be approximately 1.4g. Median factors of safety and variabilities are listed in Table 4-11. Failure of this wall is expected to lead to damage of the water pumps and other safety-related equipment. Other intake screen house potential failure modes investigated included the diaphragm failure of the floor slab at EL. 308'. The median acceleration capacity was estimated to be approximately 2.9g for this failure mode.

4.2.4 Intermediate Building

The intermediate building is located between the diesel generator building and the reactor building and separated from these structures by gaps of 1.5" and 3", respectively. The structure is founded on continuous wall footings which bear on the bedrock at EL. 276'. The intermediate building houses the emergency feed pumps and instrument air supply system. The primary lateral force resisting system is composed of reinforced concrete slabs at various floors and 3-foot and 5-foot thick interior and exterior reinforced concrete shear walls.

The failure mode having the lowest acceleration capacity is flexural failure of the E-W direction wall adjacent to the reactor building between EL. 295' and EL. 322' due to in-plane overturning moment. The capacity of this wall against overturning moment was found using the approach described in Section 4.1.1.5. Any additional capacity provided by load redistribution and flanges formed by the intersection walls was conservatively neglected. The median acceleration capacity of this failure mode was found to be approximately 1.3g. Median factors of safety and variabilities for this failure mode are listed in Table 4-12. Shear wall failure is expected to lead to damage to the critical equipment located in this structure.

Diaphragm failure was also investigated. The concrete floor slabs serve as diaphragms transmitting inertia forces to the walls and redistributing shear wall loads due to changes in relative wall stiffnesses from story-to-story. The slab at EL. 322' is perforated by a series of openings and the stiffnesses of the walls above and below the slab change significantly. Failure of this portion of the slab is expected at a median acceleration capacity of approximately 1.8g.

4.2.5 Diesel Generator Building

The diesel generator building is a one-story, box-type reinforced concrete structure supported on a base mat bearing on a 30-foot overburden overlying the bedrock. The diesel generator building contains the emergency diesel generators and related equipment. The main lateral load resisting system consists of interior and exterior reinforced concrete shear walls.

The controlling failure mode of this structure was found to be impact between the intermediate building and the diesel generator building due to sliding of the latter structure. Sliding initiates when the seismic base shear overcomes the shearing resistance of the structural backfill retained within the shear keys. However, as noted in Section 4.1.1.7, the initiation of structure sliding does not necessarily imply failure. Sliding-induced failure does not occur until sufficient displacement is developed to damage safety-related equipment. The diesel generator building does not contact the adjacent intermediate building until the 1.5 inch separation gap is closed. Failure of the diesel generator building is expected to correspond to a sliding displacement towards the intermediate building of approximately 2.5 inches. For this sliding displacement, the ability of the south exterior wall of the diesel generator building to support safety-related equipment and resist seismic loads may be lost.

The median acceleration capacity against sliding-induced failure of the diesel generator building in the south direction was calculated as shown in Section 4.1.1.8. The median bedrock acceleration capacity for sliding-induced failure was found to be approximately 1.3g. Median factors of safety and variabilities for this failure mode are listed in Table 4-13. Sliding is not expected to initiate until approximately 0.66g which may be considered a lower bound cut-off for this mode of failure.

Safety-related piping and ducting pass between openings in the diesel generator building south wall and the intermediate building north wall. This piping and ducting may fail due to sliding displacement of the diesel generator building in the E-W direction. Resistance to sliding in this direction is essentially the same as that in the N-S direction. A median sliding displacement of four inches is expected to cause failure of the safety-related piping and ducting based on information available on the wall opening size and the piping layout. Because the permissible sliding displacement is greater in the E-W direction than in the south direction, sliding in the E-W direction does not govern. Other potential diesel generator building failure modes evaluated included shear wall failure and diaphragm failure. These failure modes were found to have median acceleration capacities greater than 3g.

4.2.6 Borated Water Storage Tank

The borated water storage tank (BWST) is fabricated from SA 240-304 stainless steel. It has a radius of 16'-6" and is 52'-0" at the top of the side wall with plate thicknesses varying from 0.25 inches to 0.421 inches. A total of 39 two-inch diameter high strength (A540 Grade 21 Class 2) anchor bolts are provided around the tank perimeter at the base mat. The base mat is located on top of a 30-foot deep overburden as discussed in Section 4.1.7.1.

A fixed-base, lumped mass dynamic model was used to determine the seismic response at the BWST. The impulsive fluid masses were determined using the approach described in Reference 45. The tank shell stiffness was modeled by beam elements between mass points distributed up the tank shell. Impulsive fluid effective weights were added to the tank shell weights at each of the mass node points at and below the top surface of the fluid. Seismic input consisted of the seven percent damped median site-specific ground response spectrum anchored to 0.12g bedrock peak ground acceleration. The horizontal fluid sloshing mode was accounted for by a separate analysis.

Capacity of the BWST was found to be governed by buckling of the lowest shell course due to the overall structure overturning moment. The buckling stress was evaluated in accordance with the criteria in Reference 46. To account for the amplification of the ground motion by the overburden, a median soil amplification factor of safety of 0.83 was included. Derivation of this factor is described in Section 4.1.7.1. The median bedrock acceleration capacity for the BWST was found to be approximately 0.62g. Median factors of safety and variabilities are listed in Table 4-14. Buckling of the tank wall is assumed to lead to a loss of contents due to the potential for cracking at a weld.

4.2.7 Condensate Storage Tank

The condensate storage tank (CST) is fabricated from A-283C carbon steel. It has a radius of 24'-0" and stands 20'-0" to the top of the side wall with a plate thickness of 0.25 inches. A total of sixteen 1-1/2" diameter A36 anchor bolts are provided around the tank perimeter at the base mat. The base mat is also located on top of the 30-foot deep overburden. The CST was evaluated in the same manner as the BWST as described in Section 4.2.6. The failure mode of this tank consists of the anchor bolts yielding in tension due to the overturning moment followed by compressive buckling of the tank wall. This is assumed to result in failure of plate welds and loss of tank contents. The median bedrock acceleration capacity of this tank is approximately 2.0g.

TABLE 4-1

RESULTS OF CONCRETE COMPRESSIVE STRENGTH TESTING (From Reference 16)

Structure	Specified Design Strength (psi)	Age at Testing (days)	Average Test Strength (psi)	Test Strength Standard Deviation (psi)
Containment Wall	5000	28	6100	580
All Other Category I Structures	5000	28	5900	610
	3000	28	5000	790

TABLE 4-2

MEDIAN CONCRETE COMPRESSIVE STRENGTH AND VARIABILITIES

Structure	Specified Design Strength (psi)	Median Strength, $\bar{f}'_c$ (psi)	Logarithmic Standard Deviation, β
Containment Wall	5000	7300	0.13
All Other Category I Structures	5000	7100	0.14
	3000	5900	0.19

TABLE 4-3

SCALE FACTORS NEEDED TO ACHIEVE $\mu = 1.85$

a) Due to 6.5 - 7.5 Richter magnitude earthquakes

Earthquake Record (Comp)	Model Structure Frequency			
	8.54 Hz	5.34 Hz	3.20 Hz	2.14 Hz
Olympia, WA., 1949 (N86E)	1.36	1.11	1.49	1.70
Taft, Kern Co., 1952 (S69E)	1.20	1.25	1.50	1.78
El Centro Array No. 12 Imperial Valley, 1979, (140)	1.34	1.56	1.29	1.48
Pacoima Dam San Fernando, 1971 (S14W)	1.25	1.38	1.26	2.19
Hollywood Storage PE Lot, San Fernando, 1971 (N90E)	1.45	1.65	1.58	1.39
El Centro Array No. 5 Imperial Valley, 1979, (140)	1.58	1.60	1.34	1.51

Mean = 1.47 Median = 1.47 Range = 1.11 - 2.19

b) Due to 4.5 - 6.0 Richter magnitude earthquakes

Earthquake Record (Comp)	Model Structure Frequency			
	8.54 Hz	5.34 Hz	3.20 Hz	2.14 Hz
UCSB Goleta Santa Barbara, 1978 (180)	1.35	1.65	1.41	1.49
Gilroy Array No. 2, Coyote Lake, 1979, (050)	1.36	1.93	2.00	1.86
Gavilan College Hollister, 1974 (S67W)	1.61	1.55	1.62	1.93
Melendy Ranch Barn, Bear Valley, 1972 (N29W)	1.45	1.96	2.18	1.98

Mean = 1.71 Median = 1.64 Range = 1.35 - 2.18

TABLE 4-4

SCALE FACTORS NEEDED TO ACHIEVE μ = 4.27

a) Due to 6.5 - 7.5 Richter magnitude earthquakes

Earthquake Record (Comp)	Model Structure Frequency			
	8.54 Hz	5.34 Hz	3.20 Hz	2.14 Hz
Olympia, WA., 1949 (N86E)	1.56	1.54	2.61	3.75
Taft, Kern Co., 1952 (S69E)	1.25	1.65	2.05	3.38
El Centro Array No. 12 Imperial Valley, 1979, (140)	1.56	2.29	2.10	2.14
Pacoima Dam San Fernando, 1971 (S14W)	1.70	1.86	2.67	3.89
Hollywood Storage PE Lot, San Fernando, 1971 (N90E)	1.94	2.50	2.60	2.05
El Centro Array No. 5 Imperial Valley, 1979, (140)	2.38	2.66	2.33	3.45

Mean = 2.33 Median = 2.22 Range = 1.25 - 3.89

b) Due to 4.5 - 6.0 Richter magnitude earthquakes

Earthquake Record (Comp)	Model Structure Frequency			
	8.54 Hz	5.34 Hz	3.20 Hz	2.14 Hz
UCSB Goleta Santa Barbara, 1978 (180)	1.52	2.05	2.05	1.96
Gilroy Array No. 2, Coyote Lake, 1979, (050)	1.56	3.85	4.36	3.03
Gavilan College Hollister, 1974 (S67W)	2.84	2.97	2.71	8.49
Melendy Ranch Barn, Bear Valley, 1972 (N29W)	1.89	5.48	5.16	3.36

Mean = 3.33 Median = 2.91 Range = 1.52 - 8.49

TABLE 4-5

COMPARISON OF RECENT STUDIES (REFERENCE 26)
WITH AMENDED RIDDELL-NEWMARK PROCEDURE

		Reference 26		Amended Riddell-Newmark	
Magnitude Range	μ	Median	Range	Median	Range
$6.5 < M < 7.5$	4.27	2.22	1.25 - 3.89	2.24	1.28 - 3.92
$4.5 < M < 6.0$	4.27	2.91	1.52 - 8.49	2.89	1.41 - 5.94
$6.5 < M < 7.5$	1.85	1.47	1.11 - 2.19	1.49	1.14 - 1.92
$4.5 < M < 6.0$	1.85	1.64	1.35 - 2.18	1.84	1.21 - 2.69

TABLE 4-6

Structure: Reactor Building

Failure Mode: Shear Failure of Containment Wall

Factor	Median F.S.	β_R	β_U	β_C
Strength	11	0	0.22	0.22
Inelastic Energy Absorption	2.0	0.24	0.18	0.30
Spectral Shape	2.1	0.18	0.12	0.22
Damping	1.0	0.10	0.10	0.14
Modeling	1.0	0	0.15	0.15
Modal Combination	1.0	0.03	0	0.03
Combination of EQ Components	1.0	0.01	0	0.01
Soil-Structure Interaction				
Soil Amplification	1.0	0	0	0
Method of Analysis	1.0	0	0.05	0.05
Total	46	0.32	0.36	0.48

Median Acceleration Capacity = 46(0.12g)
= 5.5g

TABLE 4-7

Structure: Concrete Internal Structure

Failure Mode: Failure of Secondary Shield Wall

Factor	Median F.S.	β_R	β_U	β_C
Strength	12	0	0.23	0.23
Inelastic Energy Absorption	1.7	0.18	0.14	0.23
Spectral Shape	1.0	0.15	0.10	0.18
Damping	1.0	0.08	0.08	0.11
Modeling	1.0	0	0.18	0.18
Modal Combination	1.0	0.01	0	0.01
Combination of EQ Components	1.0	0.01	0	0.01
Soil-Structure Interaction				
Soil Amplification	1.0	0	0	0
Method of Analysis	1.0	0	0.05	0.05
Total	20	0.25	0.35	0.43

Median Acceleration Capacity = 20(0.12 g)
= 2.4g

TABLE 4-8

Structure: Concrete Internal Structure

Failure Mode: Shear Failure of the Primary Shield Wall

Factor	Median F.S.	β_R	β_U	β_C
Strength	13	0	0.25	0.25
Inelastic Energy Absorption	1.7	0.18	0.14	0.23
Spectral Shape	1.0	0.15	0.10	0.18
Damping	1.0	0.08	0.08	0.11
Modeling	1.0	0	0.18	0.18
Modal Combination	1.0	0.03	0	0.03
Combination of EQ Components	1.0	0.01	0	0.01
Soil-Structure Interaction				
Soil Amplification	1.0	0	0	0
Method of Analysis	1.0	0	0.05	.05
Total	22	0.25	0.37	0.44

Median Acceleration Capacity = 22(0.12 g)
= 2.6g

TABLE 4-9

Structure: Control Building

Failure Mode: Shear Wall Failure

Factor	Median F.S.	β_R	β_U	β_C
Strength	4.9	0	0.24	0.24
Inelastic Energy Absorption	1.7	0.18	0.14	0.23
Spectral Shape	1.0	0.16	0.10	0.19
Damping	1.0	0.08	0.08	0.11
Modeling	1.0	0	0.19	0.19
Modal Combination	1.0	0.07	0	0.07
Combination of EQ Components	1.0	0.04	0	0.04
Soil-Structure Interaction				
Soil Amplification	1.0	0	0	0
Method of Analysis	1.0	0	0.05	0.05
Total	8.3	0.27	0.36	0.45

Median Acceleration Capacity = 8.3 (0.12 g)
= 1.0g

NOTE: Load redistribution after yielding of this wall was not accounted for in this study such that the reported median acceleration capacity is conservative. See discussion in Section 4.2.2

TABLE 4-10

Structure: Auxiliary Building

Failure Mode: Shear Wall Failure

Factor	Median F.S.	β_R	β_U	β_C
Strength	8.6	0	0.23	0.23
Inelastic Energy Absorption	1.6	0.16	0.12	0.20
Spectral Shape	1.0	0.15	0.10	0.18
Damping	1.0	0.09	0.09	0.13
Modeling	1.0	0	0.19	0.19
Modal Combination	1.0	0.04	0	0.04
Combination of EQ Components	1.0	0.04	0	0.04
Soil-Structure Interaction				
Soil Amplification	1.0	0	0	0
Method of Analysis	1.0	0	0.05	0.05
Total	14	0.24	0.35	0.43

Median Acceleration Capacity = 14 (0.12g)
= 1.7g

TABLE 4-11

Structure: Intake Screen House

Failure Mode: Shear Wall Failure

Factor	Median F.S.	β_R	β_U	β_C
Strength	9.2	0	0.22	0.22
Inelastic Energy Absorption	1.3	0.09	0.07	0.11
Spectral Shape	1.0	0.05	0.03	0.06
Damping	1.0	0.02	0.02	0.03
Modeling	1.0	0	0.16	0.16
Modal Combination	1.0	0.05	0	0.05
Combination of EQ Components	1.0	0.03	0	0.03
Soil-Structure Interaction				
Soil Amplification	1.0	0	0	0
Method of Analysis	1.0	0	0.05	0.05
Total	12	0.12	0.29	0.31

Median Acceleration Capacity = 12 (0.12g)
= 1.4g

TABLE 4-12

Structure: Intermediate Building

Failure Mode: Shear Wall Failure

Factor	Median F.S.	β_R	β_U	β_C
Strength	7.1	0	0.22	0.22
Inelastic Energy Absorption	1.6	0.16	0.12	0.20
Spectral Shape	1.0	0.11	0.08	0.14
Damping	1.0	0.06	0.06	0.08
Modeling	1.0	0	0.18	0.18
Modal Combination	1.0	0.01	0	0.01
Combination of EQ Components	1.0	0.06	0	0.06
Soil-Structure Interaction				
Soil Amplification	1.0	0	0	0
Method of Analysis	1.0	0	0.05	0.05
Total	11	0.21	0.33	0.39

Median Acceleration Capacity = 11 (0.12 g)
= 1.3g

TABLE 4-13

Structure: Diesel Generator Building

Failure Mode: Structure Impact Due to Sliding (Southward)

Factor	Median F.S.	β_R	β_U	β_C
Strength	11	0	0.41	0.41
Inelastic Energy Absorption	1.0	0	0	0
Spectral Shape	1.0	0.19	0.06	0.20
Damping	1.0	0	0	0
Modeling	1.0	0	0	0
Modal Combination	1.0	0	0	0
Combination of EQ Components	1.0	0.13	0	0.13
Soil-Structure Interaction				
Soil Amplification	1.0	0.01	0.05	0.05
Method of Analysis	1.0	0	0	0
Total	11	0.23	0.42	0.48

Median Acceleration Capacity = 11 (0.12g)
= 1.3g (0.66 lower bound cut-off)

TABLE 4-14

Structure: Borated Water Storage Tank

Failure Mode: Buckling of the Tank Wall

Factor	Median F.S.	β_R	β_U	β_C
Strength	6.3	0	0.33	0.33
Inelastic Energy Absorption	1.0	0	0	0
Spectral Shape	1.0	0.22	0.15	0.27
Damping	1.0	0.08	0.08	0.11
Modeling	1.0	0	0.15	0.15
Modal Combination	1.0	0.02	0	0.02
Combination of EQ Components	1.0	0.02	0	0.02
Soil-Structure Interaction				
Soil Amplification	0.83	0.03	0.13	0.13
Method of Analysis	1.0	0.02	0.10	0.10
Total	5.2	0.24	0.43	0.49

Median Acceleration Capacity = 5.2(0.12 g)
= 0.62g

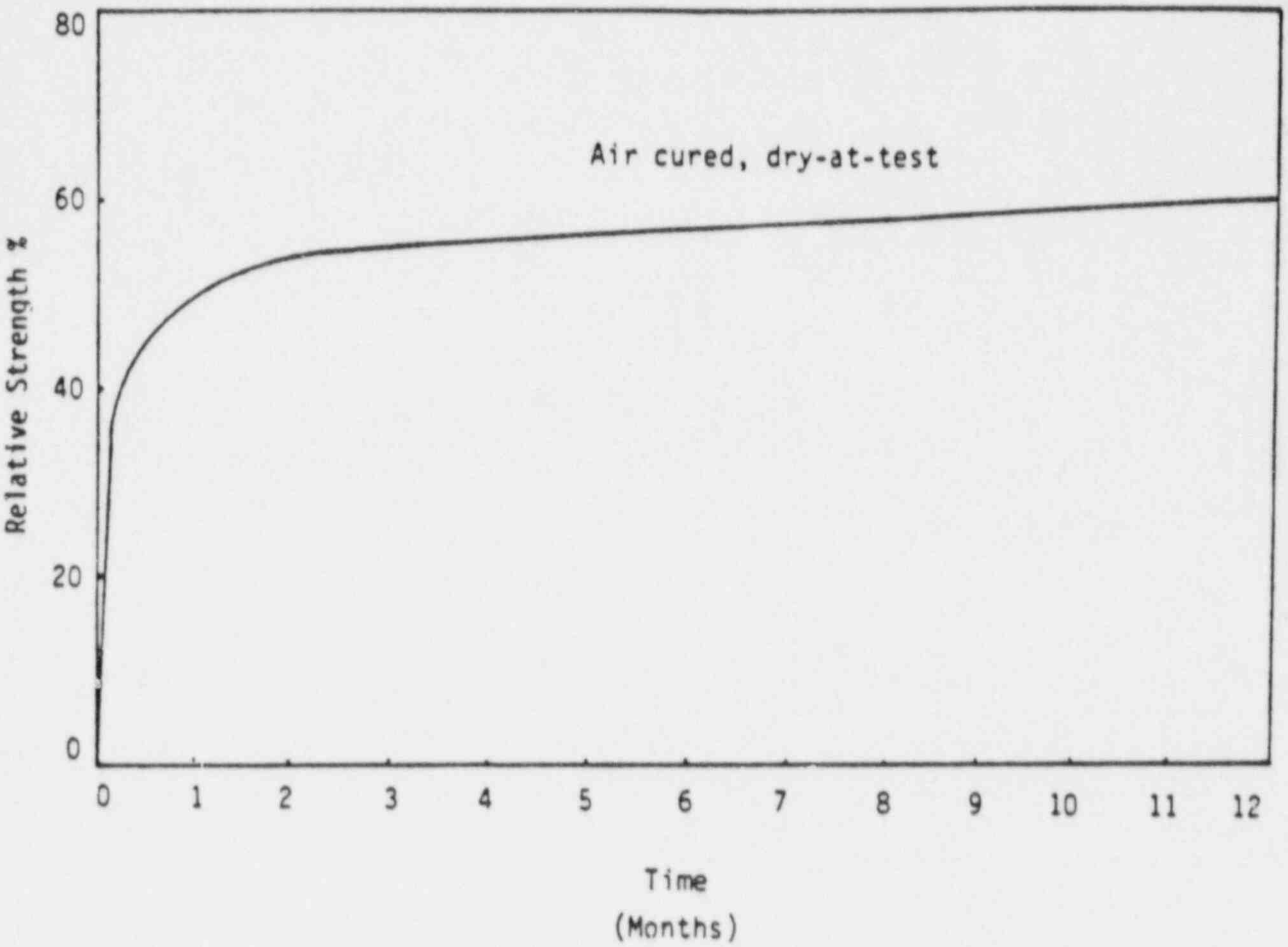


FIGURE 4-1. EFFECTS OF TIME AND CURING CONDITIONS ON CONCRETE STRENGTH (FROM REFERENCE 17)

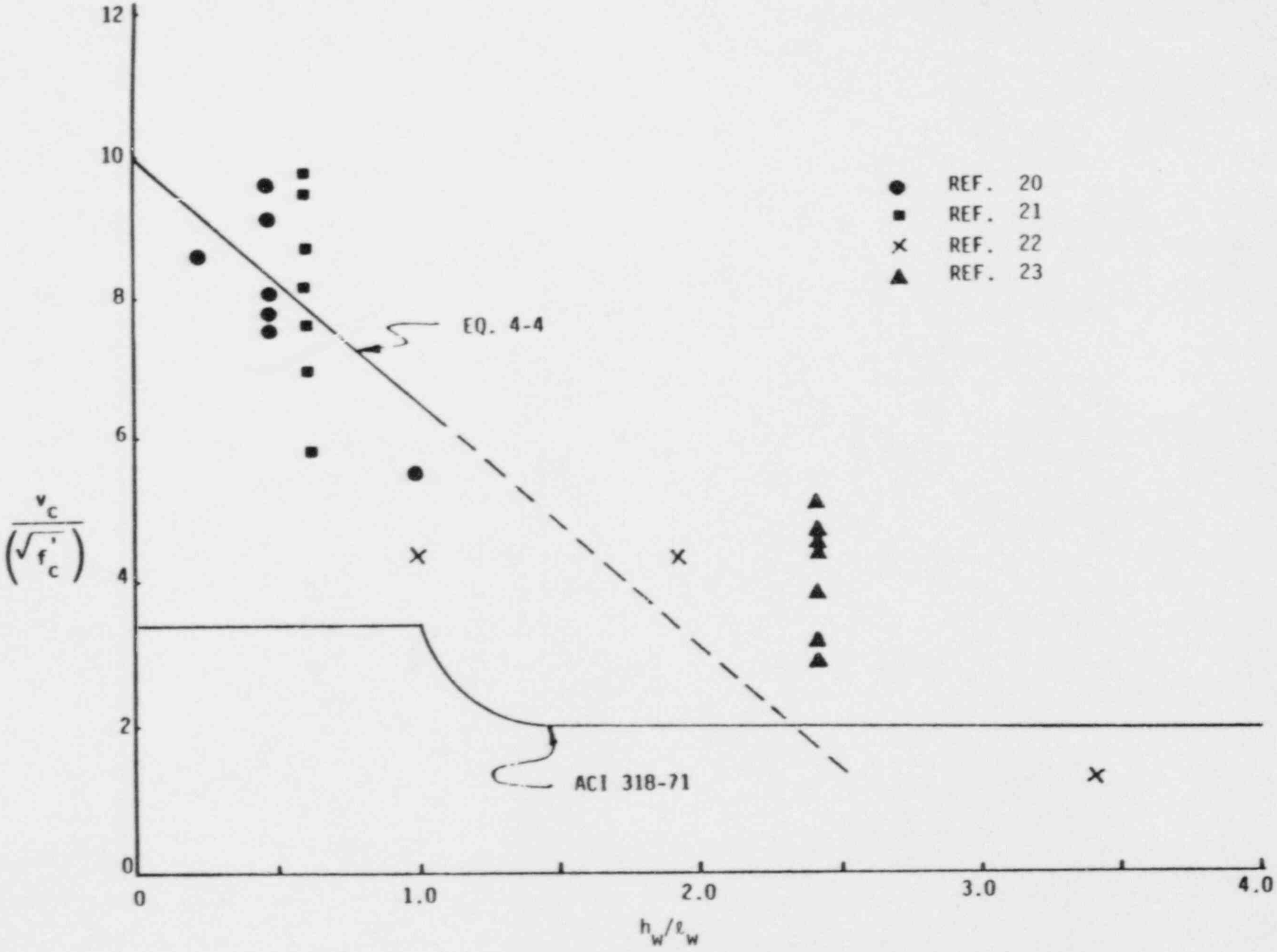
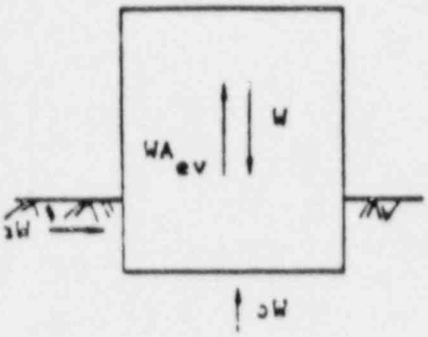


FIGURE 4-2. STRENGTH OF CONCRETE SHEAR WALLS



Effective vertical equivalent acceleration = $A_{ev}\ g$

Hydrostatic uplift = ρW

Net vertical contact force on base = $W\ (1-\rho-A_{ev})$

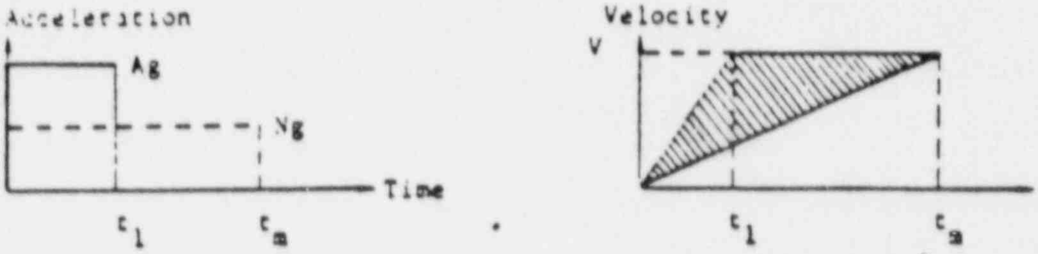
Static horizontal force = αW

Net horizontal resistance = $NW = NMg$

Coefficient of friction = μ

Then $N = \mu(1-\rho-A_{ev}) - \alpha$

Consider a single pulse of horizontal acceleration Ag lasting for a time t_1, giving a velocity V



$$t_1 = \frac{V}{Ag},\ t_m = \frac{V}{Ng}$$

$$u_m = (t_m - t_1)\,\frac{V}{2} = \frac{V^2}{2g}\left(\frac{1}{N} - \frac{1}{A}\right)$$

$$= \frac{V^2}{2gN}\left(1 - \frac{N}{A}\right) \qquad (4\text{-}11)$$

FIGURE 4-3. NEWMARK SLIDING APPROACH (FROM REFERENCE 24)

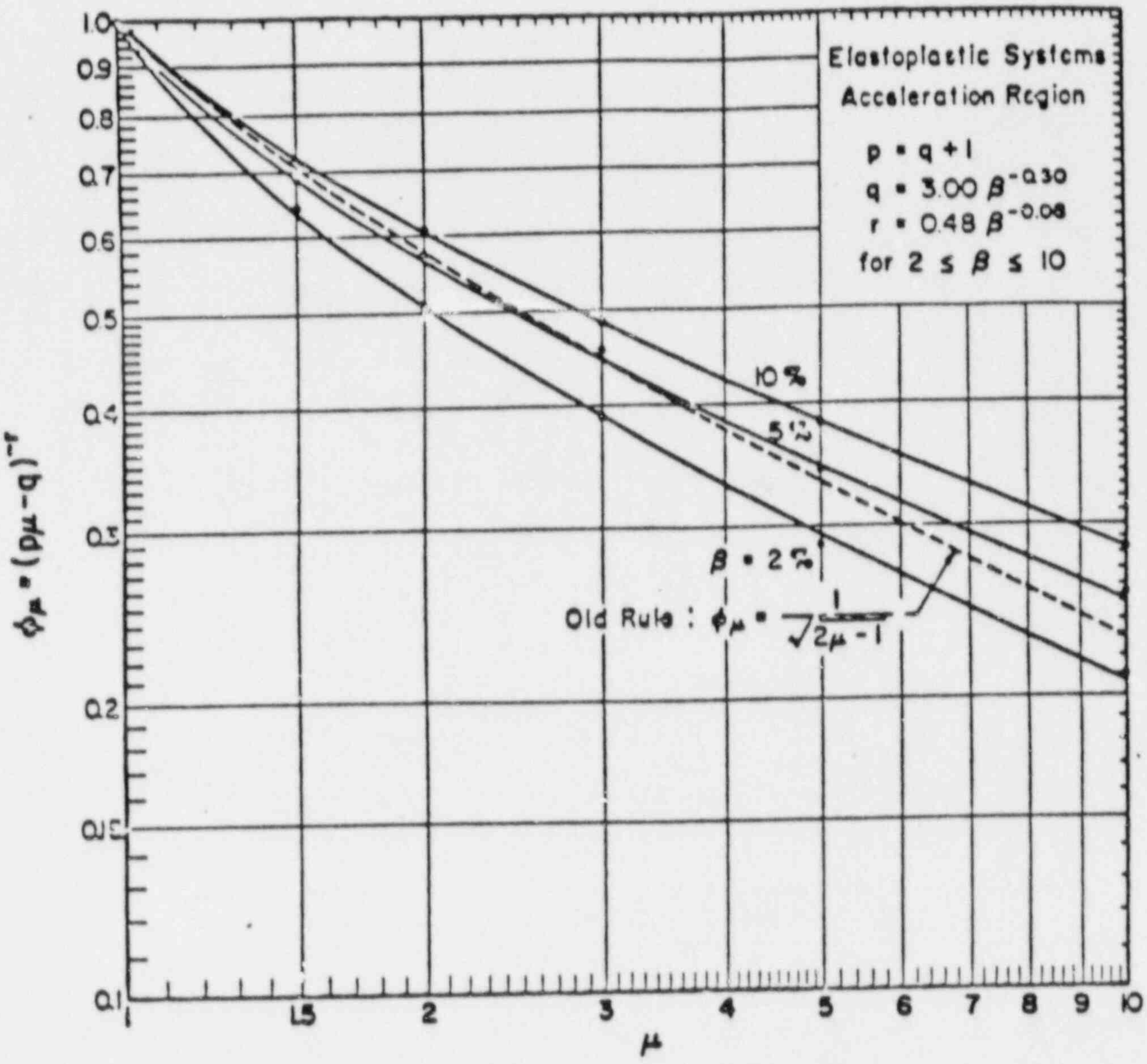


FIGURE 4-4. DEAMPLIFICATION FACTORS FOR ELASTIC-PERFECTLY PLASTIC SYSTEMS IN THE ACCELERATION AMPLIFIED RANGE (FROM REFERENCE 7)

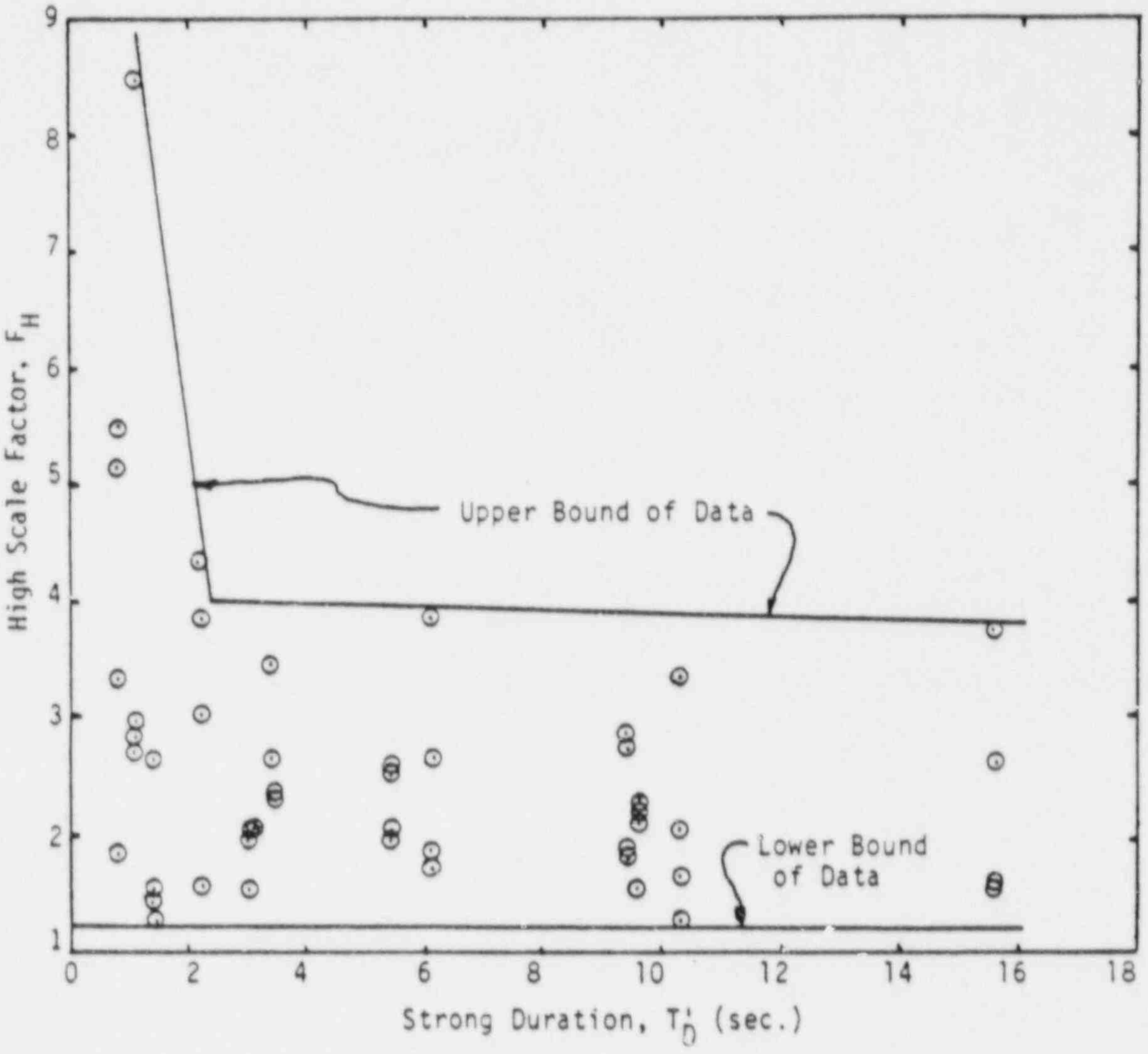


FIGURE 4-5. SCALE FACTOR, F_H VERSUS DURATION, T_D'
(From Reference 26)

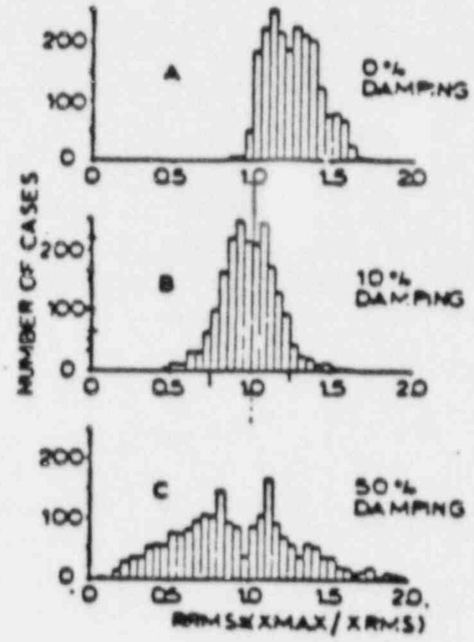


FIGURE 4-6. HISTOGRAMS OF RATIO OF PEAK RESPONSE TO SRSS COMPUTED RESPONSE FOR FOUR-DEGREE-OF-FREEDOM DYNAMIC MODELS (FROM REFERENCE 27)

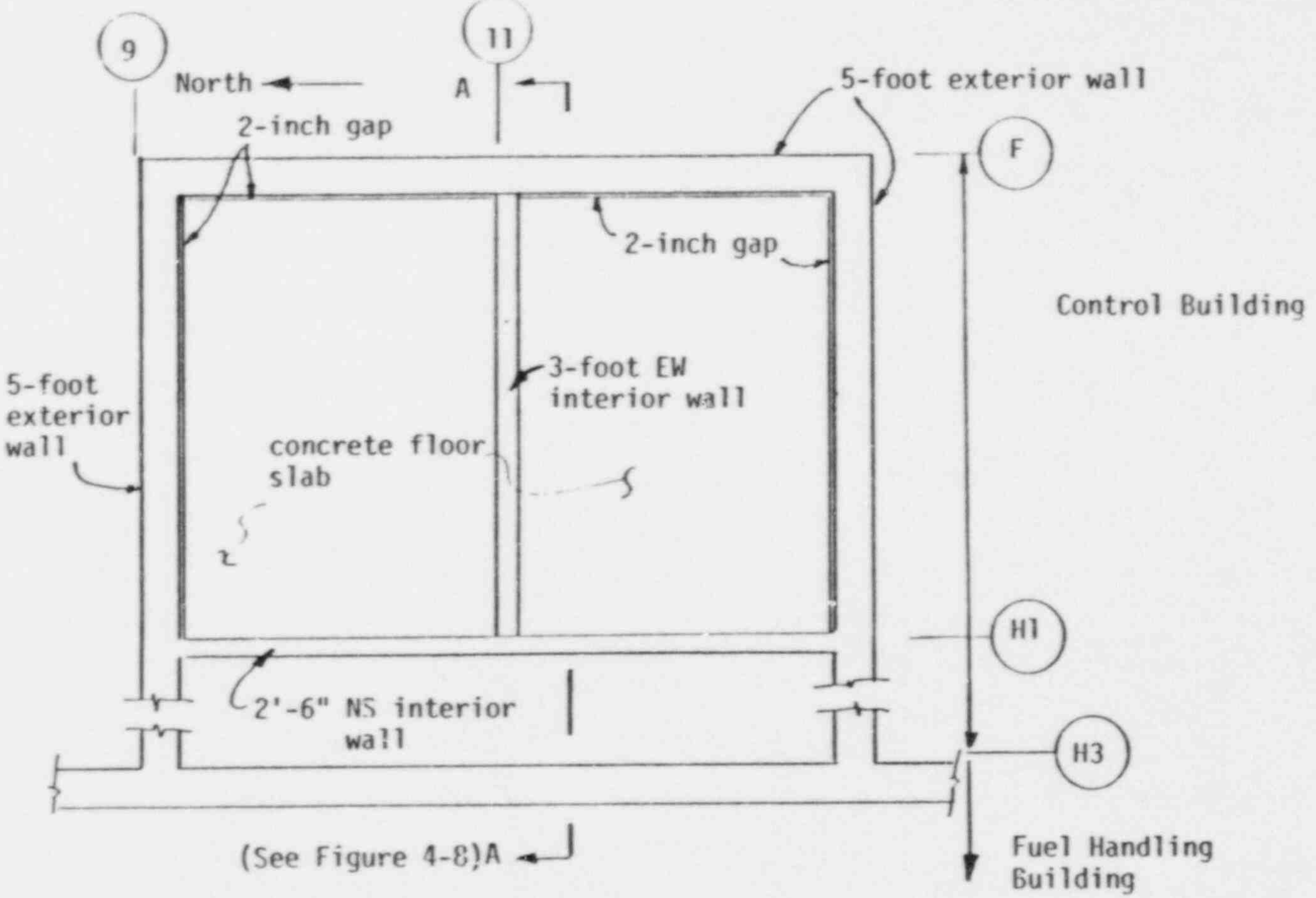
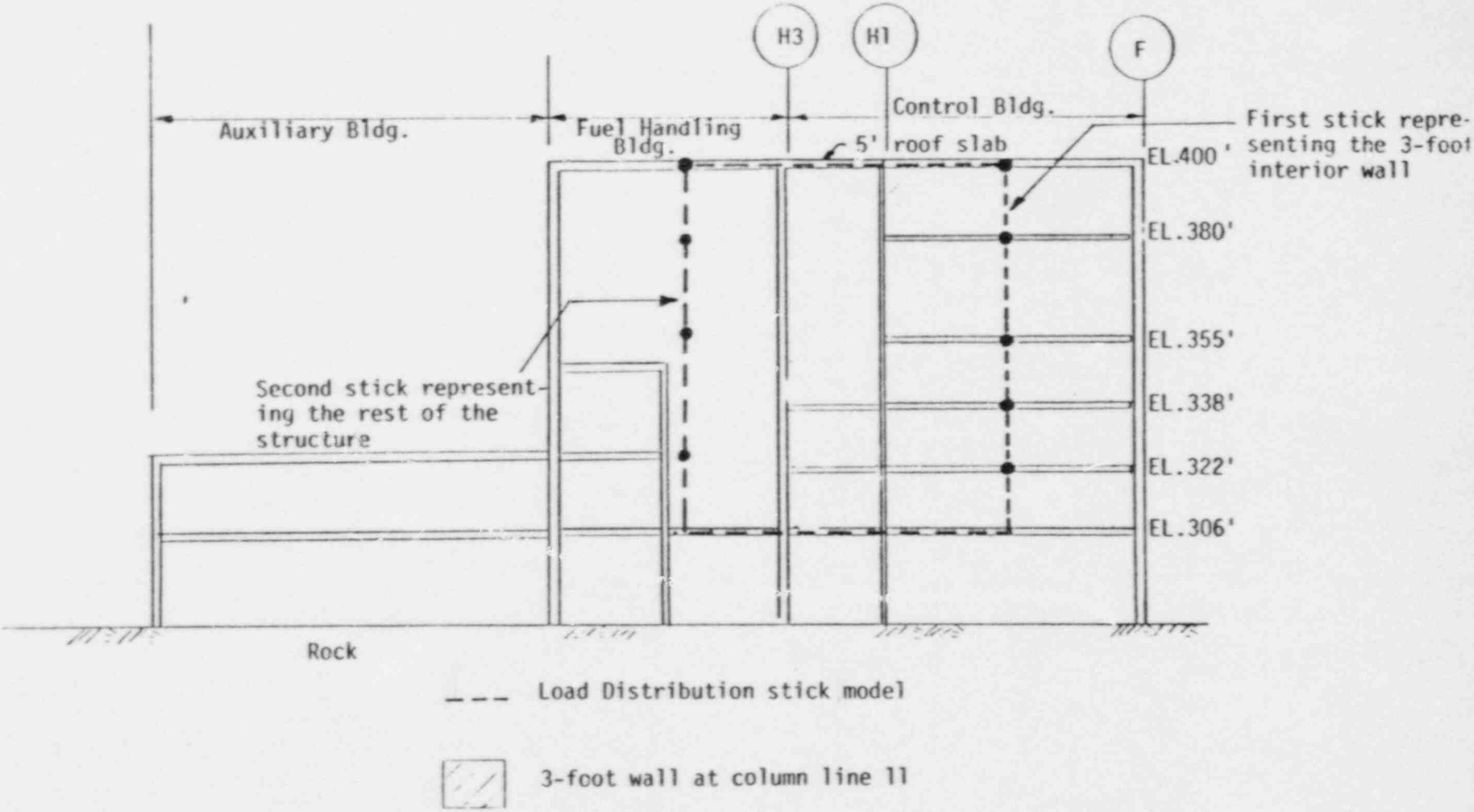


FIGURE 4-7 CONTROL BUILDING TYPICAL FLOOR PLAN



Section A-A

FIGURE 4-8 CONTROL BUILDING LOAD DISTRIBUTION MODEL

5. EQUIPMENT FRAGILITY

This chapter describes the fragility development for the seismically critical equipment within the TMI-1 Nuclear Power Plant. PL&G has identified those equipment items which are essential to plant safety during and after a seismic event. Based on a review by PL&G of the seismic hazard curves for the TMI site, frequencies of earthquakes greater than about 1.0g are so low that they could not result in system failure frequencies that approach what can be expected for other initiating events. Consequently, components exhibiting a median ground acceleration capacity (Å) of 1.0g or greater will have a negligible impact upon risk associated with the TMI-1 plant operation. Therefore, plant-specific fragilities were not derived for those components which could be shown to possess an Å greater than 1.0g and for which there was a high confidence of a low probability of failure of at least 0.4g. The remaining list of TMI-1 equipment which could not be shown to possess an inherently high capacity had their fragilities derived in two phases. Phase 1 consisted of using conservative lower bound values for the component fragilities based on the past PRA fragility data base and on actual earthquake experience. These conservative fragility descriptions were then run through plant system models in order to determine governing accident sequences. From these accident sequences, critical components which dominated core damage and the plant risk were identified. Phase 2 consisted of developing plant-specific fragilities for only those components that dominate core damage and risk. Updated plant-specific fragilities were used together with the balance of conservative fragilities from Phase 2 in conducting the final risk analysis.

Section 5.1 contains a general description of the equipment fragility methodology with a more in-depth treatment than was provided in Chapter 3. Section 5.2 presents a set of representative example fragility derivations which provide the reader with further insight into the equipment fragility determination process. Section 5.3 presents the resulting equipment fragilities for the TMI-1 plant.

5.1 EQUIPMENT FRAGILITY METHODOLOGY

Fragility as used in probabilistic seismic safety studies is defined as a conditional probability of failure for a given hazard input. In this case, the fragility of a component or system is defined as the failure fraction as a function of effective peak ground acceleration. The development of these fragility levels combined with a discussion of the available information sources and the selection of equipment categories are all part of the equipment fragility methodology. Section 5.1 describes and defines the derivation procedure for equipment fragilities. Section 5.1.2 describes the methodology used in Phase 1 to develop the conservative fragilities for Phase 1 of the TMI risk study. Section 5.1.3 specifies the information source from which component fragilities are typically derived.

5.1.1 Fragility Derivation

The procedure used in deriving fragility descriptions is similar to that used for structural fragility descriptions, wherein, factors of safety and their variability are first developed for equipment capacity and equipment response. These two factors, along with the factor of safety on structural response, are then multiplied together to obtain an overall factor of safety for the equipment item.

$$\check{F}_E = \check{F}_{EC} \cdot \check{F}_{ER} \cdot \check{F}_{SR} \qquad (5\text{-}1)$$

$\check{F}_{EC}$ is the capacity factor of safety for the equipment relative to the floor acceleration used for the design, $\check{F}_{ER}$ is the factor of safety inherent in the computation of equipment response, and $\check{F}_{SR}$ is the factor of safety in the structural response analysis that resulted in floor spectra for equipment design. Sections 5.1.1.1, 5.1.1.2, and 5.1.1.3 of this report contain a more thorough explanation of these three factors ($\check{F}_{EC}$, $\check{F}_{ER}$, and $\check{F}_{SR}$), respectively. The overall factor of safety, $\check{F}_E$, is then multiplied by the reference earthquake peak ground acceleration to obtain fragility in terms of peak ground acceleration.

$$\check{A} = \check{F}_E \cdot A_{SSE} \qquad (5\text{-}2)$$

where:

$\check{A}$ = Median ground acceleration capacity

A_{SSE} = Peak ground acceleration of the safe shutdown earthquake

In most instances, the SSE was used as the reference earthquake; however, the OBE was used as a reference for those cases where the OBE acceptance criteria governed the equipment design.

The logarithmic standard deviation, β, for each of these factors is obtained using the logarithmic standard deviations for each of the above factors and based upon the lognormal model (Appendix A).

$$\beta_E = (\beta_{EC}^2 + \beta_{ER}^2 + \beta_{SR}^2)^{\frac{1}{2}} \qquad (5\text{-}3)$$

where β_{EC}, β_{ER}, and β_{SR} are the logarithmic standard deviations of the equipment capacity, equipment response and structural response, respectively. The logarithmic standard deviations are further divided into random variability, β_R, and uncertainty, β_U, as described in Chapter 3.

5.1.1.1 Equipment Capacity Factor

The equipment capacity factor is defined as the failure threshold divided by seismic design level. For the purposes of this study, the ultimate failure threshold is the acceleration level at which the component ceases to perform its intended function. This failure threshold could consist of a breaker tripping on a motor control center, excessive deflection of the control rod guide tubes or a support failure of the reactor vessel. Where several failure modes pertaining to the same component are found to have roughly the same capacity level, all significant failure modes are analyzed and reported.

The factor of safety for the equipment seismic capacity consists of two parts:

1. The strength factor, F_S, based on the components static strength and
2. The ductility factor, F_μ, related to the equipment's inelastic energy absorption capability.

$$F_{EC} = F_S \cdot F_\mu \tag{5-4}$$

The logarithmic standard deviation on the capacity can be derived by taking the SRSS of the logarithmic standard deviations on the strength factor and the ductility factor. The randomness and the uncertainty portion of the variability can each be derived individually from Equation 5-5, by substituting the random or the uncertainty for the strength factor and the ductility factor (i.e., β_{S_R} for β_S and β_{μ_R} for β_μ, etc.).

$$\beta_{EC} = (\beta_S^2 + \beta_\mu^2)^{\frac{1}{2}} \tag{5-5}$$

5.1.1.1.1 Strength Factor - The strength factor, F_S, is derived from the equation:

$$F_S = \frac{\frac{P_C}{P_D} - \frac{P_N}{P_D}}{\frac{P_T}{P_D} - \frac{P_N}{P_D}} \tag{5-6}$$

where P_C is the median limit state load or stress, P_N is the normal operating load or stress, P_T is the total normal plus seismic load or stress and P_D is the code design allowable load or stress.

Alternatively, this equation can be written:

$$F_S = \frac{P_C - P_N}{P_{SSE}} \tag{5-7}$$

where P_{SSE} is the seismic load or stress corresponding to the safe shutdown earthquake. The normal and the seismic loads (P_N and P_{SSE}) are typically derived from the seismic qualification reports and the other information sources described in Section 5.1.3. The calculation of the collapse or limit load, P_C, is a function of the failure mode for the specific equipment item. Equipment failures can be classified into three categories:

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

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

1. Elastic buckling in tank walls and component supports.
2. Chatter and trip in electrical components.
3. Excessive blade deflection in fans.
4. Shaft seizure in pumps.

The limit state load for this type of a failure is defined as the load or stress level where functional failure occurs.

Brittle failures are defined in this study as those failure modes which have little or no system inelastic energy absorption capability. Examples of brittle type failures include:

1. Anchor bolt failures.
2. Component support weld failures.
3. Shear pin failures.

Each of these failure modes have the ability to absorb some inelastic energy on the component level, but the plastic zone is very localized and the system ductility for an anchor bolt or a support weld is very small. Thus, the collapse load for a brittle failure mode is defined as the median ultimate strength of the material. For example, consider a transformer structure whose anchor bolts have been determined to be the critical failure mode. Under seismic loading, the massive transformer will typically be stressed well below its yield level while the bolts are being stressed well above the bolt yield level. The amount of system inelastic energy absorption provided by the bolts' plasticity is negligible when compared to the seismically-induced kinetic energy of the transformer structure, and thus, these bolts will fail in a brittle mode once the ultimate bolt strength is reached.

Ductile failures coincide much more closely with the structure failures which were described in Chapter 4. Ductile failure modes are those in which the structural system can absorb a significant amount of energy through inelastic deformation. Examples of ductile failure modes include:

1. Pressure boundary failure of piping
2. Structural failure of cable trays
3. Structural failure of ducting
4. Polar crane failure.

The collapse load for ductile failure modes consists of the median yield strength of the material for tensile type loading conditions. For bending type failure modes, the yield point is defined as the limit load or stress to develop a plastic hinge. The ductility factor will then quantify the inherent safety factor above the yield strength to the failure threshold.

Each variable within Equations 5-6 and 5-7 has an associated lognormal probability distribution to express its combined randomness and uncertainty. To find the overall variance on the strength factor, a technique commonly referred to as the "Second Moment Method" is utilized. The mean and variance of a function comprised of lognormally distributed variables can be derived utilizing the moments (i.e., the mean and variances) of the logarithms of the distribution of each variable (Reference 29). The resulting equation for the logarithmic standard deviation on the strength factor derived from Equation 5-6 is given below:

$$\beta_S = \left[\frac{P_C^2}{\left(P_C - P_N\right)^2} \cdot \beta_C^2 + \frac{P_T^2}{\left(P_T - P_N\right)^2} \cdot \beta_T^2 + \frac{\left(P_C - P_T\right)^2 \cdot P_N^{\,2}}{\left(P_T - P_N\right)^2 \cdot \left(P_C - P_N\right)^2} \cdot \beta_N^2 \right]^{1/2} \qquad (5\text{-}8)$$

where:

β_C = Logarithmic standard deviation on the collapse or limit load (stress).

β_T = Logarithmic standard deviation on the total load (stress).

β_N = Logarithmic standard deviation on the normal load (stress).

Similarily, the equation for the logarithmic standard deviation on the strength factor derived from Equation 5-7 is:

$$\beta_S = \left[P_C^2 \cdot \beta_C^2 + (P_N - P_C)^2 \cdot \beta_{SSE}^2 + P_N^2 \cdot \beta_N^2\right]^{1/2} \Big/ (P_C - P_N) \qquad (5\text{-}9)$$

where:

β_C and β_N have previously been defined, and

β_{SSE} = logarithmic standard deviation on the seismic load (stress).

5.1.1.1.2 Inelastic Energy Absorption Factor - The inelastic energy absorption capability of a piece of equipment is quantified by the inelastic energy absorption factor (or ductility factor). Brittle failure modes and functional failure modes typically have a ductility factor of 1.0, while ductile type failure modes have ductility factors which are a function of a deamplification factor. Section 4.1.2 of this report describes in great detail the methodology utilized in deriving an appropriate ductility factor for TMI-1. The ductility factor is based on the Riddell-Newmark methodology presented in Reference 7, but is has been updated to reflect the correlation between earthquake magnitude and system ductility. The median ductility factors and their variabilities were established in Section 4.1.2 as a function of the component's natural frequency, and are summarized below:

a. For the 2 Hz to 8 Hz range,

$$F_\mu = \left[(q+1) \cdot \mu^* - q\right]^r \qquad (5\text{-}10)$$

where

$q = 3.0 \times j$

$r = 0.48 \times j$

j = percent of critical damping to be used.

μ^* = effective ductility ratio

$= 1.0 + C_D(\mu - 1.0)$

C_D = factor accounting for the earthquake duration; for TMI equipment, this factor equals 1.1. This differs from the $C_D = 1.0$ which was reported in Chapter 4 for shear wall structures because of the different hysteresis characteristics of the two types of components.

b. For the rigid range,

$$F_\mu = \mu^{*\,-0.13} \tag{5-11}$$

where μ^* is as previously defined.

c. For the range 8 Hz < f < rigid range.

A linear interpolation utilizing log-log paper is applicable for ductile equipment with natural frequencies in this range. A point at 8 Hz should be plotted using F from Equation 5-10 and another point should be plotted at the lowest unamplified (rigid) frequency for the floor spectrum using Equation 5-11. A line drawn between these two points on log-log graph paper will uniquely determine the ductility factors within this frequency range.

The variabilities for these median ductility factor derivations are evaluated by estimating a 1% probability (-2.33β) that the actual ductility factor is less than 1.0. Thus, the following equations determine the composite variability, randomness and uncertainty, respectively.

$$\begin{aligned} \beta_{\mu_C} &= \frac{1}{2.33} \ln(F_\mu) \\ \beta_{\mu_R} &= 0.8 \times \beta_{\mu_C} \\ \beta_{\mu_U} &= 0.6 \times \beta_{\mu_C} \end{aligned} \tag{5-12}$$

The ductility ratio, μ, itself is based upon the recommendations given in Reference 6. This reference gives a range of ductility values to be used for design. The upper end of this range is considered to be a median value. Engineering judgment was utilized to match the applicable category from Reference 6 to a particular failure mode for the equipment component.

5.1.1.2 Equipment Response Factor

The response factors are an estimate of the conservatism or unconservatism that may have existed in the computation of seismic response during the design process. In this section, individual response factors are described for both plant specific and generic equipment. These factors differ according to the seismic qualification procedure which was used in the equipment design.

There are three types of seismic qualifications which were performed for TMI-1 plant equipment:

1. Dynamic Analysis.
2. Static Analysis.
3. Testing.

For equipment qualified by dynamic analysis, the important variables that affect the computed response and its dispersion are:

1. Qualification Method (F_{QM})
2. Spectral Shape (F_{SS})
3. Modeling (effects mode shape and frequency results) (F_M)
4. Damping (F_D)
5. Combination of Modal Responses (for response spectrum method) (F_{MC})
6. Combination of Earthquake Components (F_{ECC})

For equipment qualified by static analysis, two subdivisions must be considered. For rigid equipment, variabilities due to spectral shape, combination of modal responses, damping, and for the most part, modeling errors are eliminated. If the equipment is flexible and was designed via the static coefficient method, the dynamic characteristic variables and their variability must be considered. This involves estimating the range of frequency of the equipment and introduces a much larger uncertainty in quantifying the response factor.

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

1. Qualification Method (F_{QM})
2. Spectral Shape (F_{SS})
3. Boundary Conditions in the Test vs Installation (F_{BC})
4. Damping (F_D)
5. Spectral Test Method (sine beat, sine sweep, complex waveform, etc.) (F_{STM})
6. Multi-directional Effects (F_{MDE}).

The overall equipment response factor is the product of each of these variables. The overall variabilities (uncertainty and randomness) are calculated by taking the SRSS of the individual logarithmic standard deviations for each of the variables. A brief description of each of the variables used to develop the equipment response factor is provided below. A more detailed discussion is contained within Reference (30).

5.1.1.2.1 Qualification Method Factor - The qualification method factor is a measure of the conservatism/unconservatism involved in the seismic qualification method used to seismically qualify the component. Analytical qualifications can be separated into static analysis and dynamic analysis techniques. The inherent safety factor in using these qualification techniques is discussed below, while the variability on this factor is generally accounted for within the damping, modeling and mode combination factors (i.e., $\beta_{QM_R} = \beta_{QM_U} = 0.0$).

5.1.1.2.1.1 Static Analysis - The static coefficient method is intended to be a conservative upper bound method by which simple components may be qualified. Typically, the peak spectral acceleration is multiplied by a coefficient and this product is multiplied by the weight of the component to determine an equivalent static load to be applied at the subsystem center of gravity. If the component is comprised of more than one lumped mass, the same procedure may be applied at each lumped mass point in the static model or may be applied as a uniformly distributed load on the static model. If the component is rigid (i.e., its fundamental frequency is above the frequency where the response spectrum returns to the zero period acceleration), the degree of conservatism in the response level used for design is the ratio of the specified static coefficient divided by the zero period acceleration of the floor level where the equipment is mounted. If the equipment is flexible and responds predominantly in one mode, the degree of conservatism is the ratio of the static coefficient to the spectral acceleration at the equipment fundamental frequency.

5.1.1.2.1.2 Dynamic Analysis - Response spectrum, mode superposition time-history and direct integration time-history dynamic analysis methods may be applied in subsystem response analyses. If response for a single degree-of-freedom model with best estimate material properties and damping are computed by the response spectrum method, the mode superposition time-history method or the direct integration time-history method, we would expect to obtain equal median centered results assuming that the response spectrum and time-history inputs are compatible.

The response spectrum method was extensively used for dynamic analysis of components and systems within the TMI plant. If the applicable TMI floor response spectra were utilized in the design analysis, the qualification method factor, F_{QM}, is equal to unity and the variability is zero. If conservative generic spectrum were used to seismically qualify a component, F_{QM} is the ratio of the spectral acceleration from the generic spectrum divided by the spectral acceleration from the TMI site-specific spectra evaluated at the components' fundamental frequency.

5.1.1.2.1.3 Testing - In vibration testing, the test response spectrum generally envelopes the required response spectrum by approximately ten percent or more depending on the frequency range. If the test response spectra are available within the test report, the overtest safety factor will be accounted for in the strength factor. The qualification method factor (F_{QM}) and variability (β_{QM}) will therefore be unity and zero, respectively. If the component fragility is being based on testing where the test response spectra are not available, F_{QM} and β_{QM} account for the overtest safety factor and variability on a generic case-by-case basis.

Fragilities derived on the basis of generic U.S. Army Corps of Engineers shock test results (see Section 5.1.3.6 of this report) have the following fragility parameters:

$$\check{F}_{QM} = 1.04$$

$$\beta_{QM_R} = 0.0$$

$$\beta_{QM_U} = 0.11$$

These values are based on the data within Reference 30.

5.1.1.2.2 Equipment Spectral Shape Factor - The TMI floor response spectra were computed by means of a simplified time-history (T/H) seismic analysis. The overall dynamic response of each of the critical buildings was modeled by lumping the mass of the structure and rigidly attached components generally at each of the floor levels. The conservatism/unconservatism involved in developing the floor response spectra from the ground response spectra is quantified with the equipment spectral shape factor. The conservatism/unconservatism involved in using the specified TMI design response spectra in lieu of median Safe Shutdown Earthquake spectra is quantified in development of the spectral shape factor associated with the structural response factor (See Section 5.1.1.3).

The response spectrum method is often referred to as being conservative, however, the conservatism compared to a time-history analysis is primarily due to the method of developing the spectrum. Spectra used for design purposes are generally smoothed and the peaks are widened such that the resulting design spectrum is conservative. In addition, conservatism is generally introduced in the development of the artificial time-history. The combined effect of the two conservatisms make up the equipment spectral shape factor.

5.1.1.2.2.1 <u>Peak Broadening and Smoothing</u> - The effect of smoothing and peak broadening varies with structure, elevation, frequency and damping. For any particular frequency, this peak broadening and smoothing safety factor can be computed from Equation 5-13 below.

$$F_{SS_1} = \frac{S_a \text{ (broadened and smoothed)}}{S_a \text{ (unbroadened and unsmoothed)}} \qquad (5\text{-}13)$$

where:

F_{SS_1} = Spectral shape factor due to peak broadening and smoothing

S_a = Spectral acceleration value

The variability in this factor is a function of how well the frequency can be defined. If the frequency can be defined within a certain range, then the variability can be established by calculating the range of F_{SS_1}

values for this frequency range. Since the variability, β_{SS}, is due to the shift in the frequency, it is considered to be all uncertainty.

5.1.1.2.2.2 Artificial Time-History Generation - Studies have been conducted which show that conservatism is involved in the current practice of generating floor spectra in structures using artificial time-histories. These artificial time-histories result in response spectra that conservatively envelope the applicable ground spectra. For instance, Reference 33 indicates that the average industry-generated artificial time-history tends to introduce about 10 percent conservatism except at high frequencies for which the conservatism is about 20 percent at 33 Hz.

The floor response spectra for TMI-1 were generated using the Biggs' methodology in Reference 34. This simplified methodology utilizes the ground response spectrum and the results of the response spectrum analysis of the supporting structure. Mode shapes and frequencies from the supporting structure are used in conjunction with amplification curves obtained from a model which was subjected to four actual strong motion earthquake time histories. Comparison studies within Reference 34 shows the Biggs method to be slightly conservative throughout the frequency range. A median factor of 1.1 is judged to be appropriate, which is identical to the factor of conservatism identified in Reference 33 for artificial time history analyses. The lower bound (-1.65β) on this spectra generation factor is taken as 1.0 since the comparison curves show this to be the case. The variability is all uncertainty since it varies with the frequency and it is calculated to be equal to 0.06.

The overall spectral shape factor was generated by taking the product of the peak broadening and smoothing factor times the artificial time-history factor.

5.1.1.2.3 Modeling Factor - In any dynamic analysis there is uncertainty in resonse due to assumptions made in modeling the structure, modeling boundary conditions and representing material behavior. Modeling of complex systems is usually conducted using nominal dimensions, weights, and material properties and is done in such a manner that further refinement of mesh size in a finite element representation will not significantly alter the calculated response. Representation of boundary conditions in a model may have a significant influence on the response. The misrepresentation of boundary conditions in the dynamic model by assuming greater or lesser stiffness or treating nonlinear gap effects linearly cannot be quantified generically and each model must be treated specifically to determine a response factor for modeling. Assuming that the analyst does his best job of modeling, modeling accuracy could be considered to be median centered (i.e., $F_M = 1.0$) with the variability in each of the modeling parameters amounting to variability in calculated mode shapes and frequencies. The error in calculation of mode shapes and frequencies then has an effect on the computed response.

For complex equipment which have been analyzed using state-of-the-art dynamic analysis, the coefficient of variation on response (approximate logarithmic standard deviation) is about 0.20. For simple single-frequency systems with fundamental frequencies in the amplified portion of the spectra, the coefficient of variation is about 0.15. For single frequency systems with fundamental frequencies out into the rigid range, the coefficient of variation is 0.0. These variabilities are considered to be all uncertainty and are based on past experience and engineering judgment.

5.1.1.2.4 <u>Damping Factor</u> - The basis for the damping factor has been addressed in Section 3.3.2 of this report. Table 3-1 shows the damping values used for the SSE design analysis of TMI equipment. Median damping values and their variabilities are a function of the material, construction details, size and stress level. Reference (30) suggests that median damping for equipment at the SSE level is about five percent. Thus, for single-degree-of-freedom systems the damping factor for equipment is:

$$F_D = \frac{S_a \text{ (qualification)}}{S_a \text{ (median)}} \qquad (5\text{-}14)$$

where:

$S_{a(qual)}$ = Spectral acceleration using the qualification design analysis damping and evaluated at the equipment fundamental frequency

$S_{a(median)}$ = Spectral acceleration using the expected median damping and evaluated at the equipment fundamental frequency.

For multi-degree-of-freedom systems, Equation 5-14 can be altered to reflect the summation of the spectral accelerations at each of the frequencies multiplied by their associated mass participation factors.

There is variability in damping and associated response that must be considered. It is indicated within Reference 30 that for a median damping value of 5 percent, the minus one logarithmic standard deviation value is about 3.5 percent. The variability in damping results in a logarithmic standard deviation in response equal to:

$$\beta_{D_u} = \ell n \left(\frac{S_{a_{\zeta = 3.5\%}}}{S_{a_{\zeta = 5.0\%}}} \right) \tag{5-15}$$

where $Sa_{\zeta = 5\%}$ is the 5 percent damped spectral acceleration and $Sa_{\zeta = 3.5\%}$ is the 3.5 percent damped spectral acceleration taken at the equipment fundamental frequency using the applicable floor response spectra. The resulting logarithmic standard deviation on the damping response factor, from Equation 5-15 above, is considered to be all uncertainty. An additional randomness variability estimated at approximately 20 percent of the uncertainty variability reflects the earthquake time-histories' effect on the median damping value.

5.1.1.2.5 Mode Combination Factor - The modal combination technique utilized within the TMI seismic design analysis was described in general in Chapter 3 of this report. A square-root-of-the-sum-of-the-squares (SRSS) methodology was used for all TMI equipment. This SRSS method is considered median centered.

The response factor for combination of modes is then considered to be 1.0. The variability associated with mode combination depends upon the complexity of the model. For multi-degree-of-freedom systems, Reference (30) recommends that the coefficient of variation due to mode combination is approximately 0.15. For single-degree-of-freedom flexible systems, the coefficient of variation due to mode combination is estimated within Reference 30 to be approximately 0.10. For a single-degree-of-freedom rigid system, the COV is by definition zero. The variability due to mode combination is considered to be all random due to the random phasing of modes.

5.1.1.2.6 Earthquake Component Combination Factor - The TMI plant design analyses earthquake components were typically combined by the absolute sum of the worst horizontal plus the vertical components. Two methods of combining earthquake components have been determined to provide median centered results. The first method is to combine the components by square-root-of-the-sum-of-the-squares (SRSS) and the second method is the 100%, 40%, 40% method contained in Reference 13. Reference 13 recommends that the response can be represented by combining the worst case horizontal response with 40 percent of the orthogonal horizontal response and 40 percent of the vertical response. The SRSS method must be applied to the end item of interest, while the 100%, 40%, 40% method can be applied at the input seismic load stage or at the stress intensity of interest stage with equivalent results. For this reason, comparing this 100%, 40%, 40% methodology to the TMI design criterion results in a response factor for combination of earthquake components. The magnitude of the factor depends, however, on the orientation, failure mode and response characteristics of the component under consideration.

A generic study was conducted to develop earthquake component combination response factors and their variabilities for common two- and three-dimensional equipment idealizations. The amount of conservatism/ unconservatism and the associated variability on this factor are generally a function of the following:

1. The number and direction of earthquake components which affect the failure mode under consideration (e.g., piping failures can be influenced by all three directional responses, but a particular relay can fail due to a particular horizontal seismic excitation while remaining unaffected by the vertical and the other horizontal directions)

2. The amount of coupling that exists between directional response (i.e., does an x direction excitation cause a response in the y and z directions)

Table 5-1 contains the earthquake component combination response factors for those cases which were applicable to TMI equipment. The variability involved in the phasing of the three earthquake directional components was considered to be all random, while the variability due to the degree of coupling involved between directions was considered to be all uncertainty.

5.1.1.2.7 <u>Boundary Conditions Factor (Testing)</u> - The boundary conditions utilized in equipment seismic testing can be a significant source of variability that depends almost solely upon the diligence of the test laboratory and the qualification review organization. In general, a component that is bolted to the floor in a nuclear power plant and which is similarly bolted to a shake table for qualification testing, will experience little variability in response factor due to boundary conditions. Carelessness on the part of the various organizations involved in design, fabrication, testing and installation can result in a significant variability. For instance, the lack of a specified bolt torque at the mounting interface can result in a difference between the testing and installation condition which could have a pronounced impact on the response factor.

The variability of the subsystem response due to test boundary conditions would come primarily from mode shape and frequency shift. The variability of mode shape and frequency and resulting response due to boundary conditions varies considerably for different generic types of equipment. For a large majority of tests conducted by reputable testing laboratories, the boundary condition factor is 1.0. Engineering judgment must be utilized in calculating boundary condition factors for those cases where the component to test table attachment mechanism is not representative of the actual in-plane condition. The variability is all uncertainty and can be calculated based on spectral accelerations obtained from estimating a 90 percent confidence interval on the equipment frequency. The boundary condition uncertainty is generally estimated to be 0.11 based on values derived in the SSMRP study (Reference 30).

5.1.1.2.8 Spectral Test Method - Synthesized time-histories are currently developed directly from the Required Response Spectrum at most testing laboratories. A much better approach, as recommended in Reference 31, is to synthesize a time-history that corresponds to a power spectral density which closely envelopes the RRS rather than make the direct step from the RRS to the synthesized time-history. This approach tends to smooth out the input time-history, resulting in less chance for an equipment mode to coincide with a significant peak or valley. Reference 32 recommends a spectral test method factor of unity and a total variability of 0.20. This variability is entirely uncertainly since the use of better equipment and techniques could eliminate most of the uncertainty.

5.1.1.2.9 Multi-Directional Effects - The multi-directional effects factor is a measure of the conservative/unconservatism and corresponding variability involved in testing the three different earthquake directional components. TMI equipment fragilities were developed from plant-specific and generic test data and are based on two types of testing: biaxial and uniaxial. Biaxial qualification tests are conducted by exciting the equipment in one horizontal direction at a time along with the vertical direction, using randomly phased input time-histories. Uniaxial qualification tests, on the other hand, are conducted in each of the three directions independently. Biaxial testing was conducted for most plant-specific equipment qualified for the TMI plant. The shock tests conducted during the SAFEGUARD program were, in many cases, single axis tests with complex waveforms consisting of superimposed sine beats. Some biaxial testing data were included when deriving the generic SAFEGUARD fragilities, but were scaled to an equivalent uniaxial input. Thus, multi-directional effect factors were developed for biaxial testing (used for fragilities developed for most plant specific TMI testing) and uniaxial testing (used for fragilities based on generic SAFEGUARDS test data).

5.1.1.2.9.1 Biaxial Testing - There is a slight unconservatism involved in biaxial testing in that the actual input during a seismic event is three-dimensional. This unconservatism along with its associated variability is a function of both the phasing and the coupling between earthquake directional components. The degree of unconservatism associated with biaxial testing can be defined as the median response vector for biaxial testing divided by the median three-axis response. The resulting response factor based on both phasing and coupling is calculated to be 0.853. The variability due to phasing is a function of the earthquake, and thus, is all random. The variability due to coupling is all uncertainty.

The multi-directional effects factor and its associated β's for random vibration biaxial testing are:

F_{MDE} = 0.853

β_{MDE_R} = 0.08

β_{MDE_U} = 0.06

5.1.1.2.9.2 Uniaxial Testing - A uniaxial test is, in general, unconservative in that coupling and phasing between the three-directional earthquake components is not accounted for. The degree of unconservatism associated with uniaxial testing can be defined as the median response vector for uniaxial testing divided by the median three-axis response. The resulting response factor based on both phasing and coupling is calculated to be 0.769. The phasing variability is random and is identical to that for the biaxial case, i.e., 0.08. The uncertainty variability due to coupling, based on the uncoupled case and the 100 percent coupling case being $\pm 1.65\beta$ extremes, is calculated to be 0.11.

Thus, the multi-directional effects factor and its associated β's for uniaxial testing is:

F_{MDE} = 0.769

β_{MDE_R} = 0.08

β_{MDE_U} = 0.11

5.1.1.3 Structural Response Factors

Structural response factors as they relate to structural capacity for the safety-related structures within TMI are derived in Chapter 4. The variables pertinent to the structural response analyses used to generate floor spectra for equipment design are the only variables of interest relative to equipment fragility. The applicable variables for equipment from those analyses are:

1. Spectral Shape
2. Damping
3. Modeling
4. Soil-Structure Interaction.
5. Inelastic Energy Absorption of the Building

The explanation of each of these variables is contained in Chapter 4 and will not be repeated here. Note, the combination of earthquake components is not included in structural response since that variable is addressed for specific equipment orientation in the treatment of equipment response. As discussed in Chapter 4, a totally independent evaluation of the capacities of the important structures was undertaken in this effort. As a result, the generated median structural response factor was 1.0 and included its associated variabilities. This independent analysis employed the median ground spectra to define seismic input. In evaluating equipment acceleration capacities which are based upon design analysis results, a spectral shape factor associated with structural response must be computed which compares the 5% damped median spectrum. The resultant structural response factors pertaining to the equipment fragility derivation are included in Table 5-2. Note that the structural response factors for each particular structure was broken up into two segments. Equipment with capacities less than the approximate building yield strength have Structural Response Factors in the "a" row, and equipment with capacities approximately equal to or greater than the structure yield strength have Structural Response Factors in the "b" row. The approximate yield level for each of the buildings was estimated by taking the ground acceleration capacity for the lowest structural failure mode (see Chapter 4) and dividing it by the inelastic energy absorption factor.

The structural response factors have been derived on the basis of the structure being at its failure threshold level. These factors, as shown in Chapter 4, apply directly to equipment whose acceleration capacities are greater than the buildings acceleration capacity. For equipment whose seismic capacity level has been reached before the structure has reached its seismic capacity, these factors are optimistic. Reference 13 recommends 10% median damping for reinforced concrete at or just below the yield condition and five percent median damping for reinforced concrete at the one-half yield condition. In addition, the structures ductility does not modify the response of the equipment unless the equipment fragility is just above the building's yield level. Thus, for the condition of the equipment capacity being less than the structure's yield level, 5% structural damping is considered median and the structure's ductility factor is effectively unity. For the case where the equipment capacity is greater than the structure's yield level, 10% structural damping is considered median and the inelastic energy absorption factor (ductility factor) is appropriate to include.

It should be noted that when the building goes inelastic that the actual floor level acceleration will be decreased over that which is predicted using the elastic structural model. At the same time, the displacement will increase over that which is predicted by the elastic model. Thus, equipment which are acceleration sensitive must have their capacities scaled up (as described in the previous paragraph) to reflect the actual lowering of the floor acceleration due to building ductility, and equipment which are displacement sensitive must have their capacities similarily scaled down. The great majority of the equipment within nuclear plants are acceleration sensitive. The exception to this are interconnecting piping systems which run between separate buildings. If these piping systems are designed such that differential displacements between buildings causes a high stress in either the supports or the piping itself, then it is much more critical to have either structure go inelastic than to remain elastic. For TMI-1, all of the critical structures remain elastic up to the 1g cut-off level, except for the control building. The control building does not contain any critical interconnecting piping which might be affected adversely due to building inelasticity. Therefore, the adverse effects of increasing displacements due to the inelasticity of structures is felt to have little impact on TMI equipment.

5.1.2 Conservative Fragility Methodology

The first phase of the TMI seismic fragility analysis incorporated fragilities with conservative median values and realistic lower tails. These so-called "conservative fragilities" were based on the results of 14 previous seismic PRA's conducted by SMA. The purpose of utilizing these conservative fragilities was to identify those components which would not contribute significantly to the plant risk, even when conservatively low fragilities were run through the plant system models. This methodology allows SMA to "screen out" components which do not affect the risk significantly and to concentrate resources on the more critical components. Actual fragilities are then derived in Phase 2 of this study for these critical components using the methodology presented in Section 5.1. The final risk analysis is then conducted using the updated plant-specific fragilities along with the balance of the conservative fragilities. Sections 5.1.2.1 and 5.1.2.2 describe this Phase 1 methodology in greater detail for the conservative median capacities and the realistic lower bound values, respectively. The Phase 2 actual fragility derivation methodology has already been presented in Section 5.1.1.

5.1.2.1 Conservative Median Values

As previously stated, the conservative median values were based on the results of 14 previous PRA's conducted by SMA. The median fragility values for a particular type of equipment were tabulated for each of the 14 different studies. The lowest median ground acceleration capacity of this group was taken as a conservative lower bound, $\overset{*}{A}$, on the TMI component's median capacity. Table 5-3 contains an example for the conservative median derivation for Emergency Batteries.

Table 5-4 contains the conservative fragilities for all of the seismically critical TMI-1 equipment. Equipment which could be shown to possess a median capacity greater than 1g together with a HCLF (high confidence of a low frequency of failure, i.e., 95% confidence of less than a 5% frequency of failure) value greater than 0.4 g's are identified with " > 1.0 g's" in the table. Components with fragilities in this category are not expected to influence the risk, based on PLG's assessment of the TMI-1 hazard curves.

5.1.2.2 Realistic Lower Bounds

The High Confidence of a Low Fraction of failure (HCLF) values for TMI PRA purposes is defined as the 95% confidence of less than a 5% failure fraction. For logarithmic distributions like the component fragilities, this results in the following equation.

$$\mathrm{HCLF} = \check{A} \times e^{-1.65(\beta_R + \beta_u)} \qquad (5\text{-}16)$$

The HCLF values derived for Phase 1 of the TMI-1 PRA were based primarily on the earthquake experience data collected for the Seismic Qualification Utilities Group (SQUG). SQUG experience data (Reference 35) is derived from past seismic experiences of conventional power plant equipment. The Senior Seismic Review and Advisory Panel (SSRAP, Reference 36) has reviewed the available data base on eight classes of equipment and has concluded that the equipment installed in nuclear plants is generally similar to and at least as rugged as that installed in conventional plants. SSRAP has established certain minimum seismic capacities for these equipment which are judged to be representataive of the HCLF values. These minimum seismic capacities endorsed by SSRAP had several restrictions attached to them (e.g., anchorage and functionality must be verified). SMA conducted a detailed walkthrough of TMI-1 in order to verify that anchorage was adequate and that the equipment types within this plant were represented by the group of similar equipment from past PRA's. Operability was established for environments up to at least the

SSE during qualification testing. Table 5-4 contains β_R and β_U values for all of the critical equiment which have been derived from HCLF values using Equation 5-16.

5.1.3 Information Sources

Several sources of information are utilized in a PRA from which to develop plant specific and generic fragilities for equipment. These sources include:

1. Seismic Qualification Design Reports
2. Seismic Qualification Test Reports
3. Final Safety Analysis Report (FSAR)
4. Seismic Qualification Review Team (SQRT) Submittals
5. Past Earthquake Experience
6. United States Corps of Engineers Shock Test Reports
7. Specifications for the Seismic Design of Equipment

The first five of these information sources are termed "plant specific" since they pertain to specific equipment within the TMI plant. The remaining three information sources are termed "generic" since they constitute data generated for similar types of equipment or are definitions of design requirements, in lieu of actual design results. Plant-specific sources are preferred since they have been generated for the specific items in question and their uncertainty level is reduced from those of the generic sources.

Depending upon the uniqueness of the equipment, the failure mode, inelastic energy absorption capability and the dynamic characteristics of the equipment, a plant-specific or a generic derivation of the fragility description may be appropriate. The factors of safety relative to the

Safe Shutdown Earthquake are widely variable. In general, flexible equipment such as piping, which possesses the ability to undergo large inelastic deformation, will have a factor of safety against failure of many times the Safe Shutdown Earthquake even if stressed to the maximum code allowable stress. Such equipment is a prime candidate for a generic derivation of fragility descriptions. The increased uncertainty inherent in a generic derivation does not have much influence on the outcome of the seismic risk analysis if large safety factors can be demonstrated. On the other hand, if rigid equipment with relatively brittle failure modes are stressed to code allowable for the Safe Shutdown Earthquake, the factor of safety against failure may be considerably smaller and a generic treatment may result in unsatisfactory risk predictions. Fortunately, plant-specific analyses have shown that most rigid equipment have stresses well below the allowable and large safety factors are present.

Each of the seven information source categories will be discussed briefly below.

5.1.3.1 Seismic Qualification Analysis Reports

Several seismic qualification analysis reports were reviewed in deriving fragility levels for TMI equipment. Stress and load summary information are used in deriving the capacity factors and information on the analysis methodology are used in deriving the response factors.

5.1.3.2 Seismic Qualification Test Reports

Some examples of test reports for equipment qualified by testing were reviewed. Qualification test reports, by themselves, cannot be utilized to develop fragility relationships unless the equipment has been tested to increased vibration levels up to failure. Consequently, most equipment qualified by test was treated generically with the test qualification report data (when reviewed) being considered as part of the population of test data on similar generic equipment.

5.1.3.3 Final Safety Analysis Report

The FSAR contained very little stress summary information on equipment components within the TMI plant. The FSAR information was primarily utilized in helping to develop response factors since it contains some of the qualification criteria and methodology for TMI.

5.1.3.4 Seismic Qualification Review Team (SQRT) Summaries

SQRT summaries have not been conducted for TMI, thus no SQRT information was used in the seismic PRA study.

5.1.3.5 Past Earthquake Experience

Past earthquake experience is valuable for establishing fragilities for equipment which have historically been vulnerable. Most equipment survives without any apparent damage and the historic experience must be treated the same as a qualification test. Earthquake experience has typically been used to estimate fragility levels for off-site power systems and non-seismically qualified equipment.

5.1.3.6 United States Corps of Engineers Shock Tests

Qualification tests usually are conservative compared to the required test level, but the test levels are generally not severe enough to reach a state of malfunction. In these cases and in cases where qualification information is not readily available, generic fragility data from the SAFEGUARD program is a possible source of information. In the SAFEGUARD program, the U.S. Corps of Engineers conducted fragility testing on a large number of electrical, mechanical, electro-mechanical and instrumentation and control equipment, References 37, 38 and 39. During the SAFEGUARD program, off-the-shelf equipment was procured rather than specially-engineered equipment qualified for shock and vibration environments. The equipment was very similar to equipment installed in some of the earlier nuclear power plants. Consequently, the test performance of selected SAFEGUARD equipment is indicative of similar nuclear power plant equipment. In the Seismic Safety Margin Research

Program (SSMRP), the Corps of Engineers test data and methodology were used to develop generic fragility descriptions of equipment which can be utilized in PRA's (Reference 32).

5.1.3.7 Specification on the Design of Equipment

Specifications for seismic qualification of selected TMI equipment were provided by GPU. In cases where plant-specific qualification reports were not reviewed, knowledge of the vendor requirements plus generic fragility and qualification test data were combined to develop fragility descriptions.

5.2 EQUIPMENT FRAGILITY EXAMPLES

Because of the amount of equipment to be included within the risk model, it is impractical to describe the specific fragility derivation for each piece of equipment. This section contains selected examples of fragility derivations which are judged to be representative of the different types of analyses which had to be undertaken for TMI equipment. The equipment fragility derivation categories applicable to the TMI-1 PRA are:

1. Conservative fragility derivation with the lower bound HCLF value derived from earthquake experience data.
2. Plant-specific fragility derivation based on seismic qualification reports.
3. Plant-specific fragility derivation based on similarity to an identical equipment item in another nuclear plant.
4. Generic fragility derivation based on past earthquake experience (non-seismically qualified components).

An example of TMI-1 equipment whose fragility derivation stems from each of the above categories is included in this section.

5.2.1 Example of a Conservative Fragility Description

The diesel generator system air receiver tanks will be utilized as an example for this category. The air receiver tank fragility data from past PRA studies has been collected and is shown in Table 5-5. The minimum value from past PRA data is shown to be 0.68 g's. This minimum value of 0.68 g's is then utilized as a conservative capacity of the TMI-1 air receiver tank. It is recognized that a plant-specific fragility analysis of this component would almost assuredly lead to a higher median capacity level. The purpose of the Phase 1 conservative fragility derivation is to screen out the non-critical components, not to produce actual median capacity levels.

The HCLF value for the air receiver tank was based on the SSRAP recommendations provided in Reference 35 (see Section 5.1.2 of this report). Anchored tanks within conventional power plants have demonstrated seismic capacities of up to at least 0.3 g's ground accelerations. This 0.3 g's is judged to be an appropriate lower bound on capacity. In order to use Equation 5-16 to define β_R and β_u, an estimate must be made for either one of these unknowns in order to solve for the remaining β value. The randomness variability is predominantly a function of the earthquake characteristics and has been shown in the past to be in the neighborhood of 0.25. Using $\beta_R = 0.25$ as an estimate, β_u can be calculated to be:

$$\beta_u = \frac{-1}{1.65} \ln \left(\frac{0.3}{0.68}\right) - 0.25$$

$$\beta_u = 0.25$$

The derived fragility parameters ($\overset{*}{A} = 0.68$ g's, $\beta_R = 0.25$, $\beta_u = 0.25$) are shown in Table 5-4.

It should be noted that the SSRAP recommendations applied to anchored equipment within 40 feet of the grade level. The air receiver tank anchorage was visually inspected and found to be adequate during a plant walkdown. The tank itself is located on the diesel generator ground floor. Thus, usage of the earthquake experience data is appropriate for the TMI air receiver tank.

5.2.2 Example of a Specific Fragility Derivation

The Nuclear Service River Water Pumps (NR-P-1A/B/C) were chosen as the example of equipment whose fragilities were derived from plant-specific information. The NSRW pumps are vertical pumps with a 43-foot unsupported cantilever column. The pump shaft consists of four 10-foot lengths connected in series by flexible couplings. These pumps are located at EL. 308' in the Intake Screen and Pumphouse.

The seismic qualification analysis contained within Reference 40 is the basis for the NSRW pump's fragility. The stress and loading results within the qualification analysis are utilized without an independent SMA analysis since complete checking and acceptance of the design analysis was already required by GPU at the time of the qualification. Conservatisms and uncertainty in the design procedures and methodology (spectra, damping, frequency, mode combination, etc.) are evaluated and quantified in development of the pump fragility. The individual fragility parameters are summarized in Table 5-6, and each of these factors is discussed briefly below.

5.2.2.1 NSRW Pump Capacity Factors

An evaluation of the pump qualification report revealed that the most highly stressed areas were:

1. Motor Mounting Bolts
2. Discharge Head Mounting Bolts
3. Soleplate Anchor Bolts
4. Pump Column in Bending
5. Top Column Flange Bolts
6. Tube in Bending
7. Shaft Deflection

Strength factors were derived for each of these possible failure locations. The lowest strength factor exists at the top column flange bolts and, thus, these bolts will be the governing seismic failure mode. The following information can be obtained from the stress report:

Pump Column Frequency = 1.1 Hz
Pump Tube Frequency = 2.2 Hz
Top Column Flange Bolt Material = SAE J429 Grade 2
Bolt Yield Strength = 57,000 psi (minimum)
Bolt Tensile Strength = 74,000 psi
Faulted Condition Bolt Stress = 52,300 psi
Upset Condition Bolt Stress = 33,200 psi
Bolt Preload Stress = 6,600 psi

The faulted condition includes SSE loads, normal loads and bolt preloads, while the upset condition includes OBE loads in place of SSE loads. Since the SSE is twice the OBE, the SSE can be calculated as twice the difference between the faulted and the upset stresses.

$$\sigma_{SSE} = 2x(52.3 - 33.2) \text{ ksi}$$

$$\sigma_{SSE} = 38.2 \text{ ksi}$$

The normal stress can then be computed from given information to be:

$$\sigma_N = 52.3 - 38.2 - 6.6 \text{ ksi}$$

$$\sigma_N = 7.5 \text{ ksi}$$

Note that the bolt preload stress is not considered for the fragility analysis since it is relieved once the seismic stress overcomes it.

The limit capacity stress, σ_C, is judged to occur when each of the bolts in the bolt pattern reaches yield. Once the outer bolt in the bolt circle reaches yield, the neutral axis shifts down into the compression region and the remaining bolts pick up any further increase in load. Once all of the bolts reach yield, then the pump column is capable of deflecting relatively large amounts for further increases in loading. This point is felt to be a limit load or failure point because small angular displacements at the bolted flange connection result in large end-deflections on the 43-foot long column. Large pump column deflections are judged to result in shaft binding failures.

The design analysis was based on the neutral axis being shifted to the lowest bolt. This configuration is judged to be applicable for the inelastic condition where a "heel" load in compression balances the tensile load in all of the bolts. The limit capacity stress is calculated below:

$$\sigma_C = 1.25 \text{x} 57 \text{x} 1.33 \text{ ksi}$$

$$\sigma_C = 94.8 \text{ ksi}$$

where

1.25 = Factor to increase the minimum yield strength to the median yield strength.

57 ksi = Minimum yield strength for bolts.

1.33 = Factor to reflect the increased moment carrying capability between the linearly increasing bolt stress configuration and the constant bolt stress configuration.

The strength factor can be derived from Equation 5-7 to be:

$$\tilde{F}_S = \frac{\sigma_C - \sigma_N}{\sigma_{SSE}} = \frac{94.8 \text{ ksi} - 7.5 \text{ ksi}}{38.2 \text{ ksi}} = 2.28$$

The variability on F_S is all uncertainty and is the result of the σ_C variable alone since both σ_N and σ_{SSE} have been uniquely defined in the design analysis. Since the minimum material properties within the ASME Code have been defined as 95% confidence values, the uncertainty on the 1.25 factor is:

$$\beta_{1.25} = \frac{1}{1.65} \ln \left(\frac{1.25}{1.0}\right) = 0.14$$

The uncertainty on the failure threshold is judged to be defined as having a 95% confidence lower bound of failure occurring when the outer bolt first reaches yield. Thus,

$$\beta_{1.33} = \frac{1}{1.65} \ln \left(\frac{1.33}{1.0}\right) = 0.17$$

The uncertainty on the strength factor is determined by taking the SRSS of the contributing uncertainties.

$$\beta_{S_u} = (0.14^2 + 0.17^2)^{\frac{1}{2}} = 0.22$$

The top column flange bolt failure is expected to be a functional type failure mode of the long shaft. Thus, very little system ductility exists and most of the pump and its supporting structure will remain elastic at the point of failure. Therefore, the ductility factor will be unity and its variability will be zero. The capacity factor and its variability will then be equivalent to the strength factor and its associated variability.

$$\check{F}_C = \check{F}_S = 2.28$$

$$\beta_{C_R} = \beta_{S_R} = 0.0$$

$$\beta_{C_u} = \beta_{S_u} = 0.22$$

5.2.2.2 NSRW Pump Equipment Response Factors

The dynamic loads on the NSRW pump supports are generated from a response spectra dynamic analysis of the pump using the design floor response spectra. The conservatisms/unconservatisms involved in the design analysis will be addressed by the individual response factors delineated below.

5.2.2.2.1 Qualification Method Factor - The response spectrum qualification analysis was performed using the El Centro-based design ground response spectrum. The spectral acceleration value for the design analysis taken at the pump column 1.1 Hz fundamental frequency is 0.246g. The Intake Screen and Pumphouse structure is rigid (24.4 Hz) and the response at the pump level (308 feet) is essentially not amplified over that of the ground response. Using the median ground response spectra at 5% damping, the pump spectral acceleration at 1.1 Hz is 0.0936g. The qualification method factor is then computed to be 2.63.

$$F_{QM} = \frac{S_a \text{ (El Centro)}}{S_a \text{ (Median)}} = \frac{0.246 \text{ g}}{0.0936 \text{ g}} = 2.63$$

The uncertainty and randomness involved in using the median response spectra are accounted for within the structural response factors.

$$\check{F}_{QM} = 2.63$$

$$\beta_{QM_R} = 0.0$$

$$\beta_{QM_u} = 0.0$$

5.2.2.2.2 Spectral Shape Factor - Since the ground response spectrum was utilized in qualifying this component, no peak broadening or smoothing was applied for the pump floor response spectrum. Thus, the spectral shape factor is unity and the variabilities are zero.

5.2.2.2.3 Damping Factor - Median damping for the pump response is estimated to be approximately 5% (Reference 32). Since 5% damping was used in deriving the qualification method factor in Section 5.2.2.2.1, the damping factor is unity. Using Equation 5-15, the uncertainty is calculated to be:

$$\beta_{D_u} = \ln\left(\frac{0.88\ g}{0.78\ g}\right) = 0.12$$

Therefore,

$$F_D = 1.0$$

$$\beta_{D_R} = 20\% \times \beta_{D_u} = 0.02$$

$$\beta_{D_u} = 0.12$$

5.2.2.2.4 Modeling Factor - Section 5.1.1.2.3 reflects a factor of 1.0 with a coefficient of variation of 0.15 for components such as the long column vertical pumps.

$$F_M = 1.0$$

$$\beta_{M_R} = 0.0$$

$$\beta_{M_u} = 0.15$$

5.2.2.2.5 Mode Combination Factor - Section 5.1.1.2.5 specifies a median factor of 1.0 with $\beta_R = 0.10$ for systems which respond primarily in one mode.

5.2.2.2.6 Earthquake Component Combination Factor - The NSRW pump qualification analysis was conducted using the "worst horizontal plus the vertical" combination of directional components. This is unconservative for a case such as the cantilevered pump column where both horizontal components contribute to the failure mode. Case No. 3 from Table 5-1 applies since coupling will not occur on a circular cylinder such as the pump column.

$\check{F}_{ECC} = 0.95$

$\beta_R = 0.06$

$\beta_U = 0.0$

5.2.2.2.7 Overall Equipment Response Factor - The combined response factor is:

$\check{F}_{ER} = 2.63 \times 0.95 = 2.50$

$\beta_{ER_R} = (0.02^2 + 0.10^2 + 0.06^2)^{1/2} = 0.12$

$\beta_{ER_U} = (0.12^2 + 0.15^2)^{1/2} = 0.19$

5.2.2.3 NSRW Pump Structural Response Factors

The values presented within Table 5-2 are not applicable to the NSRW pump fragility derivation since these pumps are essentially anchored to the basemat and ground spectra are applicable. The structural response factor is unity since the effects of using the design spectrum in place of the median spectrum was accounted for in the equipment response factor. The uncertainty and randomness associated with the median ground spectra are calculated using the methodology presented in Section 5.1.1.3 to be:

$F_{SR} = 1.0$

$\beta_{SR_R} = 0.37$

$\beta_{SR_U} = 0.26$

5.2.2.4 NSRW Pump Ground Acceleration Capacity

The ground acceleration capacity for the NSRW pump was calculated using Equations 5-1 and 5-2.

$$\check{A} = 2.50\times2.28\times1.0\times0.12g = 0.68g$$

The variability was calculated by taking the SRSS of the variabilities for each of the three factors contributing to overall capacity (Equation 5-3).

$$\beta_R = (0.37^2 + 0.12^2)^{\frac{1}{2}} = 0.39 \qquad \text{(randomness)}$$

$$\beta_u = (0.26^2 + 0.19^2 + 0.22^2)^{\frac{1}{2}} = 0.39 \qquad \text{(uncertainty)}$$

The combined variability, β_C , is a measure of the overall variability contributed by earthquke randomness and uncertainty and can be obtained by taking the SRSS of β_R and β_u.

$$\beta_C = (0.39^2 + 0.39^2)^{\frac{1}{2}} = 0.55$$

This value of β_C, along with the three factors making up the overall fragility (F_{EC}, F_{ER}, F_{SR}) are tabulated in Table 5-7 along with the rest of the equipment which were addressed in Phase 2 of the TMI-1 PRA study.

5.2.3 Example of a TMI-1 Component Fragility Based on Similarity

The control rod drive mechanisms (CRDMs) have been chosen to be the example of a fragility derived from the qualification data of similar nuclear power plant equipment. An information gathering trip to Babcock and Wilcox (B&W) revealed that seismic qualification information was not available for the TMI CRDMs.

Since a specific seismic analysis was not available to derive the TMI CRDM fragility, similarity was judged to be an appropriate data source. B&W responsible engineers stated that the TMI CRDMs are identical to the Midland units. They also stated that the ground response amplification up to the CRDM mounting locations were approximately the same for these two plants. Thus, the Midland CRDM design stresses are judged to be applicable for the TMI PRA.

5.2.3.1 CRDM Capacity Factor

Two sources of information were utilized in deriving the CRDM fragility:

1. B&W seismic qualification stress reports for Midland (Structural Portion of the CRDM).
2. Midland FSAR stress summaries (Pressure Boundary Portion of the CRDM).

CRDM Supports and Structure

The most critically stressed point within the finite element model of the CRDM structure was determined to be Node Number 230. The resulting bending stresses at this node (taken from Table 5-8 of the B&W stress report) are:

SSE stress = 5,787 psi
LOCA stress = 20,067 psi
Thermal stress = 1,305 psi
Deadweight stress = 219 psi

From this summary, it is observed that the CRDM design is dominated by LOCA loading. Since LOCA has a very low probability of coinciding with a seismic event, it is not included in the fragility derivation. Thus, the CRDMs will have a relatively high seismic capacity since the combined seismic and LOCA stresses were required to be below the allowable stress.

This allowable stress was not documented in the information which SMA received, but A36 steel can be conservatively assumed since it is about the lowest grade of steel utilized in designing critical nuclear components.

The limit capacity stress for the CRDM is estimated to be the point where a plastic hinge is formed. Any further excursion into the plastic range is judged to cause control rod insertion problems.

$$\sigma_C = 36 \text{ ksi} \times 1.25 \times 1.5 = 67.5 \text{ ksi}$$

where

36 ksi = minimum material yield strength.

1.25 = factor to raise the minimum yield to median

1.5 = plastic section modulus (bending)

The strength factor is then computed to be:

$$F_S = \frac{67.5 - 1.5}{5.8} = 11.4$$

The ground acceleration capacity based on the strength factor alone would be:

$$\check{A} > 11.4 \times 0.12g = 1.37 \text{ g's}$$

The remaining response factors will raise this capacity even higher. Since PL&G have determined that components with capacities greater than 1.0g will not contribute significantly to the risk, there is no need to calculate the CRDM support fragility any more accurately.

CRDM Pressure Boundary

Table 3.9-22 from the Midland FSAR contains stress summaries for the CRDM pressure boundary. The stress summaries show that the pressure boundary design is also controlled by LOCA loads. A strength factor was derived in a manner similar to the one previously derived for the CRDM supports and the resulting factor was even larger.

Thus, it has been shown for both the CRDM pressure boundary and for the CRDM supports that the seismic capacity is relatively large and that it will not contribute significantly to the TMI-1 plant risk. The plant-specific fragility based on similarity is recorded in Table 5-7 as being greater than 1g.

5.2.4 Example of Fragility Based on Earthquake Experience

There are typically several equipment items within the list of components for a PRA for which no seismic qualification was required. These components are not designed for seismic loading; thus, they will generally have a lower capacity and a higher uncertainty than seismically qualified components. The methodology which has been utilized on the previous two examples of developing capacity factors, response factors, etc., is generally not applicable for unqualified components. The fragility levels for these unqualified components must be derived based on earthquake experience and engineering judgment. The example which has been chosen in this category is the station offsite power sytem. Figure 5-1 shows a picture of the offsite power transmission lines and their supporting stuctures for the TMI-1 site.

Failure of offsite power is governed primarily by failure of ceramic insulators. Offsite power is also frequently tripped off-line due to various electrical malfunctions during the earthquake. Reference 35 contains a list of past experience data for conventional power plants which have been subjected to an earthquake. A probabilistic assessment of these data concluded that the median ground acceleration capacity for the offsite power system is approximately 0.3g. The high confidence of a low frequency of failure point is estimated from this data to be 0.1g. Variabilities of $\beta_R = 0.25$ and $\beta_u = 0.50$ are derived from these values.

5.3 EQUIPMENT FRAGILITY RESULTS

Table 5-4 contains conservative fragility descriptions for all of the equipment which were selected for this PRA study. Conservative fragility derivations were conducted for each of these components and are reported for those items which have a ground acceleration capacity less than 1.0g. Equipment with ground acceleration capacities greater than 1.0g and with a high confidence of a low frequency (HCLF) of failure greater than 0.4g are not expected to contribute significantly to the overall plant risk.

The conservative fragility descriptions within Table 5-4 were run through the plant system models. The results of these preliminary runs determined the dominant accident sequences and the corresponding components which were critical to core damage and risk. PL&G identified the following components as potential major risk contributors and requested a more detailed evaluation where possible.

1. Offsite Power System
2. Reactor River Pumps
3. CRDM and Assemblies
4. RPV Internals
5. Nuclear Service River Water Pumps
6. Borated Water Storage Tank
7. Control Building
8. Diesel Generator Fuel Oil Day Tank

All of these items were reanalyzed using additional information except for the offsite power system. The offsite power system fragility is based on actual earthquake experience data and the values given in Table 5-4 are judged to be appropriate for the TMI PRA. The control building and the BWST are addressed within Chapter 4 of this report. The remaining critical equipment have their derived actual fragilities listed in Table 5-7. All of the median ground acceleration capacities increased when the plant-specific information was utilized, which was expected since the initial values were conservatively low. A brief description of the fragility derivation for each of these components is documented below.

5.3.1 Vertical Long-Column River Water Pumps

Three river water pumps were analyzed using TMI-1 plant-specific information.

1. Reactor River Pumps (RBEC System)
2. Nuclear Service River Water Pumps (NSR System)
3. Decay Heat River Pumps (DHR and CC System)

Figure 5-2 shows the pump motor assembly for one of these long-column pumps. The motor is securely anchored to the pumphouse operating floor. Figure 5-3 shows the vertical long-column as it penetrates the operating floor and extends down into the river.

All three of these pumps were designed and manufactured by Peerless Pumps. These pumps have unsupported-cantilevered columns extending 40-feet or more down into the river water. This large column length produces natural frequencies in the 1-3 Hz range which is well below the amplified acceleration portion of the spectrum. These pumps develop relatively large stresses in the pump column and in the top flange anchor bolts due to seismic inputs. Thus, the resulting ground acceleration capacities (0.58g to 1.16 g's) are lower than those for other types of pumps typically seen in nuclear plants. Section 5.2.2 contains a specific derivation example for the nuclear service water pump, and the remaining two pump fragility derivations are similar.

5.3.2 Reactor Internals and CRDM

Seismic design analyses for both of these components could not be located at Babcock and Wilcox. The responsible engineers at B&W stated that Midland qualification information should be utilized for the PRA study since the internals and the CRDM for these two plants are essentially identical. Thus, Midland stress summary information was utilized on the CRDM fragility (see example calculation, Section 5.2.3) and on all of the reactor internals except for the fuel rods. The Midland fuel rod analysis was not available, but the Oconee fuel rods were stated to be of the same design. Thus, the fuel rod fragility derived in the Oconee PRA was used in this analysis after correcting for differences in the response amplifications which occur at each of the sites. An additional uncertainty was included in the fragility to reflect the fact that the Oconee derived fragility might not be identical to that of TMI.

5.3.3 Diesel Generator Fuel Oil Day Tank

Figure 5-4 shows the support arrangement for the diesel day tank. This tank had no seismic design and contained no anchorage between the concrete saddles and the tank. Past earthquake experience has shown that unanchored tanks do not fare well during large earthquakes. In addition, the piping attched to this tank has threaded joints which are also considered to have low capacities during the earthquake. The saddle itself was stated to have steel reinforcing by GPU personnel and should have adequate capacity.

An analysis for tank sliding was conducted based on the manufacturers drawings. The lower bound on the tank capacity was assumed to be the onset of sliding. The median fragility was calculated for a sliding-type failure utilizing an analysis technique derived by N. M. Newmark (Reference 24). Soil-structure interaction effects were accounted for as described in Chapter 4 for the Diesel Building foundation.

A median ground acceleration capacity of 0.6 g's is calculated for the diesel day tank as shown in Table 5-7. The median ground acceleration capacity of the day tank could be increased by retrofitting the unanchored tank and properly analyzing the threaded piping.

TABLE 5-1

EARTHQUAKE COMPONENT COMBINATION FACTORS

ABSOLUTE SUM OF (H_1 + V)

Case	Description	$\check{F}_{ECC}$	β_R	β_U
1	3D Case - All 3 directional components contribute to failure	1.09	0.12	0.01
2	2D Case - Median Coupling - Both horizontal contribute to failure	1.00	0.10	0.04
3	2D Case - No Coupling - Both horizontals contribute to failure	0.95	0.06	0.0
4	2D Case - Median Coupling - 1 horizontal and the vertical contribute to failure	1.12	0.09	0.03
5	2D Case - No Coupling - 1 horizontal and the vertical contribute to failure	1.17	0.03	0.0
6	1D Case - Any one of the directional components alone is responsible for the failure	1.0	0.0	0.0
7	Systems of components for which any one of the above cases could apply (piping, cable trays, ducting, etc.)	1.05	0.12	0.07

TABLE 5-2

STRUCTURAL RESPONSE FACTORS FOR EQUIPMENT

BUILDING	GROUND ACCELERATION RANGE (g)*	F_{SR}	β_{SR_R}	β_{SR_U}
Control Building	a) <0.59 b) $\geq$0.59	1.81 2.11	0.21 0.17	0.24 0.22
Auxiliary Building	<1.0	1.68	0.15	0.24
Reactor Building	<1.0	1.40	0.27	0.25
Containment Internal Structure	<1.0	1.86	0.18	0.22
Intermediate Building	a) <0.85 b) $\geq$0.85	1.69 1.91	0.13 0.13	0.26 0.26
Diesel Generator Building	<1.0	1.24	0.04	0.31
Intake Screen House	<1.0	1.32	0.05	0.18

* Acceleration value shown reflects yield capacity of structure

TABLE 5-3

CONSERVATIVE MEDIAN CAPACITY DERIVATION FOR EMERGENCY BATTERIES AND RACKS

PLANT	GROUND ACCELERATION CAPACITY	FRAGILITY BASIS [1]
1	1.01 g's	S
2	1.37 g's	G
3	1.56 g's	G
4	1.09 g's	G
5	1.17 g's	G
6	2.29 g's	S + G
7	2.56 g's	S + G
8	>2.0 g's	S
9	1.28 g's	S + G
10	0.95 g's	S
11	5.6 g's	S + G
12	>3.0 g's	S
13	1.74 g's	S
MINIMUM	= 0.95 g's = $\overset{*}{A}$	

[1] S = plant specific data source
G = generic data source
S + G = combination of specific and generic data

TABLE 5-4

CONSERVATIVE FRAGILITIES FOR TMI-1 EQUIPMENT

SYSTEM	COMPONENT (ID)	$\overset{*}{A}^{1,2}$	β_R	β_U
ECCS (Emergency Core Cooling System)	BWST	0.53	0.25	0.44
	HPI Makeup Pumps	>1g	--	--
	Isolation Valves	>1g	--	--
	LPI/DHR Pumps	>1g	--	--
	DHR Heat Exchangers (Decay Heat Closed Cooling Heat Exchangers)	0.75g's	0.25	0.31
	Isolation Valves	>1g	--	--
	Dropline Valves	>1g	--	--
	Piggyback Valves	>1g	--	--
	Reactor Building Sump	>1g	--	--
	Isolation Valves	>1g	--	--
Reactor Bldg Spray	RB Spray Pumps	>1.0g's	--	--
	Spray Header & Nozzles	>1.0g's	--	--
	Motor Operated Valves	>1.0g's	--	--
Reactor Bldg Emergency Cooling System	Reactor River Pumps	0.36	0.25	0.17
	Cooling Coils	0.9g's	0.25	0.42
	Isolation Valves	>1.0g's	--	--
	Fans and Motors	>1.0g's	--	--
Emergency Feedwater System	Motor Driven Pumps	>1.0g's	--	--
	Turbine Driven Pumps	>1.0g's	--	--
	Flow Control Valves	>1.0g's	--	--
	Block Valves (MOV's)	>1.0g's	--	--

TABLE 5-4 (cont.)

SYSTEM	COMPONENT (ID)	$\overset{*}{A}^{1,2}$	β_R	β_U
ESAS (Engineered Safeguards Actuation System)	1) Sensors	0.88g's	0.25	0.40
	2) Actuation Cabinets A & B	0.4/0.8	0.25/0.25	0.48/0.34
	3) Engineered Safeguards Relay Cabinets	0.4/0.8	0.25/0.25	0.48/0.34
	4) Bistable Cabinets	0.4/0.8	0.25/0.25	0.48/0.34
Reactor Protection System	CRDM's & assemblies	0.66g's	0.25	0.34
Electric Power A. AC Power	1) 4160 Switchgear	0.4/0.8	0.25/0.25	0.48/0.34
	2) 4160/480 Transformer	0.73	0.25	0.29
	3) 480 V Switchgear	0.4/0.8	0.25/0.25	0.48/0.34
	4) 480 V MCC	0.4/0.8	0.25/0.25	0.48/0.34
B. DC Power	1) Batteries	0.95g's	0.25	0.56
	2) Chargers	0.49g's	0.25	0.60
	3) Inverters	0.49g's	0.25	0.60
	4) DC Distribution Panels 1A & 1B	0.28g's(4)	0.25	0.26
	5) DC Subpanels 1E,1C,1H,1D, 1F and 1J	0.28g's(4)	0.25	0.26
	6) Vital AC Instrument Buses VBA/B/C/D, ATA/B, TRA, PRB	0.4/0.8	0.25/0.25	0.48/0.34
	7) 120 V Transformers	0.73g's	0.25	0.29
C. Offsite Power	Ceramic Insulators, etc.	0.3g's	0.25	0.50

TABLE 5-4 (cont.)

SYSTEM	COMPONENT (ID)	$\overset{*}{\hat{A}}{}^{1,2}$	β_R	β_U
D. Emergency Power	Diesel Generators (Everything on the skid)	0.75g's	0.25	0.44
	Air Receiver Tank	0.68g's	0.25	0.25
	Fuel Oil Transfer Pump	>1.0g's	--	--
	Air Start Compressor	>1.0g's	--	--
	Batteries for Air Start Comp.	0.3g's	0.25	0.31
	DG Control/Breaker Panel	Functional 0.37	0.25	0.42
		Structural >1.0	--	--
	Fuel Oil Day Tank	0.3g's	0.25	0.31
RCS (Reactor Coolant System)	Reactor Pressure Vessel	>1.0g's	--	--
	Reactor Coolant Pumps	>1.0g's	--	--
	Pressurizer	>1.0g's	--	--
	Steam Generator	>1.0g's	--	--
	RPV Internals	0.49g's	0.25	0.18
	Pressurizer Safety Valves	>1.0g's	--	--
	PORV	>1.0g's	--	--
	R.C. Drain Tank	0.7g's	0.25	0.40
	Aux. Spray Line	>1.0g's	--	--
CBVS (Control Bldg. Vent. System)	Normal Supply Fans	>1.0g's	--	--
	Emergency Supply Fans	>1.0g's	--	--
	Chilled Water Supply Pumps	>1.0g's	--	--
	Air Operated Dampers	>1.0g's	--	--
	Booster Fans	>1.0g's	--	--
	Return Fans	>1.0g's	--	--

TABLE 5-4 (cont.)

SYSTEM	COMPONENT (ID)	$\overset{*}{A}$ [1,2]	β_R	β_U
NSR & CCWS[1] (Nuclear Service River & Closed Cooling Water Systems)	1) NS River Water Pumps	0.36 g's	0.25	0.17
	2) NS Heat Exchangers	0.75 g's	0.25	0.31
	3) Inter. Closed Cooling Water Heat Exchanger	0.75 g's	0.25	0.31
	4) Nuclear Service Cooling Water Pumps	> 1.0 g's	NA	NA
	5) Nuclear Service Surge Tank	0.7 g's	0.25	0.40
	6) Supply & Return Isolation Valves	> 1.0 g's	--	--
Decay Heat River & Closed Cooling Water Systems	1) Decay Heat River Pumps	0.36 g's	0.25	0.17
	2) Decay Heat Service Heat Exchanger	0.75 g's	0.25	0.31
	3) D. H. Closed Cooling Water Pumps	> 1.0 g's	NA	NA
	4) D. H. Removal Heat Exchangers Closed Cooling	0.75 g's	0.25	0.31
	5) D. H. Surge Tanks	0.7 g's	0.25	0.40
	6) Supply & Return Isolation Valves	> 1.0 g's	--	--
Main Steam System	M.S. Safety Valves	> 1.0 g's	--	--
	Atmos. Dump Valves	> 1.0 g's	--	--
	Turbine Bypass Valves	> 1.0 g's	--	--
	MSIV	> 1.0 g's	--	--
	Main Steam Lines	> 1.0 g's	--	--
	Turbine Stop Valves	> 1.0 g's	--	--
	Turbine Control Valves	> 1.0 g's	--	--

TABLE 5-4 (cont.)

SYSTEM	COMPONENT (ID)	$\overset{*}{A}^{1,2}$	β_R	β_U
Containment Isolation System	1) Containment Purge Valves	> 1.0 g's	--	--
	2) Letdown Isolation Valves	> 1.0 g's	--	--
	3) RCP Seal Isolation Valves	> 1.0 g's	--	--
Air Systems	1) Air Bottles (2-hr Emergency)	> 1.0 g's	--	--
	2) Regulating Valves	> 1.0 g's	--	--
	3) Piping	> 1.0 g's	--	--
Intermediate Closed Cooling Water System	1) Inter. Cooling Pumps	> 1.0 g's	--	--
	2) Surge Tanks	0.7 g's	0.25	0.40
	3) Intermediate Coolers	0.75 g's	0.25	0.31
	4) Isolation Valves	> 1.0 g's	--	--

NOTES FOR TABLE 5-4

1. $\overset{*}{\breve{A}}$ is a conservative median capacity level which has been derived from the results of past SMA PRAs.

2. Fragilities labeled "$\geq$ 1.0 g's" are not expected to influence the risk, based on PL&G's assessment of the hazard curves. These components or structures have a high confidence (95%) of a low probability of failure (5%) at 0.4 g's or greater.

3. Electrical components may have two values given in the attached table, i.e., "a/b". These two values "a" and "b" represent recoverable (chatter and trip) and non-recoverable failures, respectively.

4. The values given for the DC Distribution Panels and the DC Subpanels are recoverable failure levels. The non-recoverable levels are > 1.0 g.

TABLE 5-5

PAST PRA DATA ON AIR RECEIVER TANKS

PLANT NUMBER	$\overset{*}{A}$	β_R	β_U	SPECIFIC OR GENERIC
1	>2 g's	NA	NA	G
2	1.17g's	0.19	0.49	G
3	>2 g's	NA	NA	S
4	>2 g's	NA	NA	S
5	0.68g's	0.29	0.40	G
6	>2 g's	NA	NA	S
Minimum = 0.68g's = $\overset{*}{A}$				

TABLE 5-6

FRAGILITY DERIVATION OF NUCLEAR SERVICE RIVER WATER PUMP

	MEDIAN SAFETY FACTOR	RANDOM VARIABILITY β_R	UNCERTAINTY VARIABILITY β_U
Capacity Factor (F_{EC})			
1. Strength Factor	2.28	0.0	0.22
2. Ductility Factor	1.0	0.0	0.0
Combined ⟶ F_{EC}	2.28	0.0	0.22
Equipment Response Factor (F_{ER})			
1. Qualification Method	2.63	0.0	0.00
2. Spectral Shape	1.0	0.0	0.00
3. Damping	1.0	0.02	0.12
4. Modeling	1.0	0.0	0.15
5. Mode Combination	1.0	0.10	0.0
6. Multi-Directional Effects	0.95	0.06	0.0
Combined ⟶ F_{ER}	2.50	0.12	0.19
Structural Response Factor (F_{SR})	1.0	0.37	0.26
Ground Acceleration Capacity (A)	0.68g's	0.39	0.39

TMI-1 PLANT SPE

SYSTEM & COMPONENTS	DESCRIPTION	LOCATION	EQUIPMENT CHARACTERISTIC	SEISMIC QUALIFICATION METHOD	FAILURE M
Reactor Bldg. Emergency Cooling System					
1. Reactor River Pumps	RR-P-1A/B Vertical 40' Unsupported Column	IS & PH 308'	Active, 0.9 hz	Response Spectrum	Top Column Fl
Reactor Protection System					
1. CRDM and Assemblies	Similar to Midland's.	RB	Active, Flexible	Similarity to Midland's	Functional
Offsite Power System	Power lines, Transformer, etc	Outside	Passive, Flexible	None	Ceramic Insul
Emergency Power System					
1. Fuel Oil Day Tank	Cylindrical Horizontal Tank Mounted on Saddles	DG Bldg. 305'	Rigid, Passive	None	Slidng
Reactor Coolant System					
1. RPV Internals	1. Fuel Rods 2. All Other Internals	RB (RPV) RB (RPV)	Active, Flexible Passive, Flexible	Similarity to Oconee Similarity to Midlands	Deflection, B Upper Grid Jo
Nuclear Service River System					
1. NS River Water Pumps	NR-P-1A/B/C Vertical 43' Unsupported Column	IS & PH 308'	Active, 1.1 hz	Response Spectrum	Top Column Fl
Decay Heat River & Closed Cooling System					
1. Decay Heat River Pumps	DR-P-1A/B Vertical Pump 43' Unsupported Column	IS & PH 308'	Active, 1.2 hz	Response Spectrum	Bending of Co

1) CRDM fragility was based on seismic qualifcation of the Midlands Nuclear Plant CRDM.

2) These fragilities have proven to have median capacities greater than 1.0 g's with a "high confidence of a low frequency
0.4 g's based on TMI specific information. These fragility levels have been determined not to significantly affect the
fragility parameters have not been derived.

3) These factors for the fuel rod are based on the Oconee SSE ground response spectrum, ie 0.10 g as opposed to the 0.12

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TABLE 5-7

CIFIC FRAGILITY DESCRIPTIONS

DE	INFORMATION SOURCE	Equipment Response Factors			Equipment Capacity Factors			Structural Response Factors			Ground Acceleration Capacities				NOTES
		F_{ER}	β_R	β_U	F_{EC}	β_R	β_U	F_{SR}	β_R	β_U	A (g's)	β_R	β_U	β_C	
nge Bolts	Peerless Qual (SMA-M083)	2.85	0.12	0.18	1.74	0.0	0.22	1.0	0.37	0.26	0.50	0.39	0.39	0.55	
	Midland FSAR and Calcs.	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	1.0g's	N/A	N/A	N/A	1,2
tors	Past Earthquake Experience	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	0.3g's	0.25	0.50	0.56	
	SMA Analysis	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	0.6g's	0.25	0.42	0.49	
nding nt	Oconee FSAR Midlands Stress Summary	3.05 N/A	0.24 N/A	0.32 N/A	4.0 N/A	0.0 N/A	0.36 N/A	0.71 N/A	0.16 N/A	0.10 N/A	0.86 1.0g's	0.29 N/A	0.50 N/A	0.58 N/A	3 2
nge Bolts	Peerless Pump Qual(M-083)	2.50	0.12	0.19	2.28	0.0	0.22	1.0	0.37	0.26	0.68	0.39	0.39	0.55	
umn	Peerless Pump Qual (M-083)	2.52	0.12	0.19	3.05	0.0	0.21	1.0	0.37	0.26	1.16	0.39	0.39	0.55	

of failure" of greater than
risk and, thus, their complete

g TMI SSE ground spectra.

8806210079-01

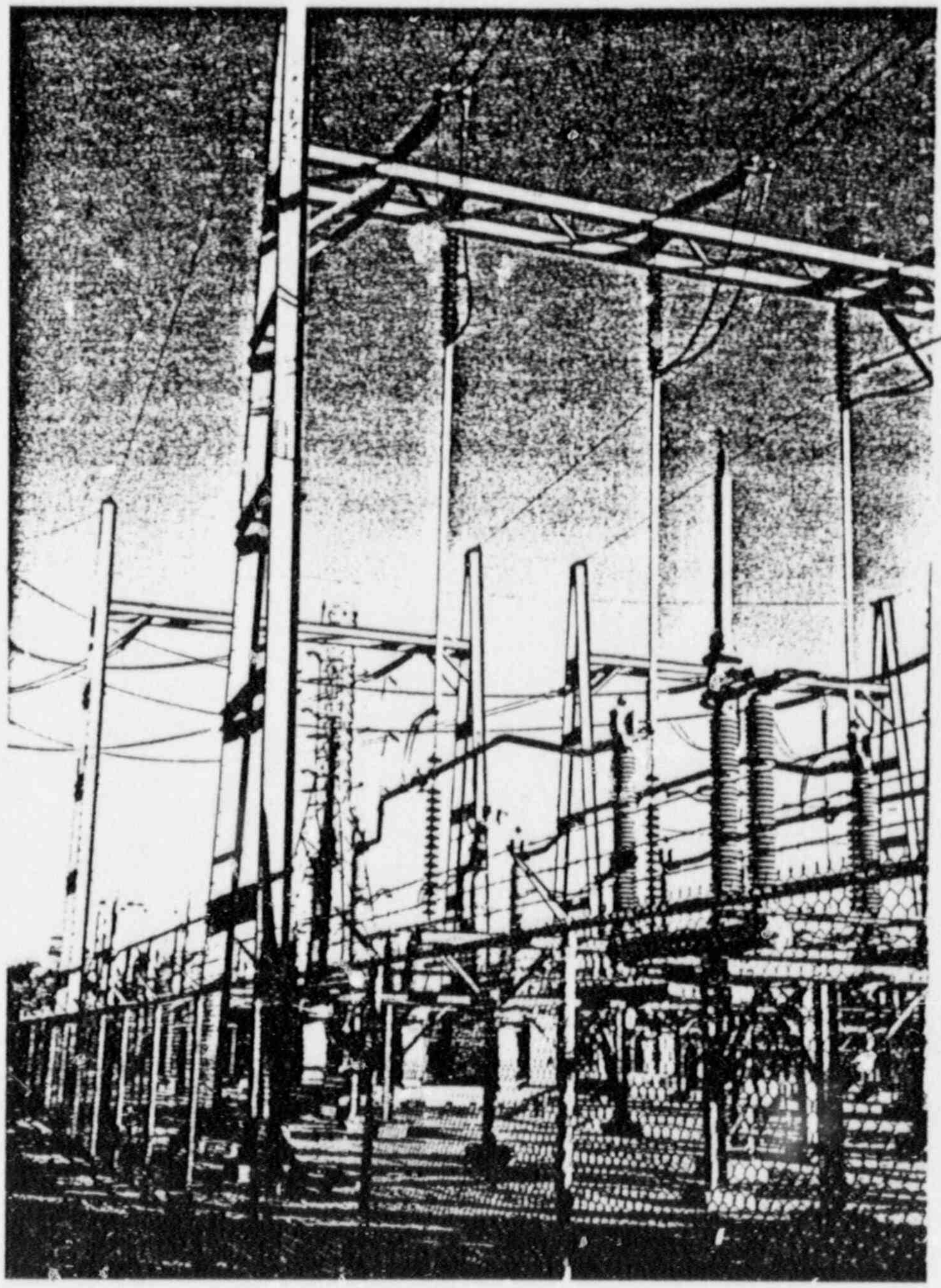

FIGURE 5-1: OFFSITE POWER TRANSMISSION STRUCTURES

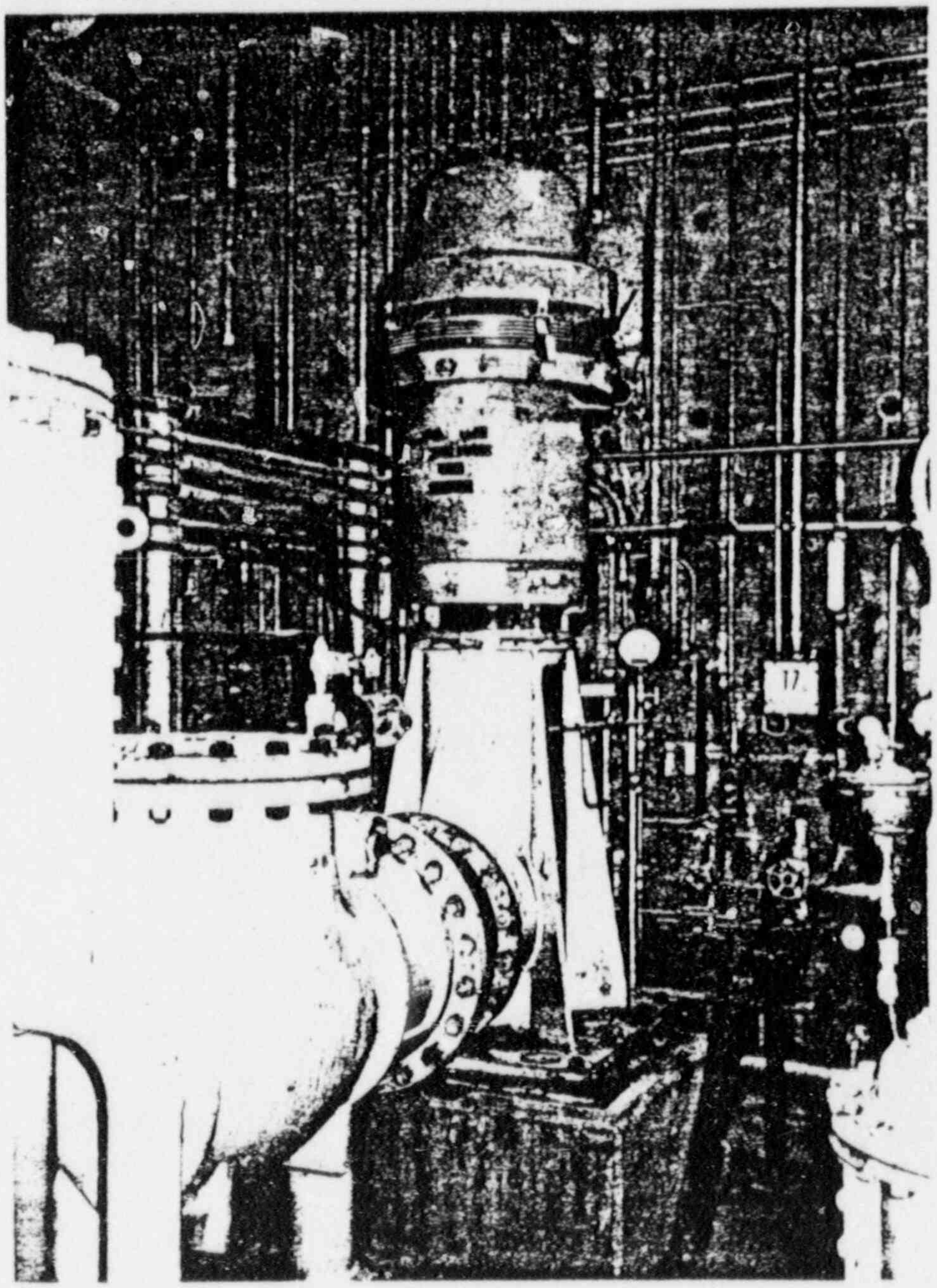

FIGURE 5-2: PUMP MOTOR FOR VERTICAL RIVER WATER PUMPS

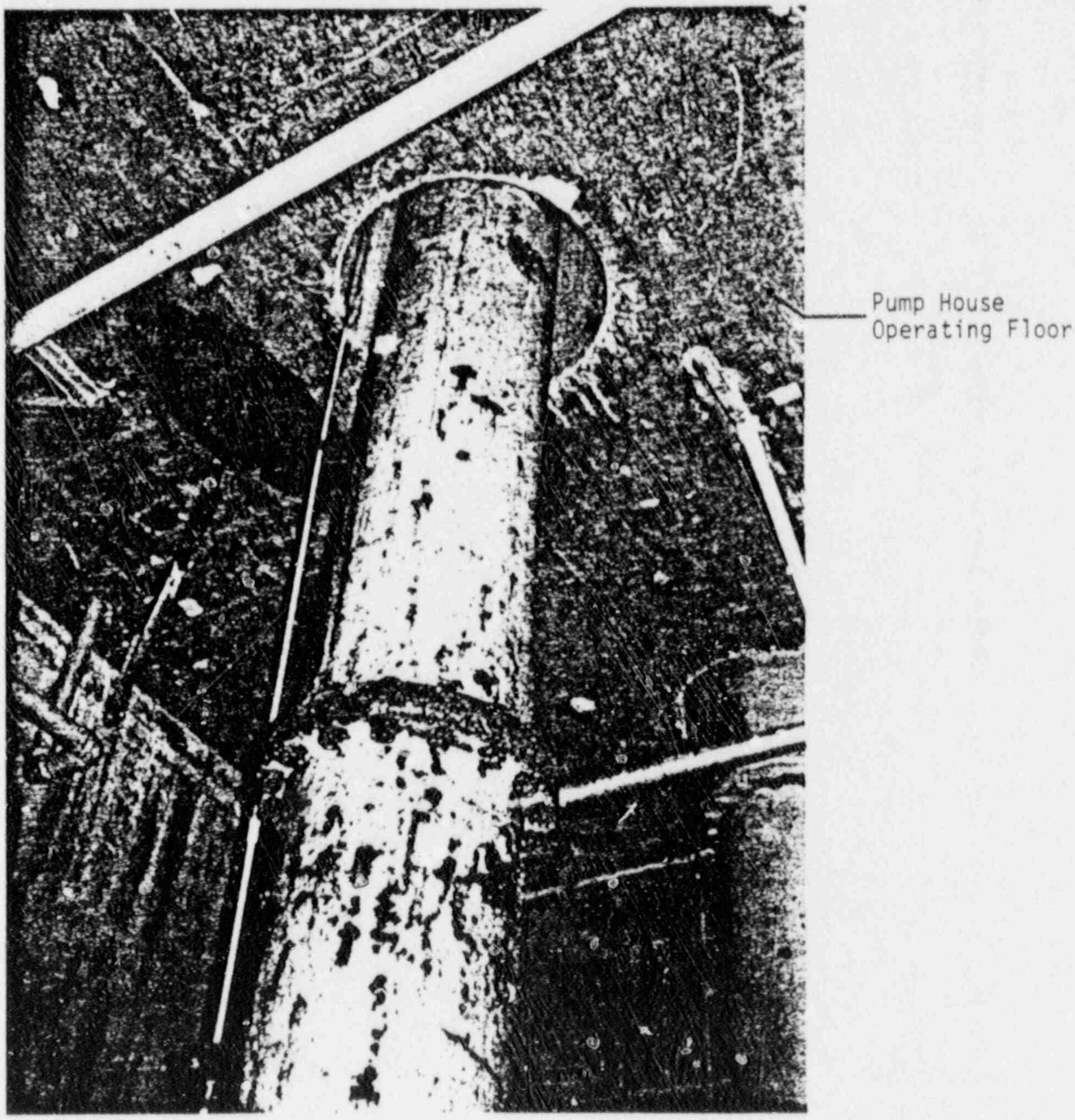

FIGURE 5-3: PUMP COLUMN FOR VERTICAL RIVER WATER PUMPS

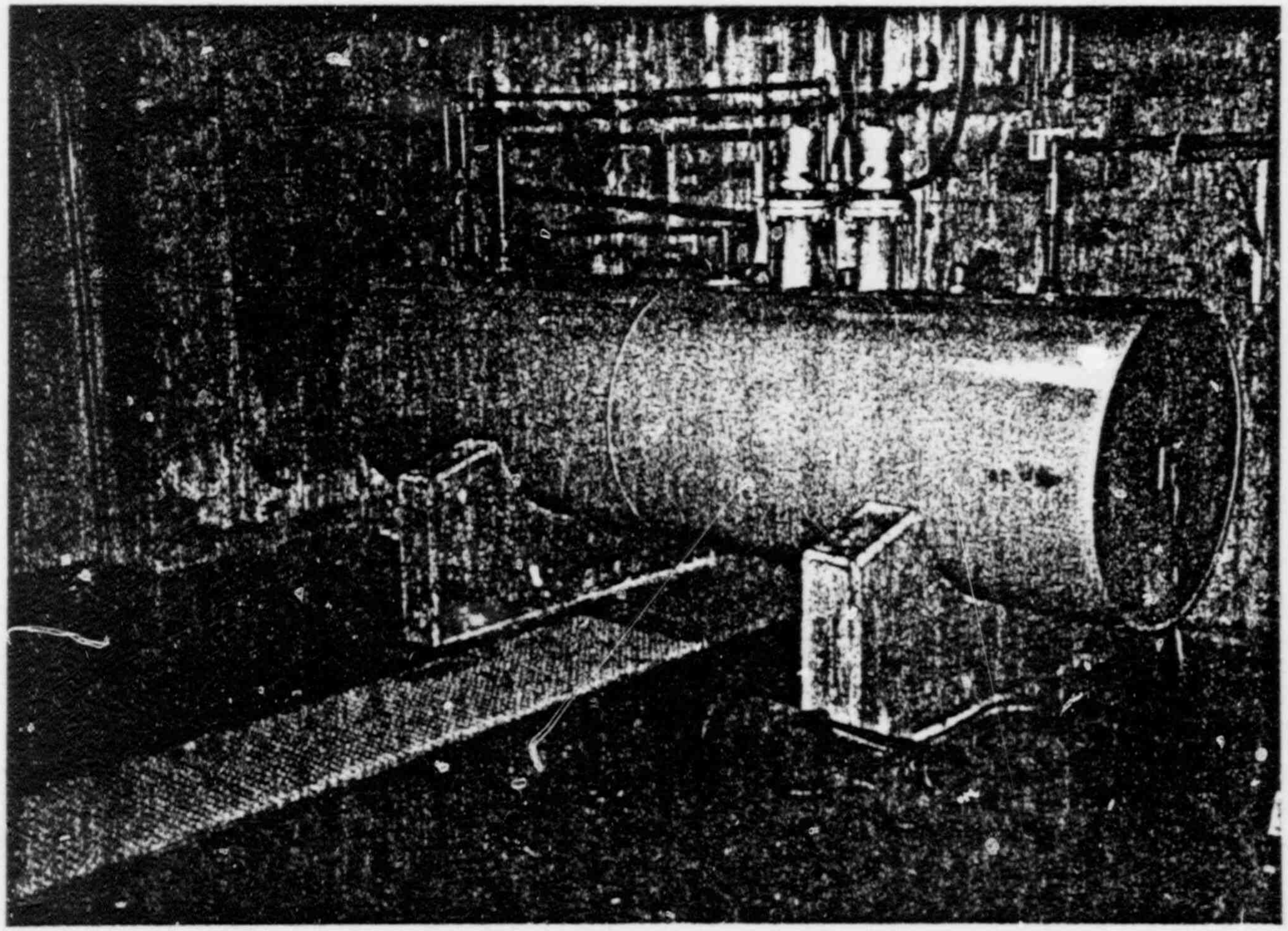

FIGURE 5-4: DIESEL FUEL OIL DAY TANK

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APPENDIX A

CHARACTERISTICS OF THE LOGNORMAL DISTRIBUTION

APPENDIX A

CHARACTERISTICS OF THE LOGNORMAL DISTRIBUTION

Some of the characteristics of the lognormal distribution which are useful to keep in mind when generating estimates of $\check{A}$, β_R, and β_U are summarized in References A1 and A2. A random variable X is said to be lognormally distributed if its natural logarithm Y given by:

$$Y = \ell n\ (X) \qquad \text{(A-1)}$$

is normally distributed with the mean of Y equal to $\ell n\ \check{X}$ where $\check{X}$ is the median of X, and with the standard deviation of Y equal to β, which will be defined herein as the logarithmic standard deviation of X. Then, the coefficient of variation, COV, is given by the relationship:

$$\text{COV} = \sqrt{\exp(\beta^2)-1} \qquad \text{(A-2)}$$

For β values less than about 0.5, this equation becomes approximately:

$$\text{COV} \approx \beta \qquad \text{(A-3)}$$

and COV and β are often used interchangeably.

For a lognormal distribution, the median value is used as the characteristic parameter of central tendency (50 percent of the values are above the median value and 50 percent are below the median value). The logarithmic standard deviation, β, or the coefficient of variation, COV, is used as a measure of the dispersion of the distribution.

The relationship between the median value, $\check{X}$, logarithmic standard deviation, β, and any value x of the random variable can be expressed as:

$$x = \check{X} \bullet \exp\ (\eta \bullet \beta) \tag{A-4}$$

where η is the standardized Gaussian random variable, (mean zero, standard deviation one). Therefore, the frequency that X is less than any value x' equals the frequency that η is less than η' where:

$$\eta' = \frac{\ell n\ (x'/\check{X})}{\beta} \tag{A-5}$$

Because η is a standardized Gaussian random variable, one can simply enter standardized Gaussian tables to find the frequency that η is less than η' which equals the probability that X is less than x'. Using cumulative distribution tables for the standardized Gaussian random variable, it can be shown that $\check{X} \bullet \exp\ (+\beta)$ of a lognormal distribution corresponds to the 84 percentile value (i.e., 84 percent of the data fall below the $+\beta$ value). The $\check{X} \bullet \exp\ (-\beta)$ value corresponds to the value for which 16 percent of the data fall below.

One implication of the usage of the lognormal distribution is that if A, B, and C are independent lognormally distributed random variables, and if

$$D = \frac{A^r \bullet B^s}{C^t}\ q \tag{A-6}$$

where q, r, s and t are given constants, then D is also a lognormally distributed random variable. Further, the median value of D, denoted by $\check{D}$, and the logarithmic variance β_D^2, which is the square of the logarithmic standard deviation, β_D, of D, are given by:

$$\check{D} = \frac{\check{A}^r \bullet \check{B}^s}{\check{C}^t} q \tag{A-7}$$

and

$$\beta_D^2 = r^2 \beta_A^2 + s^2 \beta_B^2 + t^2 \beta_C^2 \tag{A-8}$$

where $\check{A}$, $\check{B}$, and $\check{C}$ are the median values, and β_A, β_B, and β_C are the logarithmic standard deviations of A, B, and C, respectively.

The formulation for fragility curves given by Equation 2-1 and shown in Figure 2-1 and the use of the lognormal distribution enables easy development and expression of these curves and their uncertainty. However, expression of uncertainty as shown in Figure 2-1 in which a range of peak accelerations are presented for a given failure fraction is not very usable in the systems analyses for frequency of radioactive release. For the systems analyses, it is preferable to express uncertainty in terms of a range of failure fractions (frequencies of failure) for a given ground acceleration. Conversion from the one description of uncertainty to the other is easily accomplished as illustrated in Figure A-1 and summarized below.

With perfect knowledge (i.e., only accounting for the random variablity, β_A), the failure fraction, f(a), for a given acceleration a can be obtained from:

$$f(a) = \Phi\left(\frac{\ln(a/\check{A})}{\beta_R}\right) \tag{A-9}$$

in which $\Phi(\bullet)$ is the standard Gaussian cumulative distribution function, and β_R is the logarithmic standard deviation associated with the underlying randomness of the capacity.

For simplicity, denote $f = f(a)$. Similarly, f' is the failure fraction associated with acceleration a', etc. Then, with perfect knowledge (no uncertainty in the failure fractions), the ground acceleration a' corresponding to a given frequency of failure f' is given by:

$$a' = \check{A} \exp\left[\beta_R \, \Phi^{-1}(f')\right] \tag{A-10}$$

The uncertainty in ground acceleration capacity corresponding to a given frequency of failure as a result of uncertainty of the median capacity can then be expressed by the following probability statement:

$$P\left[A > a'' \middle| f'\right] = 1 - \Phi\left[\frac{\ln(a''/A)}{\beta_U}\right] \tag{A-11}$$

in which $P[A > a'' | f']$ represents the probability that the ground acceleration A exceeds a'' for a given failure fraction f'. This probability is shown shaded in Figure A-1. However, it is desirable to transform this probability statement into a statement on the probability that the failure fraction f is less than f' for a given ground acceleration a'', or in symbols $P[f \le f' | a'']$. This probability is also shown shaded in Figure A-1. It follows that:

$$P[f \le f' | a''] = P[A > a'' | f'] \tag{A-12}$$

Thus, from Equations A-10 and A-11:

$$P[f \le f' | a''] = 1 - \Phi\left[\frac{\ln\left(a''/\check{A} \exp\left[\beta_R \, \Phi^{-1}(f')\right]\right)}{\beta_U}\right] \tag{A-13}$$

from which:

$$P\left[f > f' \middle| a''\right] = \Phi\left(\frac{\ell n\left(a''/\check{A}\ \exp\left[\beta_R\ \Phi^{-1}\ (f')\right]\right)}{\beta_U}\right) \qquad \text{(A-14)}$$

which is the basic statement expressing the probability that the failure fraction exceeds f' for a ground acceleration a" given the median ground acceleration capacity $\check{A}$, and the logarithmic standard deviations β_R and β_U associated with randomness and uncertainty, respectively.

As an example, if:

$\check{A} = 0.77$, $\beta_R = 0.36$, $\beta_U = 0.39$

then from Equation A-14 for typical values of f and a",

$$P\left[f > 0.5 \middle| a'' = 0.40g\right] = 0.05$$

which says that there is a 5 percent probability that the failure frequency exceeds 0.5 for a ground acceleration of 0.40g.

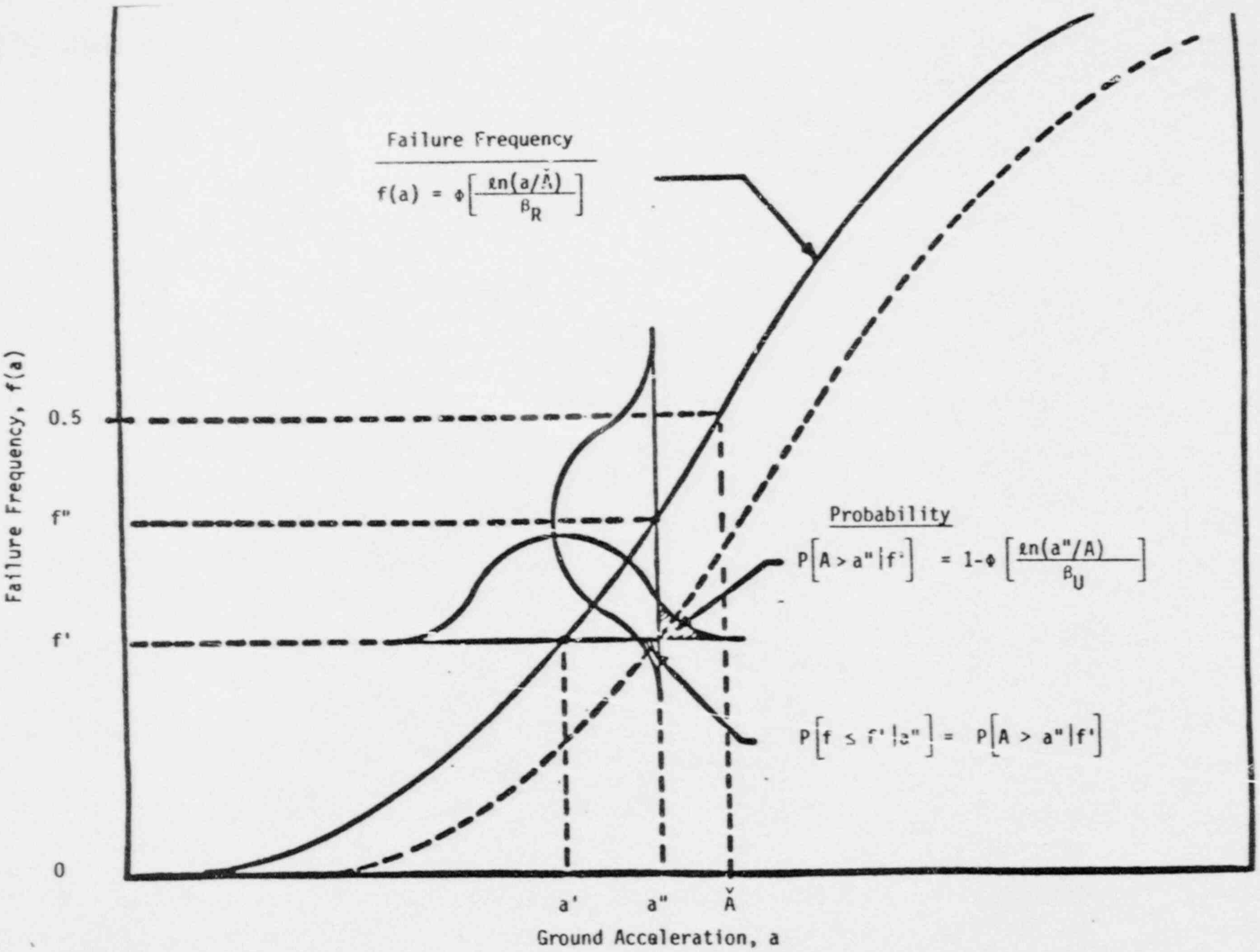

FIGURE A-1. RELATIONSHIP BETWEEN UNCERTAINTY IN GROUND ACCELERATION FOR A GIVEN FAILURE FRACTION AND UNCERTAINTY IN FAILURE FRACTION FOR A GIVEN GROUND ACCELERATION

REFERENCES

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CONTENTS

APPENDIX C

SPATIAL INTERACTION TABLES

GENERAL DESCRIPTION

The tables that are put together as part of the spatial interaction analysis are presented in this appendix. The tables are grouped by buildings and organized in ascending order by their fire zone (or area) number. For each location in the plant, three tables are presented: (1) location inventory codification table, (2) source and mitigation table, and (3) scenario table. For some scenarios that are judged to be important, an impact table is put together.

These tables and the sources of information supporting them are described in Section 3 of this report. Table C-1 describes the abbreviated system designators used in the location inventory and impact tables. In the location tables, the symbol "X" is used to minimize entering the same information several times. For example, if a valve appears in a fire zone the valve name is given under the column "valve" and the associated power and control cables are shown by "X." The references are given in several ways; they are described in the following section.

All the tables have gone through several rounds of iteration. In these iterations more attention has been spent on the important scenarios. These are the scenarios that are suspected of having a significant impact on plant risk. Therefore, they are presented in greater detail than the other ones. Similarly, there are some variations in the level of detail incorporated into these tables.

REFERENCES

In the location inventory and source tables the sources of information are referenced using the following symbols:

- "1" refers to the Fire Hazards Analysis Report and Appendix R, Section III.G, Safe Shutdown Evaluation of TMI-1; Section 3.11, List of Equipment Required for Safe Shutdown.
- "2" refers to the Fire Hazards Analysis Report and Appendix R, Section III.G, Safe Shutdown Evaluation of TMI-1; Section 3.10, Valves Required for Safe Shutdown.
- "4192-C-302-nnn" are P&ID designators.
- "Fire Hazards Report" refers to the Fire Hazards Analysis Report and Appendix R, Section III.G, Safe Shutdown Evaluation of TMI-1.
- "1-FHA-Onn" are layout drawings of the Fire Hazards Analysis Report.
- "T3.11-nn" are Tables 3.11-15 through 3.11-31 of the Fire Hazards Analysis Report.

- "T3.10-nn" are Tables 3.10-1 through 3.10-6 of the Fire Hazards Analysis Report.
- "FHA" is the same as the Fire Hazards Analysis Report.
- "Plant Visit" refers to information obtained from onsite inspection of the location by the analyst.
- "E-nnn-nnn" are piping isometric drawings.
- "C. Adams Letter, 6/19/84" refers to a letter from Charles Adams of GPU to D. L. Acey of Pickard, Lowe and Garrick, Inc., dated June 19, 1984.
- "C. Husted" refers to personal communications with C. Husted of GPU and J. K. Liming of Pickard, Lowe and Garrick, Inc.
- "Color Coded Drawings" refers to cable tray and conduit drawings color coded by T. O'Connor of GPU.
- "5/31" refers to tables in Attachments 3.5 and 3.6 of Fire Hazards Analysis Report transmitted to Pickard, Lowe and Garrick, Inc., May 31, 1985.

TABLE C-1. ABBREVIATED SYSTEM DESIGNATORS

Abbreviation	Definition
AN	air handling systems throughout the plant
BWST	borated water storage tank
CF	core flooding tank
CST	condensate storage tank
DC	decay heat closed cooling water
DH	decay heat removal system
DR	decay heat river water
EF	emergency feedwater
ES	electric power system
FH	fuel handling related systems
FW	main feedwater
IC	intermediate closed cooling system
MCC	motor control center
MOV	motor-operated valve
MSIV	main steam isolation valve
MU	make up and purification system
NS	nuclear services closed cooling water
NR	nuclear services river water
NI	nuclear instrumentation
PORV	power-operated relief valve
PSV	pressurizer safety valve
RR	river water for reactor building emergency cooling
RS	reactor building spray system
RCP	reactor coolant pump
RC	reactor coolant system
RPS	reactor protection system
TBV	turbine bypass valve

AUXILIARY BUILDING

Location Name: Heat Exchanger Vault
Designator: AB-FZ-1
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NR			NR-V5		X	X			2, 4192 C-302-202	Header valve (MOV powered from 1A-ESV control center).
			NR-V6						2, 5/31 4192 C-302-202	
			NR-V18		X	X			2, 4192 C-302-202	Exhaust valve for river water flow.
			NR-V8A		X	X			2, 4192 C-302-202	
			NR-V8B		X	X			2, 4192 C-302-202	
			NR-V8C		X	X			2, 4192 C-302-202	Heat exchanger mist valves.
			NR-V8D		X	X			2, 4192 C-302-202	
			NR-V16A		X	X			2, 4192 C-302-202	
			NR-V16B		X	X			2, 4192 C-302-202	
			NR-V16C		X	X			2, 4192 C-302-202	Heat exchanger outlet valves.
			NR-V16D		X	X			2, 4192 C-302-202	
			NR-V19		X	X			C. Adams Letter, 6/19/84	
NR			NR-V-10A NR-V-10B NR-V-15A NR-V-15B		X X	X X X NR-V-18 X			5-31 5-31 5-31 5-31 5-31	
NR			NR-V4A NR-V4B		X X	X X			5-31 5-31	Crosstie to circulating water piping.
NS								Heat Exchangers NS-C-1A NS-C-1B NS-C-1C NS-C-1D		
DC								DC-C-2A DC-C-2B		
IC								IC-C-1A IC-C-1B		
DR	A and B							Piping	4192 C-302-202 Revision 22	
DH			DH-V6A							
DH						DH-V-75A			5-31	SOV for DH pump protection.
						DH-V-76A			5-31	SOV for DH pump protection.

SOURCE AND MITIGATION TABLE

Location Name: Heat Exchanger Vault
Designator: AB-FZ-1
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling (relatively clean room)		Fire Hazards Report	Ionization Fire Detectors Portable Dry Chemical Fire Extinguisher Fire Hose Station Non-Fire Rated Reinforced Concrete Walls	Fire Hazards Report	Two cable trays on top of the walkway of the tunnel.
Flood	Pipe Sections, Heat Exchangers, Fire Hose Station		1-FHA-025 1-FHA-031	Walls and Floor Do Not Have Any Opening except at Northeast Corner toward the Corridor 10 Feet Above the Floor Room Has Sump and Sump Pump at Northeast Corner		All pipe penetrations are sealed completely. All valves are at least 5 feet off the ground. Pipes are at least 1-foot in diameter.
Spray	Piping					Does not have significant impact except for isolated MOV motor failure.
Moderate Energy Line break	Piping					Impact is judged to be minimal and overall effect similar to floods.
Missiles	Transient Pressurized Canisters			Room Boundaries are Sub-terranean		Welding generally is done by using arc-welding methods.

SCENARIO TABLE

Location Name: Heat Exchanger Vault
Designator: AB-FZ-1
Building: Auxiliary Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling or Transient Fuel	1. Cable burning due to an electrical short or transient fuel.	Localized			No, because only cables affected and all valves are in normal operating position and no hot shorts possible.	10^{-3}		Only during deicing are the valves closed and have to be opened, and all three reactor pumps have to run--10^{-3} (fire) x 5 x 10^{-3} (deicing) x 0.1 (conservative unavailability). - 5 x 10^{-7}--not significant.
		2. Engulfing.	Stairs to Pipe Tunnel	AB-FZ-4		No, severe fire would not have any further damage than scenario 1.	10^{-5}		A subset of AB-FZ-4, scenario 1.
Flood	Pipe Sections, Heat Exchangers, and Fire Hose	3. Pipe break from a limited source other than decay coolant or nuclear service.	No Propagation			No, no serious impact on plant safety.	2 x 10^{-5}		For frequency evaluation, it is assumed that a pipe break leading to another pipe break by whipping has conditional frequency of 10^{-3}. No other way a flood can lead to adjacent system failure.
	Pipe Sections and Heat Exchangers	4. Pipe break in nuclear river.	Tunnel	First Floor of Auxiliary and Fuel Buildings		Yes.	10^{-6} (10^{-4} large flood) x (10^{-2} operators not shutting off the source)	NR, NS, and MUPS Lost (comparison)	Flood may also get into AB-FA-1 and AB-FA-2 if the manway covers are left open. This is much less likely and not considered. Nuclear service deprived of cooling medium. MUPS, pump area flooded. No impact on cables. Will take ~ 3 hours to fill the room. Sump level alarms in the control room. Large flood frequency is used.
Missile	Transient Pressurized Bottles (e.g., for welding)	5. Bottle cap breaking and bottle hitting pipes and heat exchangers.	Localized Effect			Yes.	3 x 10^{-5}	No action impact is judged to be limited to one pipe or heat exchanger, and these are covered.	Frequency is product of 3 x 10^{-3} (frequency of bottle in the room) and 10^{-2} (conditional frequency bottle acting as a missile).

SCENARIO TABLE (continued)

Location Name: Heat Exchanger Vault
Designator: AB-FZ-1
Building: Auxiliary Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Flood	Decay Coolant Pipe or Heat Exchanger	6. Decay coolant or heat exchanger break.	Only AB-FZ-1			Yes.	3×10^{-5} (see below)	No action; dominated by other causes of loss of decay coolant.	Only one train of decay coolant lost.
	Pipe or Heat Exchanger	7. Nuclear service pipe or heat exchanger break.	Only AB-FZ-1			Yes.	3×10^{-5} (8×10^{-6} pipe break) x 4 (pipe pieces)	(syst	All of nuclear service lost because headered. Volume of water spilled very small compared to AB-FZ-1 dimensions.*

*3,000 gpm spill will take about 3 hours to fill AB-FZ-1 (~ 600,000 gallons up to top of the stairs).

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Makeup and Purification Pump A
Designator: AB-FZ-2a
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU		MU-P1A			X	X		Piping	1	Makeup pump.
		MU-P2A			X	X			5-31	
		MU-P3A			X				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Makeup and Purification Pump A
Designator: AB-FZ-2a
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		Fire Hazards Report	Hollow Metal Louvered Door in West Wall	Fire Hazards Report	HVAC duct opens in this room and in the MU-P-1C pump room.
	Pump Oil System		1-FHA-026	Ionization Fire Detector		
				North, South, and East Walls 3-Hour Fire Rated	Fire Hazards Report	
				Location AB-FZ-4: Fire Hose Station		
				Location AB-FZ-5: Fire Hose Station		
				Portable Dry Chemical Extinguisher		
Flood	Pipe Section			Walls; Door is Not Watertight and Opens into AB-FZ-4		Door has louvered grill at the head. Flood detector in the floor draining (>13.2 gpm).
High Energy Pipe Break	Pipe Break			Walls		Discharge and suction pipes are short.
Waterjets	Pipe Break			Walls		
Spray	Pipe Break			Walls		
Missiles	Transient Sources Like Pressurized Canisters			Walls		

SCENARIO TABLE

Location Name: Makeup and Purification Pump A
Designator: AB-FZ-2
Building: Auxiliary Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling or Pump Oil System	1. Cable burning due to an electrical short or transient fuel. Engulfing the room.	Door HVAC Duct	AB-FZ-5 AB-FZ-2C		Yes.	10^{-4}	No action; only MUPS affected. HPA-1 is 2.7-3 per demand.	Failures in AB-FZ-5 augmented with this zone, only a small fraction of equipment fails. Important cables in AB-FZ-5 are far from the door to AB-FZ-2.
Flood	Pipe Section	2. Pipe [illegible] during [illegible] operation.	Door Manway Open	AB-FZ-1, AB-FZ-5 Pipe Tunnel Dripping Down to AB-FA-1, AB-FA-2, and FH-FZ-1	Floor drains may be able to carry the spill rate from makeup tank.	Yes.	3 x 10^{-6} (10^{-4} event) x (3 x 10-2 no flood stoppage)	(comparison) Three MUPS pumps inoperable because BWST is opened. DHR and building spray pumps flooded.	Pipe break at the single single drop line of BWST is deemed very unlikely compared to valve and gasket problems. Flood frequency is a small fraction of large flood in auxiliary building because all the equipment that may cause flooding is tested and monitored closely. Operators have plenty of time to stop the spill and save DHR and building spray pumps. (84 + $123M^2$) x (1.2 pump heights) ÷ (3.29-3) ≈ 66,000 gallons to fail DHR and building spray pumps. A small fraction of spill into DHR and building spray pump room. Overall contour of the plant is flat; sloped toward drain holes. Initially makeup tank will contribute. The operator will open BWST suction valve and empty that tank into pump room. Only few inches of water on the floor. Decay heat and building spray pumps unaffected because manhole is covered and leakage is slow.

SCENARIO TABLE (continued)

Location Name: Makeup and Purification Pump A
Designator: AB-FZ-2a
Building: Auxiliary Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Flood	Pipe Section	3. Pipe break during high pressure injection.	Door Manway Open	AB-FZ-5 Pipe Tunnel Dripping Down to AB-FA-1, AB-FA-2, and FH-FZ-1		Yes.	10^{-7} (3×10^{-2} BWST recirculating test) x (1×10^{-4} pipe or other failure) (3×10^{-2} failure to stop the spill)	MUPS lost. DHR lost. Building spray lost. No action; subset of scenario 2.	Contents of the BWST may be emptied.
High Energy Pipe Break	Pipe Break	4. Localized mechanical damage; water escape.	Door Drains Opening	AB-FZ-5 Auxiliary Building Sump AB-FZ-1 AB-FZ-4		Yes.	2×10^{-7} (8×10^{-6} pipe failure) x (3×10^{-2} operator error to shut off)	No action; similar to scenario 2.	Water jet is assumed not failing any walls. Mechanical damage is localized.
		5. Wall failure; water.	Wall Door	AB-FZ-2B AB-FZ-5 Auxiliary Building Sump AB-FZ-1 AB-FZ-4		Yes.	2×10^{-9} (8×10^{-6} pipe failure) x (3×10^{-2} operator) error) x (.01 wall failure)	No action; very unlikely.	Pipe sections are very short. Maximum damage, one wall failure only. Water jet in AB-FZ-5. Loss of one wall integrity; most likely wall is the one next to discharge side of makeup pump.
Sprays and Water Jets	Pipe Break	6. Impact localized.				No, Part of scenario 2.			
Missiles	Transient Sources	7. Transient source.	Localized to the Room			Yes.	3×10^{-5}	No action; only one pump lost and very unlikely.	Only one makeup pump affected.
		8. Transient sources.	Door	AB-FZ-5 AB-FZ-4					

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Makeup and Purification Pump B
Designator: AB-FZ-2b
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU		MU-P1B			X	X		Piping	1	Makeup pump.
		MU-P2B			X	X			5-31	
		MU-P-3B			X				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Makeup and Purification Pump B
Designator: AB-FZ-2b
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
			Same as for AB-FZ-2a.			

SCENARIO TABLE

Location Name: Makeup and Purification Pump B
Designator: AB-FZ-2b
Building: Auxiliary Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation				
			Type	To					
		Same as for AB-FZ-2a.							

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Makeup and Purification Pump C
Designator: AB-FZ-2C
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU		MU-P1C			X	X		Piping	1	Makeup pump.
DH					DH-P-1B				1	
MU		MU-P-2C MU-P-3C			X X	X			5-31 5-31	

SOURCE AND MITIGATION TABLE

Location Name: Makeup and Purification Pump C
Designator: AB-FZ-2c
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
	Same as for AB-FZ-2a except that floods cannot occur during normal operation.					

SCENARIO TABLE

Location Name: Makeup and Purification Pump C
Designator: AB-FZ-2c
Building: Auxiliary Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation				
			Type	To					
		Same as for AB-FZ-2a except during floods. Normal operation is not possible because pump is valved out.							

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Valve Gallery
Designator: AB-FZ-3
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
Above the Concrete Slab at Elevation 295'0"										
MU					MU-P-1A	X		Delay Coil	1	
					MU-P-1B	X			Plant Visit	
					MU-P-1C				Plant Visit	
					MU-P-2A				5-31	
					MU-P-3A				5-31	
					MU-P-2B	X			5-31	
					MU-P-3B				5-31	
					MU-P-3C				5-31	
DH	B				DH-P-1B				5-31	
Below Elevation 295'0"										
MU			MU-V90							
			MU-V37		X	X				
			MU-V36		X	X				
			MU-V64A							Manual valve.
			MU-V64B							Manual valve.
			MU-V64C							Manual valve.
			MU-V77A							Manual valve.
			MU-V77B							Manual valve.
			MU-V76A							Manual valve.
			MU-V77B							Manual valve.
			MU-V12		X	X				
	MU		MU-V4			X				Pneumatic valve, total letdown isolation, fail closed type.
			MU-V32			X				Seal injection valve. Pneumatic stop check valve. Fails at mid-position upon ICS failure.
			MU-V5			X				
			MU-V109A							
			MU-V109B							
			MU-V98							
			MU-V97A							
			MU-V89A							
			MU-V89B							
			MU-V17				X		5-31	
			MU-V18				X		5-31	
BS			BS-V-2A			X			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Valve Gallery
Designator: AB-FZ-3
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		Fire Hazards Report	Reinforced Concrete Walls (three) Ionization Fire Detectors Location AB-FZ-4: Fire Hose Protection	Fire Hazards Report	
Missiles	Transient Source			Walls Opening to AB-FZ-4		
Flood	MUPS Piping			Opening to AB-FZ-5		
High Engergy Line Break	Energized MUPS Piping below Elevation 295'			Walls on Three Sides and Concrete Slab Above		
Waterjet	Energized MUPS Piping below Elevation 295'			Walls on Three Sides and Concrete Slab Above		

SCENARIO TABLE

Location Name: Valve Gallery
Designator: AB-FZ-[illegible]
Building: Auxiliary Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	1. Cable burning due to an electrical short or transient fuel. Localized.				Yes.	3×10^{-5} (area is small and normally roped off)	(comparison)	MUPS fails by failing power or control to makeup valves V4, V5, and V32.
		2. Engulfing.	Open Area Open Area	AB-FZ-4 AB-FZ-5		Yes.	10^{-5} (see above)	No action; subset of AB-FZ-4, scenario 1.	
Missiles	Transient Sources	3. Transient sources.	Mechanical Damage Localized Steam or Flood through Openings	AB-FZ-4 AB-FZ-5		Yes.	1×10^{-6} (workers are unlikely to bring large pressurized bottles inside)	No action; subset of scenario 5 in terms of impact and less frequent.	Steam has no impact. Steam would be generated if letdown lines are severed. Isolation of the break will be automatic.
		4. Transient sources.	Open Area	AB-FZ-4 AB-FZ-5		Yes.	1×10^{-6}	No action; subset of AB-FZ-4, scenario 4.	
Flood	MUPS Piping	5. MUPS pump in makeup mode.	Open Area	AB-FZ-4 AB-FZ-5 FH-FZ-1		Yes.	3×10^{-6} (1×10^{-4} flood) x (3×10^{-2} operator error in mitigating the flood)	(comparison)	Same scenario as AB-FZ-2, scenario 2. Same frequency BWST emptied. Loss of MUPS, DHR, and building spray. No damage from water jets, steam, or water spray because of no weak equipment and no exposed contacts. Pipe whip may fail other MUPS pipes and valve.
			Open Hatch	AB-FA-1 AB-FA-2					
		6. MUPS in test mode taking suction from BWST.	Open Area	AB-FZ-4 AB-FZ-1 AB-FZ-5 FH-FZ-1		No, frequency much smaller than scenario 4 since only a fraction of the time hooked to BWST.			
			Open Hatch	AB-FA-1 AB-FA-2					
High Energy Pipe Break	MUPS Piping					No.			Impact similar to flooding scenarios 5 and 6.
Spray	MUPS Piping	7. MUPS in test or normal operating mode.	Opening	AB-FZ-4		No, part of flood scenarios.			Not important because components in the area are generally not susceptible to sprays.
Water Jets	MUPS Piping	8. Part of scenarios 5 and 6.				No.			Impact limited, similar to scenario 7.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Penetration Area
Designator: AB-FZ-4
Building: Auxiliary Building

Sheet 1 of 3

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU	A		MU-V14A		X	X			2	BWST Isolation (normally closed).
	B		MU-V14B		X	X			2	BWST Isolation (normally closed).
	A		MU-V16A		X	X			2	HPI injection valve. Located above the grating near the reactor building wall.
	B		MU-V16B		X	X			2	HPI injection valve.
	A				MU-P2A	MU-P2A			T3.11-19	
MU			MV-V-18			X			5-31	
			MV-V-217		X	X			5-31	
MU			MU-V20						T 3.11-20	Fail closed type pneumatic valve.
			MU-V17							
MU					MU-V-36	X	MU-V-32		5-31	
					MU-V-37	X			5-31	
MU						MU-V-1A			5-31	
					X	MU-V-1B			5-31	
					MU-V-2A	X			5-31	
					MU-V-2B	X			5-31	
			MU-V-3			X			5-31	
DC	A				DC-P1A				1	
NS	A				NS-P1A				1	
NR	B					NR-P1B			5-31	Control indicator.
DH			DH-V69						T3.10-1	
			DH-V3		X	X			2	Dropline isolation valve outside the containment.
	A		DH-V4A		X	X			2	Injection valves (low pressure). Located above grating near reactor building wall.
	B		DH-V4B		X	X			2	Injection valves (low pressure).
	A		DH-V7A		X	X			2	Piggyback connections.
	B		DH-V7B		X	X			2	Piggyback connections.
	A				DH-P1A				1	
BS	A		BS-V1A		X	X			C. Adams Letter, 6-19-84	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Penetration Area
Designator: AB-FZ-4
Building: Auxiliary Building

Sheet 2 of 3

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
	B		BS-V18		X	X			C. Adams Letter, 6-19-84	
IC			IC-V3			X			C. Adams Letter, 6-19-84, 5-31	Air operated (thermal barrier cooling). Fail open if DC is lost. Fail closed if air is lost.
	A			IC-P1A					1	
ES	A B				480V ESV CC 1A				T 3.11-16 FHA	Hot shorts may occur in control circuits.
AH	A				AH-E-1A				5-31	
	B				AH-E-1B				5-31	
EF					EF-V-53	X			5-31	
					EF-V-54	X			5-31	
							EF-V-30B		5-31	
							EF-V-30D		5-31	
FW					FW-V-5A	X			5-31	
					FW-V-5B	X			5-31	
					FW-V-92A	X			5-31	
					FW-V-92B	X			5-31	
DH					DH-V-5A	X			5-31	
					DH-V-6A	X			5-31	
DH						DH-V-76A			5-31	
DH			DH-V-12A						5-31	
			DH-V-12B						5-31	
						DH-V-75A			5-31	
BS					BS-V-3A	X			5-31	
			BS-V-2B			X			5-31	
IC					IC-V-1A	X			5-31	MOV-letdown cooler isolation .
					X	IC-V-1B			5-31	MOV-letdown cooler isolation.
					IC-V-2	X			5-31	MOV-letdown cooler isolation.
					IC-V-4				5-31	Reactor building isolation; on inlet header; pneumatic valve.
					IC-V-79A				5-31	RCP-1A isolation.
					IC-V-79B				5-31	RCP-1B isolation.
					IC-V-79C				5-31	RCP-1C isolation.
					IC-V-79D				5-31	RCP-1D cooler isolation.
NS					NS-P-1A				5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Recreation Area
Designator: AJ-FZ-4
Building: Auxiliary Building

Sheet 3 of 3

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NR						NR-V-1B			5-31	NR-P-1B outlet valve.
						NR-V-5			5-31	Normally open MOV, on header after pumps.
						NR-V-18				
					NR-V4A	X			5-31	Deicing isolation.
						NR-V-10A			5-31	IC-HX inlet valve.
						NR-V-10B			5-31	IC-HX inlet valve.

SOURCE AND MITIGATION TABLE

Location Name: Penetration Area
Designator: AB-FZ-4
Building: Auxiliary Building

Sheet 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		Fire Hazards Report	North and South Walls 3-Hour Fire Rated* Ionizing Fire Detector Fire Hose Station Steel Personnel Access Hatch in Floor to AB-FA-2 Deluge Water Spray System (around perimeter of zone) Location AB-FZ-5: Fire Hose Station Portable Dry Chemical Extinguisher FH-FZ-1: Portable Dry Chemical Extinguisher	Fire Hazards Report	Open to adjacent fire zones AB-FZ-5 and FH-FZ-1.
Flood	Pipe Section			None to Contain the Water		
Missiles	Transient sources			None		

*On south boundary adjacent to the makeup pump cubicle.

SOURCE AND MITIGATION TABLE (continued)

Location Name: Penetration Area
Designator: AB-FZ-4
Building: Auxiliary Building

Sheet 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
High Energy Line Break	MUPS Piping and DHR Drop Line when in Normal Hot Leg Recirculation			None		
Waterjets	MUPS Piping			None		
Sprays	Piping (MUPS, B Spray, and DHR)			None		
Steam	DHR Pipes during Normal Hot Leg Cooling					

SCENARIO TABLE

Location Name: Penetration Area
Designator: AB-FZ-4
Building: Auxiliary Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation				
			Type	To					
Fire and Smoke	Cabling, Transient Fuel	1. Cable burning due to an electrical short or transient fuel.	Open	AB-FZ-5		Yes.	2×10^{-6} (3×10^{-3} /yr fire) x (0.3 geometric factor) x (0.5 fail to suppress) x (0.05 severity) x (0.1 hot shorts in NR-V-5 circuit and failure of IC-V-3 or IC-V-2)	Damage state: 3H (ET).	Damage to ESV-CC 1A and 1B cables. Hot short causing spurious actuation of equipment. (See Impact Table.)
		2. Engulfing.	Open Areas on East, South, and West Boundaries (no wall construction) Stairwell	AB-FZ-3 AB-FZ-5 FH-FZ-1		Yes.	3×10^{-6} (1×10^{-3} for severity and non-suppression factor)	No action; less likely than scenario 1; however, more cables lost than scenario 1.	Impact the same as scenario 1.
Flood	Pipe Section	3. Pipe break in suction side piping except for those pipes that are connected to the primary vessel.	Openings	AB-FZ-1 AB-FZ-5 AB-FZ-3 FH-FZ-1 AB-FA-1 AB-FA-2		Yes.	10^{-5} $12 \times 8 \times 10^{-7}$ per year	(comparison) Loss of MUPS, BWST, decay heat and building spray.	12 pipe pieces pose the flood hazard for this scenario. Since safety grade pipes are used, frequency is 8×10^{-7} per year. Valve leakage from valve DH-V-5A can have the same input.
Missiles		4. Transient sources.	Opening Opening	AB-FZ-5 FH-FZ-1		Yes.	3×10^{-6} (3×10^{-3} missile source in the area) x (1×10^{-2}) accident generates the missile) x (0.1 cable fragility)	No action; damage state 3H; more cables or equivalent lost than scenario 1; smaller frequency.	Breaks cables near the ceiling. Breaks valve operators.

SCENARIO TABLE (continued)

Location Name: Penetration Area
Designator: AB-FZ-4
Building: Auxiliary Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario: Source Portion	Paths of Propagation: Type	Paths of Propagation: To	Mitigation	Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
High Energy Line Break		5. MUPS piping or DHR pipes when in operation.	Localized			Yes.	10^{-5} (two pipes)	Damage state 3H. No action; similar to scenario 4.	Pipe failure frequency is conservative because pipes are better than those in data base for 8.0-6.
Waterjets		6. MUPS piping.	Open Area	FH-FZ-1		Yes.	10^{-5} (two pipes)	No action; impact similar to scenario 3.	Isolation valve operation may be damaged. Frequency conservative.
		7. MUPS piping.	Open Area	AB-FZ-5		Yes.	10^{-5} (two pipes)	(comparison) Building spray, DHR, and MUPs affected.	No significant failures at FH-FZ-1. In addition to MUPS isolation valves, DHR and building spray isolation valves affected.
Spray		8. Piping.	Localized			No.	1 x 10^{-5} (two pipe sections)		Only a few valve motors may be affected. Thus, impact very limited. Cables are not susceptible to a spray.
Steam, High Energy Line Break, Spray and Jets		9. DHR pipe break during normal hot leg recirculation.	Openings	FH-FZ-1 AB-FZ-5 AB-FZ-3		Yes.	1 x 10^{-6} (two pipe sections) x (0.1 in hot leg recirculation)	(comparison) V-Scenario via MUPS until isolation. MUPS failure. Building spray and DHR failure.	MUPS failure due to isolation valve-operator failure. DHR and building spray failure.
High Energy Line Break, Steam, Spray and Jets		10. Letdown line break.	Steam through Openings Pipe Whip Localized Jet or Spray Opening	AB-FZ-5 AB-FZ-3 FH-FZ-1 AB-FZ-5 or FH-FZ-1		Yes.	1 x 10^{-5} (two pipe sections)	(comparison) V-scenario via DHR until isolation. MUPS, building spray, and DHR failure.	MUPS failure and DHR and building spray failure from valve-operator failure.

IMPACT TABLE

Location Name: Penetration Area
Designator: AB-FZ-4
Building: Auxiliary Building

Scenario Summary: Fire; Scenario 1; Fire on the Floor or in Cables; Affects Cables Near the Ceiling; Propagates to AB-FZ-5

Systems Lost	Components Affected by the Hazard
NR All Trains	Hot short in the control cables of NR-V-5 (or normally open MOV). This valve is controlled from 480V-ESV-1A. Recovery of this valve not possible because fire in operator's path. System recovery is possible by opening bypass valves.
RCP Thermal Barrier Cooling	IC-V-2 (a normally open MOV) fails closed because of a hot short.
RCP Motor Cooling	Affects motor cooling and letdown cooling.
Letdown Cooling MU All	Damage to control or power cables of MU-V-14A and MU-V-14B (normally closed MOVs).
BS All	Damage to control or power cables of BS-V-1A and BS-V-1B (normally closed MOVs).
IC All	IC-V-3 would fail closed if copper tubing of air line to air operator fails from the fire; hot short in control cables of IC-V-2.
AH-V-1B and AH-V-1C	Hot short in the control cables of these valves (MOVs, normally closed) may open the valve.
MU Train A and C	Power cables to pumps MU-P-2A and MU-P-2C.
480V ESV-1A and ESV-1B	Power feeds to these two electrical cabinets.
CF Trains A and B	Power cables to AHE-1A and AHE-1B in this fire zone.
HL-1	Valve DH-V3 power cable in the area can be recovered by manual operation of the valve after the fire is put out.
HL-2	Valves DH-V7A and DH-V7B power cables in the area, can be recovered by manual operation of the valves after the fire is put out.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: General Area - Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Sheet 1 of 2

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
DH	A		DH-V5A		X	X			2, 5-31	Suction valves (MOVs) of the DHR pump; normally open.
DH	B		DH-V5B		X	X			2, 5-31	
DH	B								1	
DH								DH-3-LT1 DH-3-LT2	1 1	Event monitoring cables.
MU	C					MU-P2C MU-P1C			1 1	
					MU-V-14B MU-V-16C MU-V-16D MU-V-17 MU-V-217					Suction valve (MOV) for pump C. Discharge valve (MOV) for pump :.
DC	A								1	
IC	A								1	
NS	A								1	
NR	B					NR-P-1B			1	
EP	B				460V AC ESV CC-1B				1	
AH	B								1	
BS	B								Assumed	
MU							MU-V-32		5-31	
					MU-V-37	X			5-31	Seal water return cooler.
						MU-V-1A			5-31	MOV letdown isolation.
					MU-V-2A	X			5-31	MOV letdown isolation.
					MU-V-2B	X			5-31	MOV letdown isolation.
						MU-V-8			5-31	
						MU-V-11A			5-31	
			MU-V-11B			X			5-31	
EF							EF-V-30B		5-31	Flow control valve; fails closed.
							EF-V-30D		5-31	Cross-connect AOV; fails closed.
					EF-V-53	X			5-31	Isolation MOV.
					EF-V-54	X			5-31	Isolation MOV.
FW					FW-V-5B				5-31	
					FW-V-92B	X			5-31	

LOCATION INVENTORY CODIFICATION TABLE

Location Name: General Area - Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Sheet 2 of 2

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
DH						DH-V-4B			5-31	Discharge MOV (normally closed).
						DH-V-6A			5-31	Sump suction MOV (normally closed).
					DH-V-6B	X			5-31	Sump Suction MOV (normally closed).
BS						BS-V-3A			5-31	
					X	BS-V-3B			5-31	
						BS-V-2A			5-31	
						BS-V-2B			5-31	
DH						DH-V-75B			5-31	
						DH-V-76B			5-31	
IC					IC-P-1B				5-3	
					IC-V-1A	X			5-3	
					IC-V-2	X			-31	Inboard isolation MOV, normally open.
						IC-V-79B			5-31	
						IC-V-79D			5-31	
NR						NR-V-1B			5-31	
						NR-V-5			5-31	
					NR-V-4A	X			5-31	
					NR-V-4B	X			5-31	
						NR-V-6			5-31	
					X	NR-V-18			5-31	
						NR-V-10A			5-31	
						NR-V-10B			5-31	
						NR-V-15A			5-31	
					NR-V-15B	X			5-31	

SOURCE AND MITIGATION TABLE

Location Name: General Area-Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Sheet 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling Fire Locker Outside the MUPS Pump Room A		Fire Hazards Report, 1-FHA-026	Steel Personnel Access Hatch in Floor to AB-FA-1 Fire Hose Station Dry Chemical Fire Extinguisher Ionization Fire Detection Location AB-FZ-4: Fire Hose Protection FH-FZ-1: Fire Hose Protection Portable Dry Chemical Extinguishers South Wall 3-Hour Fire Rated Three Hollow Metal Louvered Doors in East Wall	Fire Hazards Report	
Flood	Pipe Section of Closed Loops and Tanks	No River Water Loops in the Area		Open to AB-FZ-1 AB-FZ-4 AB-FZ-3 AB-FZ-2A AB-FZ-2B AB-FZ-2C		This fire zone is a collection of rooms that do not generally contain components important to safety.

SOURCE AND MITIGATION TABLE (continued)

Location Name: General Area-Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Sheet 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Steam	Low Pressure Steam ~7 psig	In the Corridor and Southern Half of the Zone and Waste Evaporator Rooms				
Explosion	Hydrogen Line in the Makeup Tank Cubicle			Door is Normally Closed, but Cannot Confine Hydrogen HVAC Ducts		
Fire and Smoke	Transient Fuel					
High Energy Line Break		No Pipes that Carry Fluids Above 275°F or 200 psig in This Area				
Spray	Piping or Tanks					
Motorized Vehicle	Accidental Impact of Motorized Vehicles			All Critical Items are Well above the Floor (such as cables) or Inside Separate Rooms with Concrete Walls (such as MUPS pumps)		
Missiles	Transient Sources					

SCENARIO TABLE

Location Name: General Area - Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Sheet 1 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	1. Cable burning due to an electrical short or transient fuel. Localized near AB-F7-4.				Yes.	3×10^{-4} (3×10^{-3}/yr fire) x (0.1 spurious actuation of two valves)	(comparison)	The loss of all vital components does not lead to any major events except for loss of several standby trains needed for LOCA mitigation. LOCA not possible from this zone.
		2. Near the boundary, the passageway from FH-FZ-1 to AB-FZ-5.	Open	FH-FZ-1		Yes.	3×10^{-4} (3×10^{-3}/yr fire) x (0.1 spurious actuation of two valves)	(comparison) MU-P-2C, 3C; AH-E-1B; 480V-AC-ESV and CC1B; BS-P-1B.	Power cables to ESV-1B, A[illegible]-1B, and BS-P-1B affected. See Impact Table.
		3. Near the boundary.	Open	AB-FZ-4		Yes.	10^{-3}	No action; subset of AB-FZ-4, scenario 1.	
		4. Near the boundary.	Doors	AB-FZ-2B and C		Yes.	1.4×10^{-4} (7 x 10^{-3}/yr fire) x (0.2 geometric factor) x (0.5 failure to suppress) x (0.2 severity)	(comparison) 480V-AC-CC-1B; AH-E-1B; DH-P-1B; MU-P-2C, 3C; BS-P-1B.	See Impact Table.
		5. Near the boundary.	Open	AB-FZ-1		No, because only MOV cables are affected, and MOVs are normally in operational position.			

SCENARIO TABLE (continued)

Location Name: General Area - Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Sheet 2 of 3

Source Type	Synopsis of the Source	Scenario: Source Portion	Scenario: Paths of Propagation: Type	Scenario: Paths of Propagation: To	Scenario: Mitigation Portion	Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
Flood	Pipe Section or Tank	6. Pipe break of closed loops or tanks.	Opening	AB-FZ-1 AB-FA-1 AB-FA-2 AB-FZ-4 AB-FZ-3 FH-FZ-1 AB-FZ-2A AB-FZ-2B AB-FZ-2C		Yes.	10^{-4} (10^{-2}/yr flood) x (0.1 manhole covers are open) x (0.1 flood not stopped on time)	(systems) DHR and building spray pumps flooded.	The manhole covers for the building spray and DHR pump rooms normally closed. All auxiliary building floods either can be isolated remotely or do not have large water capacity source.
Steam	Steam Pipe	7. Steam pipe rupture (8-inch line, 6 psig steam pressure).	Openings	Most of Auxiliary Building and Fuel Handling		Yes.	10^{-5}	(comparison) 480V-AC-ESV-CC-1C, 1A, and 1B.	The equipment susceptible to this scenario are ESV - CCs, which are very far from the source. The operator that is on watch 24 hours a day on Elevat[illegible]0" will notice the steam. Conservatively assumed as affected.
Explosion	Hydrogen Line in the Makeup Tank Room	8. Inadvertent release of hydrogen gets sucked into HVAC.	Door	Filter Rooms and Slurry Pump Room	HVAC exhaust in operation.	No, only one makeup pump is lost if makeup tank is torn by the explosion.			It is assumed that explosion does not occur inside the tank room, and suction lines for MUP-1A and 1B are intact. Hydrogen line is not normally pressurized. The bottle is isolated. It is used only once every couple of days per year (~ 3 days) to pressurize hydrogen plenum in the tank. It is judged that explosion would not cause failure of MU-P-1A, 1B suction line and pump cavitation It is judged that hydrogen propagation beyond the filter and slurry pump room would lead to
			HVAC Suction	AB-FZ-3 Inside the Ventilation Ducts and Eventually Inside the Ventilation Filters	Explosion contained in the ventilation equipment room.				

SCENARIO TABLE (continued)

Location Name: General Area - Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Sheet 3 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
		9. HVAC exhaust does not work and hydrogen touches off in the pump room.	Door	Filter Rooms and Slurry Pump Room	Walls and door contain the explosion.	No, because it only impacts makeup tank and associated piping.			dilution below 4% by volume.
Spray	Piping or Tanks	10. Piping springing a leak. Similar to scenario 1.				No, impact similar to flood of scenario 1, and most vital components are cables that can withstand sprays; DHR and building spray pits need to be flooded.			
Motorized Vehicles	Accidental Impact of Motorized Vehicles on Piping	11. Leads to flooding.				No.			Very unlikely for motorized dollies or carts to be brought to this area. These vehicles do not run fast enough to be able to cause damage to piping.
Missiles	Transient Sources	12. Pressurized bottle acting as a missile; released near AB-FZ-4. Damages equipment above the DHR and and building spray vaults; ends in FH-FZ-1 near the CC-1C cabinet.	Open Area	AB-FZ-4 FH-FZ-1		Yes.	10^{-5} (1.0 bottle in the area) x (10^{-3} accident occurs) x (10^{-1} trajectory is toward AB-FZ-4 x (10^{-1} vital cables are impacted)	(comparison) 480V-AC-ESV-CC-1A, 1B, and 1C; MU-P-2C; NS-P-1A; AH E-1B; BS-P-1B and BS-P-1A.	

IMPACT TABLE

Location Name: General Area - Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Scenario Summary: Fire; Scenario 1; Localized Fire near AB-FZ-4

Systems Lost	Components Affected by the Hazard
ESV/B MU/C	Power cables for 480V-CC-1B load center power and control for train C of MUPS and suction and discharge valves of the C-train (MU-V-16C, MU-V-16D, and MU-V-14B).
ESV/B MU/C	Power cables for 480V-CC-1B load center power and control for train C of MUPS and suction and discharge valves of the train C (MU-V-16C, MU-V-16D, and MU-V-14B.
DC/A	Power cable for DC-P-1A.
NS/A	Power cable for NS-P-1A.
DH/B	Power cable for DH-P-1B and DH-V-4B.
BS/B	Power cable for BS-P-1B.
	Fire cannot have adverse impact on DM-V-5A and DM-V-5B except for failing them as they are.
Sump Recirculation	Sump suction valves (DH-V-6A and DH-V-6B).
	Failure of building spray valves, BS-V-2A, BS-V-2B, BS-V-3A, and BS-V-3B under fire conditions does not impact building spray.
NR/All	Spurious closure of NR-V-5.
IC/All	Spurious closure of IC-V-2, power cable for IC-P-1A.
AH/B	Power cable to AH-E-1B.

IMPACT TABLE

Location Name:	General Area - Elevation 281'-0"
Designator:	AB-FZ-5
Building:	Auxiliary Building
Scenario Summary:	Fire; Scenario 2; Fire in Passageway between FH-FZ-1 and AB-FZ-5

Systems Lost	Components Affected by the Hazard
ESV/B MU/C	Power cables for 480V-CC-1B load center power and control for train C of MUPS and suction and discharge valves of the C-train (MU-V-16C, MU-V-16D, and MU-V-14B.
DC/A	Power cable for DC-P-1A.
NS/A	Power cable for NS-P-1A.
DH/B	Power cable for DH-P-1B and DH-V-4B.
BS/B	Power cable for BS-P-1B.
	Fire cannot have adverse impact on DH-V-5A and DH-V-5B except for failing them as they are.
Sump Recirculation	Sump suction valves (DH-V-6A and DH-V-6B)
	Failure of building spray valves BS-V-2A, BS-V-2B, BS-V-3A, and BS-V-3B under fire conditions does not impact building spray.
NR/All	Spurious closure of NR-V-5.
IC/All	Spurious closure of IC-V-2, power cable for IC-P-1A.
AH/B	Power cable to AH-E-1B.

IMPACT TABLE

Location Name: General Area - Elevation 281'-0"
Designator: AB-FZ-5
Building: Auxiliary Building

Scenario Summary: Scenario 4, Fire Near the Boundary with AB-FZ-2B and AB-FZ-2C

Systems Lost	Components Affected by the Hazard
--	DH-V-5A and DH-V-5B cables affected, but valves remain open.
ESV/B	Power cable to ESV-1B-480V AC-CC.
AH1/B	Power cable to AH-E-1B.
MU/C	Power and control cables to pump MU-P-1C and related valves and pumps.
DM/B	Control cable for DM-V-4B--normally closed B-discharge path will remain blocked.
	Other components of location table judged to be outside the zone of influence of this fire scenario.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Demineralizers and Motor Control Center A
Designator: AB-FZ-6
Building: Auxiliary Building

Sheet 1 of 2

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU			MU-V16C		X	X			2	HPI injection valve.
			MU-V16D		X	X			2	HPI injection valve.
					MU-P-2A	X			5-31	
					MU-V-14A	X			5-31	
					MU-V-16A	X			5-31	
					MU-V-16B	X			5-31	
					MU-V-217	X			5-31	
						MU-V-20			5-31	
					MU-V-36	X			5-31	
						MU-V-1A			5-31	
			MU-V-8		X	X			5-31	
MU			MU-V-6A		X	X			5-31	
			MU-V-6B		X				5-31	
				480V ESV CC-1A	X	X			1-FHA-027	Engineered safeguard valves and heating control center.
DC	B				DC-P1B					Cable tray above the cabinet, extending from south to north.
IC	B				IC-P1B				5-31	
NS	B				NS-P1B To Be Protected					Conduit cable tray above cabinet, extending north to south.
	C				NS-P1C					
NS								Piping	Walk Down	
AH								Purge Duct	Walk Down	
								Duct Isolation Butterfly Valve AH-V-1A	Walk Down	
FW					FW-V-5A	X			5-31	
					FW-V-92A	X			5-31	
DH						DH-V-4A			5-31	Normally closed MOV; DHR injection isolation.
					DH-V-5A	X			5-31	
					DH-V-6A	X			5-31	
					DH-V-7A	X			11-7	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Demineralizers and Motor Control Center A
Designator: AB-FZ-6
Building: Auxiliary Building

Sheet 2 of 2

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
BS					BS-V-3A	X			5-31	
IC						IC-V-1A			5-31	Normally open MOVs; thermal barrier cooling return isolation.
						IC-V-79A			5-31	
						IC-V-79B			5-31	
						IC-V-79C			5-31	
						IC-V-79D			5-31	
NR					NR-V-4A	X			5-31	
						NR-V-6			5-31	
						NR-V-10A			5-31	
						NR-V-10B			5-31	
						KR-V-15A			5-31	
						NR-V-15B			5-31	
EP					EG-V-1B				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Demineralizers and Motor Control Center A
Designator: AB-FZ-6
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	1A Engineered Safeguards Valves and Heating Control Center (IA ES CC)		1-FHA-027	Fire Detection System	1-FHA-027	Area is composed of several compartments, one of which is totally isolated from the rest.
				Fire Hose Station	1-FHA-027	
	Cabling			Deluge System	1-FHA-027	
	1A Auxiliary Building Heating Control			1-Hour Fire Barrier Wall (hypothetical)	1-FHA-027	Will separate the two trains of engineered safeguards valves and heating control centers.
	Fuel Handling Building Heating and Ventilation Control Center 1A and 13			Ionization Detectors	1-FHA-027	
				Portable Extinguishers (dry chemical and CO_2)	Fire Hazard Report	Opposite end of zone.
				Location AB-FZ-9: Fire Hose Protection		
				Portable Dry Chemical Extinguisher		
Flood	Pipe Sections for Nuclear Service over (8-inch line) the Engineered Safeguards Cabinet Makeup; Core Flood and RCS Sample on Other Side of Wall					It is assumed that nuclear service piping is located around the switchgear.
Spray	Pipe Sections			Open Areas		
High Energy Line Break	MUPS Piping					
Missile	Transient Sources					
Steam	No Sources Could Be Identified					
Nitrogen	600-psig N_2 Line on Other Side of Wall					

SCENARIO TABLE

Location Name: Demineralizers and Motor Control Center A
Designator: AB-FZ-6
Building: Auxiliary Building

Sheet 1 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke in Northern Part	480V-ESV-1A	1. Fire Originates in 480V ESV.				Yes.	3 x 10^{-5} $(10^{-3}$ MCC fire) x (0.03 hot short in NR-V5 before loss of power).	(event tree) Damage state D.	ESV-CC-1A affected. See Impact Table.
		2. Large fire and Smoke.	Fire Door Open	AB-FZ-6		Yes.	1 x 10^{-6} $(10^{-3}$ fire) x $(10^{-2}$ severity) x $(10^{-1}$ door left open or barrier failure)	Not important.	ESV-CC-1A and ESV-CC-1B affected. Door failure unlikely because AB-FZ-6a not a passage for fire fighting. See Impact Table
	Transient Fuel	3. Engulfing.	Open Areas	Elevation 305'-0" of Auxiliary Building		Yes.	3 x 10^{-6} $(10^{-1}$ severity factor) (references same as scenario 1)	Not important.	Impact the same as scenario 1.
		4. Large.	Stairs and Grating Next to Tendon Shaft	Elevation 281'-0' of Auxiliary and Fuel Handling Buildings	AB-FZ-4 FH-FZ-1	Yes.	1 x 10^{-6} $(10^{-3}$ fire) x (severity factor)	(no action)	Very unlikely chain of events since the fire has to propagate downward. A difficult event since it is open under the grating and the floor below is a good distance away (has to heat up the cables under the floor slab).
Flood in Northern Part	Nuclear Service Piping	5. Pipe break. Spraying may affect electrical switchgear.	Doorway Stairs Opening	AB-FZ-7 AB-FZ-9 AB-FZ-4		Yes.	2 x 10^{-5} (several pipe break potentials)	(event tree) DHR pumps; building spray pumps. NS/all.	It is conservatively assumed that ESV-CC-1A is deenergized. It is assumed that the door to AB-FZ-6a has a curb several inches high.

SCENARIO TABLE (continued)

Location Name: Demineralizers and Motor Control Center A
Designator: AB-FZ-6
Building: Auxiliary Building

Sheet 2 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
									It is assumed that cabinets are watertight from above. It is assumed water would not collect high enough in AB-FZ-7 to damage the pumps. It is assumed nuclear service is lost because of the break. The RBS and DHR pump rooms are closed off and no water gets in to damage pumps. Freeboard in auxiliary building drain tank is 40,000 gallons.
Flood and Pipe Whip in Northern Part	MUPS Piping	6. Pipe break.	Open	FH-FZ-2	Very thick walls can protect cables and cabinets from the pipe movement.	Yes.	10^{-5} (pipe break)	(system) Seal injection and MUPS injection; one train.	Pipe movement may damage MUPS seal injection and one train of high-head injection.
Missile in Northern Part	Transient Sources	7. Bottles brought in by maintenance crew for repair fail.	Breaks Door or Wall	AB-FZ-6A		Yes.	10^{-6} (10.0 bottle in the area) x (10^{-2} bottle not on cart or no cap on) x (10^{-2} bottle has the worst damage)	No action (very unlikely to break through the dividing wall).	May fail reactor building vent line. N_2, H, O_2 gas bottles reeled through this room to get to waste gas valve room.
		8. Bottles brought in by maintenance crew for repair fail.	Doorway	AB-FZ-7		Yes.	10^{-6} (see scenario 7)	No action; (subset of scenario 11).	

SCENARIO TABLE (continued)

Location Name: Demineralizers and Motor Control Center A
Designator: AB-FZ-6
Building: Auxiliary Building

Sheet 3 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
		9. Bottles brought in by maintenance crew for repair fail.	Doorway	AB-FZ-9		Yes.	10^{-6} (see scenario 8)	No action; (subset of scenario 11).	
All Sources Southern Part of the Fire Zone		10. All sources.				No.			All hazard scenarios originating from this part do not impact important equipment.
Missile in Northern Part	Transient Sources	11. Bottles in the area by maintenance crew.	Localized			Yes.	1×10^{-5} (1 bottle in the area per year) x (10^{-2} bottle not on a cart) x (10^{-3} bottle dropped)	(event tree) 480V-AC-ESV-CC-1A.	Impact of the bottle on cabinet may lead to relay chatter. This could energize control circuits. If NR-V-5 closes, all nuclear service trains would fail.

IMPACT TABLE

Location Name:	Demineralizers and Motor Control Center A
Designator:	AB-FZ-6
Building:	Auxiliary Building
Scenario Summary:	Fire; Scenario 1; Fire Originates in 480V-ESV-1A; Localized to the Zone

Systems Lost	Components Affected by the Hazard
NS/Train B and C	Power cables to pumps (train C is wrapped in 1-hour fire barrier).
DC/Train B	Power cable to pump.
IC to at Least One RCP Thermal Barrier	Hot short in at least one IC-V-79A, IC-V-79B, IC-V-79C, or IC-V-79D, power cable to pump IC-P-1B.
480V-ESV-1A	Ignition of cables or other insulators.
NR/All Trains	Hot short in control circuit of NR-V-5 before 480V-ESV-1A is deenergized (within the cabinet).
RCP Motor Cooling	Hot short in control circuit of normally open MOV NS-V-4 (within the cabinet).
MU/All	Power cables to normally closed injection valves MU-16A, MU-16B, MU-16C, and MU-16D affected.
RCP Seal Injection	Control cable to fail closed type pneumatic valve MU-V-20 affected.
LPI/A	Control cable for DH-V-4A lost.
BS/A	Power and control to BS-V-3A lost.

IMPACT TABLE

Location Name: Demineralizers and Motor Control Center A
Designator: AB-FZ-6
Building: Auxiliary Building

Scenario Summary: Fire; Scenario 2; Large Fire in AB-FZ-6 Affects Power Cables Near the Ceiling and the ESV-1A Cabinet; Door to AB-FZ-6a Opens and Lets Smoke In

Systems Lost	Components Affected by the Hazard
480V-ESV-1A	Fire damage to 480V-ESV-1A.
480V-ESV-1B	Smoke damage to 480V-ESV-1B.
DH/All Trains	Decay heat valves 4A, 4B, 5A, and 5B fail as is (normally closed MOVs) because of ESV A and B failure.
BS/All Trains	Building spray valves 1A and 1B and decay heat valves 5A and 5B fail as is (normally closed MOVs) because of ESV A and B.
MU/All Trains	Makeup valves 14A, 14B, 16A, 16B, 16C, and 16D (normally closed MOVs); powered from ESV A and B.
NS/Train B and C	Power cables to NS-P-1B and NS-P-1C.
DC/Train B	Power cables to PC-P-1B.
IC/Train B	Power cables to IC-P-1B.

IMPACT TABLE

Location Name: Demineralizers and Motor Control Center A
Designator: AB-FZ-6
Building: Auxiliary Building

Scenario Summary: Missile; Scenario 11; Accidental Release of a Pressurized Bottle Hits MCC 480V-ESV-1A Several Times; Frequency 1 x 10^{-5} per Year

Systems Lost	Components Affected by the Hazard
NR/All	NR-V-5 closes because of relay shatter before loss of 480V-ESV-1A.
480V-ESV-1A	Loss of bus due to mechanical failures. NOTE: No other relay chatter events lead to important system train failures other than those affected by 480V-ESV-1A.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Demineralizers and Motor Control Center B
Designator: AB-FZ-6a
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EP				480V ESV CC-1B	X	X			1-FHA-027	Engineered safeguards valves and heating control center.
AH								Containment Purge Duct	Walk Down	
MU					X	MU-P-2C			5-31	
					MU-V-14B	X			5-31	
					MU-V-16C	X			5-31	
					MU-V-16D	X			5-31	
					MU-V-37	X			5-31	
					MU-V-2A	X			5-31	
					MU-V-2B	X			5-31	
FW					FW-V-5B	X			5-31	
					FW-V-92B	X			5-31	
DH						DH-V-4B			5-31	
					DH-V-5B	X			5-31	
					DH-V-6B	X			5-31	
BS					X	BS-V-3B			5-31	
						BS-V-2B			5-31	
IC					IC-V-1A	X			5-31	
					IC-V-2	X			5-31	
						IC-V-79B			5-31	
						IC-V-79D			5-31	
NR					NR-V-4B	X			5-31	
						NR-V-15A			5-31	
					NR-V-15B	X			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Demineralizers and Motor Control Center A
Designator: AB-FZ-6a
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	IA Radiation Waste Control Center		1-FHA-027	Ionization Detector	1-FHA-027	
				1-Hour Fire Barrier Wall* (hypothetical)		Will separate the two trains of engineered safeguards valves and heating.
	Cabling			Location AB-FZ-6: Fire Hose Protection		
	IB Engineered Safeguards Valves and Heating Control Center			Portable Dry Chemical Extinguisher		
				CO_2 Extinguisher		
				Location AB-FZ-9: Portable Dry Chemical Extinguisher		
				CO_2 Extinguisher		
Missile	Transient Sources					

*Encompassing a Class B personnel door and a 1-1/2 hour rated fire damper (Fire Hazard Report).

SCENARIO TABLE

Location Name: Engineered Safeguards Motor Center 3
Designator: AB-FZ-6a
Building: Auxiliary Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, Transient Fuel	1. Fire originates in the electrical cabinet.				Yes.	10^{-3}	(comparison) 480V-AC-ESV-CC-1B.	Hot shorts may cause IC-VZ close and fail IC.
	480V-ESV-1B	2. Large.		AB-FZ-6		Yes.	3×10^{-6} (3×10^{-3}/yr fire) x (0.2 severity) x (0.5 door is left open) x (0.01 smoke damage to ESV-1B)	Not important; 480V-AC-ESV-CC-1A and 1B lost.	Higher frequency for "door left open" is used because the door has to be opened for fire fighting purposes. Smoke damage to cabinet 1A. Smoke damage very unlikely because of moderate voltage equipment and openings in AB-FZ-6 to other areas. Fire fighting mishaps also not likely to damage ESV-1B because of special precautionary measures. See Impact Table.
Missile	Transient Sources	3. Bottles of pressurized gas fail.	Open on Top	AB-FZ-6		Yes.	3×10^{-7} (10.0 bottles in the area) x (10^{-2} bottle not in a cart or no cap) x (3×10^{-4} bottle mishandled) x (10^{-3} wall failure and 1A MCC cabinet failure)	No action; (subset of AB-FZ-6, scenario 8).	
Missile	Transient Sources	4. Bottles of pressurized gas transported through the area fail.	Localized	Walls		Yes.	3×10^{-5}	No action; (subset of scenario 1 in terms of impact).	Frequency of bottle mishandled smaller than scenario 8 of AB-FZ-6 because only moving through.

IMPACT TABLE

Location Name: Engineered Safeguards Motor Center B
Designator: AB-FZ-6a
Building: Auxiliary Building

Scenario Summary: Fire; Scenario 2; Large Fire Fails Cabinet 1A; Door Is Opened to Fight Fire; Smoke Damages Cabinet 1B.

Systems Lost	Components Affected by the Hazard
480V-ESV-1B	Heat damage.
480V-ESV-1A	Smoke damage.
DH/All Trains	DH-V-4A and DH-V-4B, DH-V-5A and DH-V-5B powered from ESV-A and ESV-B (normally closed MOVs).
BS/All Trains	BS-V-1A and BS-V-1B, DH-V-5A and DH-V-5B powered from ESV-A and ESV-B (normally closed MOVs).
MU-All Trains	MU-V-14A and MU-V-14B, MU-V-16A, MU-V-16B, MU-V-16C, and MU-V-16D (normally closed MOVs) powered from ESV-A and ESV-B. NOTE 1: No hot short is possible from smoke damage. NOTE 2: No initiating event.
IC-V-2	Hot short may close this valve, but RCP failure or stoppage would not occur.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Decay Heat Removal and Nuclear Service Closed Cycle Cooling Pump Area
Designator: AB-FZ-7
Building: Auxiliary Building

System	Tr. in or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
IC		IC-P1A			X				1	Intermediate closed cycle cooling pumps.
		IC-P1B			X				1	
NS		NS-P1A			X				1	Nuclear service closed cycle cooling pumps.
		NS-P1B			X				1	
		NS-P1C			X				1	
DC		DC-P1A			X				1	Decay heat closed cycle cooling pumps
		DC-P1B			X				1	
NS				NS-V56A		X X			C. Adams Letter, 6/19/84	Air-controlled, located above the concrete slab on top of the pumps.
				NS-V56B					C. Adams Letter, 6/19/84	
IC				IC-V4		X			3	Air-operated.
AH					X	X		AH-E-15A	E-311-833	Ventilation for pumps. One pump normally running.
					X	X		AH-E-15B	E-311-833	
MU						MU-V-20			5-31	If this fails, operators have several hours to start the other one. Fans stop by temperature switches in the ducts.

SOURCE AND MITIGATION TABLE

Location Name: Decay Heat Removal and Nuclear Service Closed Cycle Cooling Pump Area
Designator: AB-FZ-7
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Flood	Pipe Sections	Break at Any Point		Equipment on Pedestals		
				Stairwell in Location AB-FZ-6 where Water Can Flow Down and Not Accumulate		
Fire and Smoke	Cooling		1-FHA-027	Ionization Detectors	1-FHA-027	
	Pump Oil Systems			1-Hour Fire Barrier; Enclosures Protecting Trays and Conduits	Fire Hazard Report	
				Non-Rated Concrete Walls		
				Concrete Cubicles		All the pump cubicles are open to a common ceiling area.
				Location AB-FZ-6: Fire Hose Protection		
				Portable Dry Chemical Extinguisher		
				CO_2 Extinguisher		
				Location AB-FZ-9: Portable Dry Chemical Extinguisher		
				CO_2 Extinguisher		
Missile	Pumps, Transient Sources, and Two Bottles of Air Attached to Reactor Building Wall			Pump Missile Would Be Contained By Concrete Wall and Ceiling around the Pumps		
High Energy Pipes	Sampling Lines					

SCENARIO TABLE

Location Name: Decay Heat Removal and Nuclear Service Closed Cycle Cooling Pump Area
Designator: AB-FZ-7
Building: Auxiliary Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling Pump Oil Systems	1. Localized some place in room that increases room temperature.				Yes.	3×10^{-5} (severe fire because air handling units well above the floor)	(event tree) AH-E-15A, 15B.	Fans have temperature switches at the ducts downstream to stop the fans. Five would trigger this switch. Concrete walls separating pumps. Note: If one HVAC unit is lost, the operators have several hours to start the other one.
		2. Localized fire on top of slab.				Yes.	10^{-5} (0.1 for inaccessibility)	(event tree) AH-E-15A, 15B.	
		3. Large fire outside the stalls.				Yes.	10^{-4} (0.1 for severity)	(system) Inside containment pumps.	Inside containment pumps are located outside the stalls.
		4. Large fire behind the stalls that affects the cables above the concrete slab.				Yes.	10^{-4} (0.1 for severity)	(system) Nuclear service pumps.	Other pumps (DC and IC) are not affected.
Flood	Pipe Section	5. Nuclear service pipe break can flood place.				Yes.	10^{-5} (pipe break)	(system) IC-P-1A, 1B DC-P-1A.	The only spray incident that can get to three pumps simultaneously.
		Spraying and small volume spilled.	Proximity	Adjacent Pump	Concrete barriers surrounding pumps.				Other spray scenarios are not deemed likely to damage more than one pump.
		Substantial.	Open Area Stairwell	AB-FZ-6 AB-FZ-9 AB-FZ-4 AB-FZ-3 AB-FA-1 AB-FA-2	Equipment is off ground and stairwell leads to lower level.				Will run down the stairs, but valves under the stairs are spray proof.
		6. Inside Containment piping spraying on decay coolant pumps.	Proximity			Yes.	10^{-6} (0.1 for spraying at both pumps)	(comparison) IC-P-1A, 1B DC-P-1A, 1B.	

SCENARIO TABLE (continued)

Location Name: Decay Heat Removal and Nuclear Service Closed Cycle Cooling Pump Area
Designator: AB-FZ-7
Building: Auxiliary Building

Sheet 2 of 7

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Missiles	Transient Sources	7. Impact on cables above slab.				Yes.	10^{-5} (0.03 missile source in the area) x (10^{-2} missile source not protected) x (source is mishandled 0.03)	(comparison) Nuclear service pumps. Decay coolant pumps A and B.	Intermediate cooling pumps would be affected because they are outside the stalls.
	DC or IC pumps	8. Impact on at least two pumps inside stalls by bouncing around decay coolant pumps as source and affecting two intermediate cooling pumps or one intermediate cooling pump (a high speed pump) affecting decay coolant pumps. pumps.				Yes.	3×10^{-6} (10^{-5} preceding scenario) x (0.3 for pumps severely damaged)	(comparison) IC-P-1A, 1B; DC-P-1A, 1B.	Impact similar to scenario 5.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Waste Gas Decay Tank Room, Elevation 305'-0"
Designator: AB-FZ-B
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
				No components of interest in this location.						

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Remainder of Elevation 305'
Designator: AB-FZ-9
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-V-8			S-31	

SOURCE AND MITIGATION TABLE

Location Name: Remainder of Elevation 305'
Designator: AB-FZ-9
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-027	Reinforced Concrete Walls Fire Hose Protection Portable Dry Chemical Extinguishers Portable CO_2 Extinguisher	Fire Hazards Report	
	480V Switchgear (1H, 1M)					
	Control Center (1B Radiation waste, 1B Auxiliary Building Heating)					
	Location Indication and Alarm Transmitter Power Supply					
	Filters					
Hydrogen	Bottles in Waste Gas Valve Room					
Oxygen	Bottles in Waste Gas Valve Room					
Falling Objects	Crane Operation					

SCENARIO TABLE

Location Name: Remainder of Elevation 305'
Designator: AB-FZ-9
Building: Auxiliary Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation				
			Type	To					
Fire and Smoke	Cabling	1. Cable burning due to an electrical short or transient fuel.							
		1a. Engulfing.	Open Area (north wall)	AB-FZ-6 AB-FZ-6A		No, subset of AB-FZ-6 fires.			
Explosion	Hydrogen					No, hydrogen will be diluted sufficiently after escaping the valve room.			

LOCATION INVENTORY CODIFICATION TABLE

Location Name:
Designator: AB-FZ-10
Building:

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
				No components of interest in this location.						

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Decay Heat Removal Pit A
Designator: AB-FA-1
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
DH	A							DH-C-1A	1-FHA-025	Decay heat removal cooler.
DH	A	DH-P1A DH-V-6A			X X	X X			1-FHA-025 5-31	Decay heat removal pump.
BS	A	BS-P1A BS-V-3A	(in building spray room)		X X	X X			1-FHA-025 5-31	Reactor building spray pump. Assumed in building spray room.
DH			DH-V6A		X	X			2	
DH		DH-V-75A DH-V-76A				X X			5-31 5-31	
BS		(Assumed in decay heat pump room)	BS-V3A		X	X			Section B-9 Operations Plant Manual	Building spray pump suction valve from the BWST or reactor building sump. Assumed in decay heat pump room.
DC			DC-V2A			X			C. Adams Letter, 6/19/84	Air-operated valve - inlet to decay heat removal coolers that fails to engineered safeguards position on loss of air.
			DC-V65A			X			C. Adams Letter, 6/19/84	Air-operated valve - bypass to decay heat removal coolers that fails to engineered safeguards position on loss of air.
EG					EG-Y-1D				5-31	

SOURCE MITIGATION TABLE

Location Name: Decay Heat Removal Pit A
Designator: AB-FA-1
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		Fire Hazards Report	Ionization Fire Detectors	Fire Hazards Report	
	Pump Oil System		Fire Hazards Report	Sealed Steel Equipment Access Hatch Covers		
				Location AB-FZ-4: Hose Protection		
				East Wall Is 3-Hour Rated		
				Steel Personnel Access Hatches		Where are these located? Are there barriers around the RBS pumps?
				Remaining Walls Non-Fire Rated Concrete, but Walls in Contact with Dirt at One Side	1-FHA-025	
Flood	Pipe Sections			The Room Is Below the Ground and Has No Opening at Its Sides or Bottom	1-FHA-025	

SCENARIO TABLE

Location Name: Decay Heat Removal Pit A
Designator: AB-FA-1
Building: Auxiliary Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling Pump Oil System	1. Cable burning due to an electrical short or transient fuel. Oil leakage can ignite.							
		Localized to decay heat pump room.				Yes.	3×10^{-4} (small and inaccessible room)	(system) DH-P-1A; BS-V-3A.	
		2. Engulfing.	Equipment Hatches Open	AB-FZ-4		No, a subset of AB-FZ-4 fires.			Propagation beyond AB-FZ-4 is judged to be very unlikely.
			Closed Hatches	Incapable of Propagation					
		3. Localized fire in the building spray pump room.				No, subset of building spray failure rate.			
Flood	DHR Pump Area or RBS Pump Area	4. DHR pipe failure or RBS pipe failure.	Open Hatch	AB-FZ-3 AB-FZ-4 AB-FZ-5 AB-FA-1 (RBS side) AB-FA-2		Yes.	2×10^{-6} (two pipe failures)	(comparison) Impacts the building spray and decay heat pumps only. BWST empty.	BWST emptied. MUPS pumps unaffected. Pipe failure frequency of 8×10^{-7} per year is used because of safety grade construction.
Missile (any one of the areas)	Transient Source	5. Transient sources.	Localized			No, very unlikely; localized impact.			Impact is confined to one pump or heat exchanger.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Decay Heat Removal Pit B
Designator: AB-FA-2
Building: Auxiliary Building

System	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
DH								DH-C1B	1-FHA-025	Decay heat removal cooler.
DH		DH-P1B			X	X			1-FHA-025	Decay heat removal pump.
DH			DH-V-75B DH-V 76B			X X			5-31 5-31	
BS		BS-P-1			X	X			1-FHA-025	Reactor building spray pump.
DH			DH-V6B		X	X			2	
BS			BS-V3B		X	X			Section B-9 Operations Plant Manual	Building spray pump suction valve.
DC			DC-V2B			X			C. Adams ter, 19/84	Air-operated valve - Inlet to decay heat removal coolers that fails to engineered safeguards position on loss of air.
			DC-V65B			X			C. Adams Letter, 6/19/84	Air-operated valve - bypass to decay heat removal coolers that fails to engineered safeguards position or loss of air.

SOURCE AND MITIGATION TABLE

Location Name: Decay Heat Removed Pit 3
Designator: AB-FA-2
Building: Auxiliary Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
	Same as for Zone AB-FA-1 with the addition of the west wall being 3-hour fire rated.					

SCENARIO TABLE

Location Name: Decay Heat Removal Pit B
Designator: AB-FA-2
Building: Auxiliary Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation				
			Type	To					
		Same as for Zone AB-FA-1 with the addition of the west wall being 3-hour fire rated.							

TURBINE BUILDING

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 1 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NR						NR-P-1B			1	NR-P-1B (LR-31)
NR						NR-P-1B			1	NR-P-1B (LR-26)
MU						MV-P-2B			1	MU-P-2B (CS-112)
MU						MV-P-3B			1	MU-P-3B (ED-5022)
AH						X			1	AH-E-1C (reactor building ventilation unit) (CS-551)
EP					X				1	4,160V ES SWGR-1D (MD-1,2) (offsite power)
EP					X				1	4,160V ES SWGR-1E (ME-1,2) (offsite power) (Revision 1).
EP					1C-ESV-MU				Plant Visit	
RC					X				1	Pressurizer Heater Group 8.
RC					X				1	Pressurizer Heater Group 9.
MS			MS-V-3A			X			C. Adams Letter 6-19-84	Turbine bypass valve (AOV.)
MS			MS-V-3B			X			C. Adams Letter 6-19-84	Turbine bypass valve (AOV.)
MS			MS-V-3C			X			C. Adams Letter 6-19-84	Turbine bypass valve (AOV).
MU						MU-V-3			5-31	
MS					MS-V-8A	X			5-31	
MS					MS-V-8B	X			5-31	
MS							MS-V-4A		5-31	
MS							MS-V-4B		5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 2 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MS			MS-V-3D			X			C. Adams Letter 6-19-84	Turbine bypass valve (AOV).
MS			MS-V-3E			X			C. Adams Letter 6-19-84	Turbine bypass valve (AOV).
MS			MS-V-3F			X			C. Adams Letter 6-19-84	Turbine bypass valve (AOV).
CO								CO-C-1	1-FHA-002	Main condenser.
CO								CO-T-2	1-FHA-002	Miscellaneous drain collecting tank.
IA								IA-P-2A	1-FHA-002	Backup auxiliary air compressor.
CO								CO-C-2A	1-FHA-002	Feedwater pump condenser.
CO								CO-C-2B	1-FHA-002	Feedwater pump condenser.
LO								LO-L-2	1-FHA-002	Feedwater pump turbine oil conditioner.
CO	CO-P-2A								1-FHA-002	Condensate booster pump.
CO	CO-P-2B								1-FHA-002	Condensate booster pump.
CO	CO-P-2C								1-FHA-002	Condensate booster pump.
LO		LO-P-8							1-FHA-002	Oil recirculation pump.
MS					MS-V-2B				5-31	
MS					MS-V-10A				5-31	
MS					MS-V-13A				5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB FA-1
Building: Turbine Building

Sheet 3 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
FW								FW-J-7A	1-FHA-002	12th stage exterior drain cooler.
FW								FW-J-7B	1-FHA-002	12th stage exterior drain cooler.
IA								VA-P-1A	1-FHA-002	Inlet condenser vacuum pumps.
[illegible]								VA-P-1B	1-FHA-002	Inlet condenser vacuum pumps.
LO								LO-L-1	1-FHA-002	Main turbine oil conditioner.
CO		CO-P-3A							1-FHA-002	Powdex back wash pump.
CO		CO-P-3B							1-FHA-002	Powdex back wash pump.
CO		CO-P-1A							1-FHA-002	Condensate pump.
CO		CO-P-1B							1-FHA-002	Condensate pump.
CO		CO-P-1C							1-FHA-002	Condensate pump.
MO		MO-P-1A							1-FHA-022	Moisture separator drain pump.
MO		MO-P-1B							1-FHA-022	Mositure separator drain pump.
MO		MO-P-1C							1-FHA-022	Moisture separator drain pump.
MO		MO-P-1D							1-FHA-022	Moisture separator drain pump.
MO		MO-P-1E							1-FHA-022	Moisture separator drain pump.
AS						AS-V-4			5-31	
DH					DH-V-1	X			5-31	
DH					DH-V-2	X			5-31	
DH						DH-V-3			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 4 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MO		MO-P-1F							1-FHA-022	Moisture separator drain pump.
MO								MO-T-1A	1-FHA-022	Moisture separator drain tank.
MO								MO-T-1B	1-FHA-022	Moisture separator drain tank.
MO								MO-T-1C	1-FHA-022	Moisture separator drain tank.
MO								MO-T-1D	1-FHA-022	Moisture separator drain tank.
MO								MO-T-1E	1-FHA-022	Moisture separator drain tank.
MO								MO-T-1F	1-FHA-022	Mositure separator drain tank.
LO		LO-P-1							1-FHA-022	Turbine lube oil pump.
SC		SC-P-1A							1-FHA-003	Secondary services closed cooling pumps.
SC		SC-P-1B							1-FHA-003	Secondary services closed cooling pumps.
SC		SC-P-1C							1-FHA-003	Secondary services closed cooling pumps.
SC								SC-C-1A/B	1-FHA-003	Secondary services closed cooling heat exchanger.
SC								SC-C-1C/D	1-FHA-003	Secondary services closed cooling heat exchanger.
AH					AH-E-1C				5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 5 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
SA								SA-P-1A	1-FHA-003	Service air compressor/ receiver.
SA								SA-P-1B	1-FHA-003	Service air compressor/ receiver.
MS		MS-V-							1-FHA-004	Main steam stop and control valves.
MO								MO-T-2A	1-FHA-004	Moisture separator.
MO								MO-T-2B	1-FHA-004	Moisture separator.
MO								MO-T-2C	1-FHA-004	Moisture seperator.
FW		FW-P-1A							1-FHA-004	Turbine driven feed-water pump.
FW		FW-P-1B							1-FHA-004	Turbine driven feed-water pump.
FW		FW-V-17A							1-FHA-004	
FW		FW-V-16A							1-FHA-004	
FW								FW-J-6A	1-FHA-004	Feedwater heaters.
FW								FW-J-6B	1-FHA-004	Feedwater heaters.
MO								MO-T-2D	1-FHA-004	Moisture separator.
MO								MO-T-2E	1-FHA-004	Mositure separator.
MO								MO-T-2F	1-FHA-004	Moisture separator.
TR			TR-1						1-FHA-004	Transmitter rack.
TR			TR-2						1-FHA-004	Transmitter and instrumentation rack.
NS						NS-V-52C			5-31	
NS						NS-V-53C			5-31	
NR						NR-V-1B			5-31	
RR						RR-V-3C			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 6 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EH			1C MCC						1-FHA-004	Inside containment turbine control center.
LO								LO-T-3	1-FHA-004	Turbine oil reservoir.
CO								CO-F-4A	1-FHA-004	Condensate filter/ tank.
CO								CO-F-4B	1-FHA-004	Condensate filter/ tank.
CO								CO-F-4C	1-FHA-004	Condensate filter/ tank.
CO								CO-F-4D	1-FHA-004	Condensate filter/ tank.
TR			TR-11						1-FHA-004	Transmitter rack.
GS								GS-C-1	1-FHA-004	Gland seal exhauster.
AH		AH-P-5							1-FHA-004	Pump.
EX								EX-T1	1-FHA-004	Low pressure moisture separator.
LO								LO-T-2A	1-FHA-004	Feedwater pump turbine lube-oiled reservoir.
LO								LO-T-2B	1-FHA-004	Feedwater pump turbine lube-oiled reservoir.
AH		AH-P-4B							1-FHA-005	Vacuum pump.
CO				X					1-FHA-005	Powdex panel.
CO		CO-P-6							1-FHA-005	Powdex standby feed pump.
CO								CO-F-4E	1-FHA-005	Condensate filter/ tank.
EP				ED-SGES-1D					5-31	Bus duct.
EP				ED-SGES-1E					5-31	Bus duct.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 7 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
CO								CO-F-4F	1-FHA-005	Condensate filter/ tank.
CO		CO-P-3B							1-FHA-005	
CO		CO-P-5A							1-FHA-005	
CO								CO-E-1	1-FHA-005	
CO								CO-T-3	1-FHA-005	
SC								SC-C-3	1-FHA-005	Isolated phase bus duct cooling unit.
EH				X					1-FHA-005	Reactor coolant pump A and C electric panel (1A 6,900V switchgear).
EH				X					1-FHA-005	Reactor coolant pump B and D electric panel (1B 6,900V switchgear).
EH				X					1-FHA-005	1A 4,160V switchgear.
EH				X					1-FHA-005	1B 4,160V switchgear.
EH				X					1-FHA-005	1C 4,160V switchgear.
EH				X					1-FHA-005	1C 480V switchgear.
EH				X					1-FHA-005	1J 480V switchgear.
EH				X					1-FHA-005	11J 480V switchgear.
EH				X					1-FHA-005	Excitation switchgear.
MS								X	1-FHA-006	High pressure turbine.
MS								X	1-FHA-006	Low pressure turbine A.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 8 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MS								X	1-FHA-006	Low pressure turbine B.
MS								X	1-FHA-007	Low pressure turbine C.
FW								FW-J-4A	1-FHA-006	Feedwater heater (8th stage).
FW								FW-J-4B	1-FHA-006	Feedwater heater (8th stage).
FW								FW-J-5A	1-FHA-006	Feedwater heater (10th stage).
FW								FW-J-5B	1-FHA-006	Feedwater heater (10th stage).
HD								HD-T1	1-FHA-006	6th stage drain collection tank.
TR				TR-8					1-FHA-006	Transmitter and instrumentation tank.
MS			CIV-4						1-FHA-006	Turbine control/ intercept valves.
MS			CIV-5						1-FHA-006	Turbine control/ intercept valves.
LO			LO-P-7A, LO-P-7B, LO-P-7C, LO-P-7D, LO-P-7E, LO-P-7F						1-FHA-006	Turbine bearing lift pumps.
MS			CIV-2						1-FHA-006	Turbine control/ intercept valves.
MS			CIV-3						1-FHA-006	Turbine control/ intercept valves.
MS			CIV-1						1-FHA-007	Turbine control/ intercept valves.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 9 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MS			CI-V-6						1-FHA-007	Turbine control/ intercept valves.
LO			LO-P-7G, LO-P-7H, LO-P-7I, LO-P-7J						1-FHA-007	Turbine bearing lift pumps.
SC								SC-T-1	1-FHA-007	Secondary services surge tank.
HV				X					1-FHA-007	Turbine room heating and ventilation control panel.
AH				X					1-FHA-007	Turbine room supply air fan relay cabinet.
EH				X					1-FHA-007	Transformer.
EH				S-3-A					1-FHA-007	Core monitor.
								X	1-FHA-007	Turbine generator.
FW								FW-J-1A	1-FHA-010	Feedwater heater (2nd stage).
FW								FW-J-1B	1-FHA-010	Feedwater heater (2nd stage).
								X	1-FHA-010	Turbine room crane.
FW								FW-J-2A	1-FHA-011	Feedwater heater (4th stage).
FW								FW-J-2B	1-FHA-011	Feedwater heater (4th stage).
FW								FW-J-3A	1-FHA-011	Feedwater heater (6th stage).
FW								FW-J-3B	1-FHA-011	Feedwater heater (6th stage).

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 13 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH								AHE-100A	1-FHA-040	Feedwater pump cooling fan.
AH								AHE-100B	1-FHA-040	Feedwater pump cooling fan.
AH								AHE-100C	1-FHA-040	Feedwater pump cooling fan.
				X					1-FHA-040	Local indication and alarm panel on wall outside reactor building personnel access hatch.
AH								AH-E-9A/B	1-FHA-040	Penetrations cooling unit.
RC				X					1-FHA-041	Pressurizer heater cabinet 1A.
RC				X					1-FHA-041	Pressurizer heater cabinet 1B.
HV				X					1-FHA-041	Reactor building heating/ventilation control panel.
HV				X					1-FHA-041	Reactor building heating/ventilation control panel.
HV				X					1-FHA-041	Reactor building heating/ventilation switchgear, 1E - 480V.
HV				X					1-FHA-041	Reactor building heating/ventilation switchgear, 1F - 480V.
FW			FW-V-12A						E-304-085	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 11 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
FW			FW-V-17A						E-304-085	
FW			FW-V-16A						E-304-085	
FW			FW-V-16B						E-304-085	
FW			FW-V-92A						E-304-085	AOV.
FW			FW-V-92B						E-304-085	AOV.
FW			FW-V-5A						E-304-085	MOV.
FW			FW-V-4						E-304-085	MOV.
FW			FW-V-3A						E-304-085	MOV.
FW			FW-V-3B						E-304-085	MOV.
FW			FW-V-13						E-304-085	
FW			FW-V-9B						E-304-082	
FW			FW-V-10A						E-304-082	
FW			FW-V-10B						E-304-082	
FW			FW-V-9A						E-304-082	
FW			FW-V-7A						E-304-082	
FW			FW-V-7B						E-304-082	
FW			FW-V-11A						E-304-082	
FW			FW-V-11B						E-304-082	
CO			COV-3A						E-304-082	
CO			COV-3F						E-304-082	
FW			FW-V-2A						E-304-082	MOV.
FW			FW-V-2B						E-304-082	MOV.
FW			FW-V-6						E-304-082	MOV.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 12 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
FW			FW-V-1A						E-304-084	MOV.
FW			FW-V-1B						E-304-084	MOV.
FW			FW-V-8						E-304-084	
FW			FW-V-14						E-304-084	
CO			COV-9A						E-304-084	MOV.
CO			COV-9B						E-305-084	MOV.
FW			FW-V-15						E-304-084	
MS			MS-V-11A, MS-V-11B, MS-V-11C, MS-V-11D, MS-V-11E, MS-V-11F						E-304-011	Manual.
MS			MS-V-12A, MS-V-12B, MS-V-12C, MS-V-12D, MS-V-12E, MS-V-12F						E-304-011	Manual.
MS			CV-1, CV-2, CV-3, CV-4						4192- C-302-011	High pressure turbine inlet control valves.
MS			SV-1, SV-2, SV-3, SV-4						4192- C-302-011	High pressure turbine inlet stop valves.
MS			TD-V-1A, TD-V-1B, TD-V-1C, TD-V-1D, TD-V-1E						4192- C-302-011	
MS			TD-V-2						4192- C-302-011	
MS			TD-V-3A, TD-V-3B						4192- C-302-011	
MS			TD-V-4A, TD-V-4B						4192- C-302-011	
MS			TD-V-5A, TD-V-5B						4192- C-302-011	MOVs.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FX-1
Building: Turbine Building

Sheet 13 of 13

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MS			MS-V-5A, MS-V-5B						4192- C-302-011	
MS			MS-V-56A, MS-V-56B						4192- C-302-011	
MS			MS-V-57A, MS-V-57B						4192- C-302-011	
MS			EX-V-72A, EX-V-72B						4192- C-302-011	
MS			EX-V-73A, EX-V-73B						4192- C-302-011	
CO			CO-V-40A, CO-V-40B, CO-V-40C, CO-V-40D, CO-V-40E							MOVs.
CO			CO-V-41A, CO-V-41B, CO-V-41C, CO-V-41D, CO-V-41E							MOVs.
CO			CO-V-51							AOV.
CO			CO-V2-A, CO-V2-B							MOVs.
CO			CO-V-3A, CO-V-3B							MOVs.
CO			CO-V4							MOV.

SOURCE AND MITIGATION TABLE

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 1 of 3

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Electric Cables		1, and 1-FHA-002 through 1-FHA-016	Automatic Wet Pipe Sprinklers	1, and 1-FHA-002 through 1-FHA-016	Elevation 305'0" - entire elevation except condenser pit area. Elevation 322'0" - entire elevation except condenser bay and switchgear room.
	Switchgear Cabinets		1, and 1-FHA-002 through 1-FHA-016	Automatic Deluge Water Spray	1, and 1-FHA-002 through 1-FHA-016	Elevation 305'0" - for main turbine oil reservoir and conditioner, feedwater pump turbine oil reservoir, and generator hydrogen seal oil unit.
	Lube Oil Systems		1, and 1-FHA-002 through 1-FHA-016	Automatic Deluge Water Spray	1, and 1-FHA-002 through 1-FHA-016	Elevation 305'0" - for main turbine oil reservoir and conditioner, feedwater pump turbine oil reservoir, and generator hydrogen seal oil unit.
	Transient Fuels		1, and 1-FHA-002 through 1-FHA-016	Manually Actuated Preaction Systems	1, and 1-FHA-002 through 1-FHA-016	Elevation 322'0" - turbine feedwater pump bearings. Elevation 355'0" - main turbine bearings.
	Auxiliary Steam Boiler Fuel Oil		C. Husted	Fire Hose Stations	1, and 1-FHA-002 through 1-FHA-016	Located on all elevations.
				Portable Fire Extinguishers	1, and 1-FHA-002 through 1-FHA-016	Dry chemical, halogen, CO_2, water extinguishers located on all elevations.
				Ventilation	1, and 1-FHA-002 through 1-FHA-016	880,000 CFM capacity for smoke removal.
				Doors	1, and 1-FHA-002 through 1-FHA-016	

SOURCE AND MITIGATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 2 of 3

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
				Walls	1, and 1-FHA-002 through 1-FHA-016	
Flood	Pipe Break in Feed System, Condensate System, or Circulating Water System	1-FHA-002 through 1-FHA-016		Sump Pumps	1-FHA-015	SD-P-1A, SD-P-1B, SD-P-5. Rollup door opens on high water level in main condenser pit.
	Accidental Initiation of Fire Suppression Systems					
Steam	Pipe Break in Main Steam or Feedwater Systems	1-FHA-002 through 1-FHA-016		Ventilation Sump Pumps	1 1-FHA-015	880 CFM capacity. SD-P-1A, SD-P-1B, SD-P-5.
Missiles	Gas Bottles, Turbine Rotating Element Failure, Pump Failure	1-FHA-002 through 1-FHA-016		Walls, Floors, Ceilings, Other Equipment	1-FHA-015	
Explosion	Hydrogen, Waste Gas Explosion	1-FHA-002 through 1-FHA-016				
Caustic Attack	Spill of Caustic Fluid From Storage Tanks	1-FHA-002 through 1-FHA-016		Ventilation Sump Pumps	1 1-FHA-015	880 CFM capacity. SD-P-1A, SD-P-1B, SD-P-5.
Falling Objects	Crane	Crane, Boom, or Lifted Object Falls	1-FHA-010 1-FHA-012 1-FHA-015 1-FHA-016	Floor/Platform Gratings, Other Equipment	Plant Visit	Relatively few crane operations during plant operation; more during outages.

SOURCE AND MITIGATION TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 3 of 3

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Pipe Whip	Main Steam		Plant Visit	Pipe Supports	Plant Visit	
	Main Feedwater		Plant Visit	Walls	Plant Visit	
	Auxiliary Steam		Plant Visit	Walls	Plant Visit	

SCENARIO TABLE

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 1 of 5

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cables, Switchgear Cabinets, Lube Oil Systems, Transient Fuels	1. Localized - affecting safety-related cables in turbine building. Fire at elevation 322' near the control building.	(see Source and Mitigation table)			Yes.	5×10^{-5} (10^{-2} overall frequency) x 0.05 geometric factor) x (0.1 severity factor)	(comparison) LOSP, NR-P-1B cable, ESV-1C cable, MV-P-1B and AH-E-1C cable.	It is assumed that all safety cables are in close proximity. It is assumed that main feed-related cables not in the area.
		2. Localized - causes turbine trip only, nesr main feed pumps.				Yes.	10^{-2}	(comparison)	Note: Turbine trip may be impeded if control DC ground is lost. TB valves fail closed or loss of air. TB valves far from the main feedwater pump.
		3. Large fire, engulfing most of TB-FA-1.				Yes.	3×10^{-5} ($10^{-2} \times 3 \times 10^{-3}$ severity factor)	(comparison) TT and LOSP	
		4. Fire propogating to adjacent buildings.				Yes.		No action; deemed as very unlikely.	Fire may damage the corrugated metal wall to reactor building entry area, but is not deemed to lead to damage beyond that point.
Flood	Circulation Water-Related	5. 10^5 gpm spill; rollup door operates properly; no impact on bus bars from auxiliary station transformers.	Leak Through Underneath Doors	Control building stairwell and fuel handling bottom floor FH-FZ-6.		Yes.	3×10^{-4} (3×10^{-3} very large flood) x (10^{-2} operators fail to stop spill)	No action; subset of scenario 16.	Unlikely for water depth in the chiller room to be of sufficient height to damage chillers. Depth of water in the turbine building is estimated to be about 1 foot.
			Leak Through Underneath Doors	Change area fuel handling and into control ventilation chiller area.					

SCENARIO TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 2 of 5

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
		6. Same as scenario 5 except that rollup door is not operating.	Leak Through Underneath Doors	Change area fuel handling and into control ventilation chiller area.	Chiller room has drains that are normally open. No automatic trip signal for CCW pumps from an expansion joint rupture.	Yes.	3×10^{-5} (3×10^{-3} very large flood) x (0.1 failure of rollup door) x 0.1 operators fail to stop spill)	CB-HVAC.	Likelihood of leakage the control building and into control ventilation chiller area is large. See calculation in relation to FH-FZ-6. Depth of water is judged to be 1 foot in the chiller room to damage chiller pumps.
		7. Same as scenario 6 except that spill rate is about 30,000 gpm.	Leak Through Underneath Doors	Change area fuel handling and into control ventilation chiller area.		Yes, subset of scenario 16.	10^{-5} (10^{-2} large flood) x (0.1 door failure) x (10^{-2} operators fail to stop spill)	CB-HVAC.	
Flood/ Spray	Fire Protection Piping	8. Fire protection-related pipe failure or system actuation that fails the offsite power bus bars by spraying on them.				Yes.	5×10^{-6} (10^{-4} many pipe sections) x (0.05 geometric factor)	(comparison) LOSP + TT	Normal system actuation is judged to be unlikely to lead to bus bar failure. Turbine trip is assumed to occur. Flood must spray onto both bus bars near the control building from a failed pipe or valve to control this damage.

Location Name: Turbine Building
Designator: Tb-FA-1
Building: Turbine Building

Sheet 3 of 5

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Steam and Pipe Whip	One of Main Steam Lines between the Intermediate Building and Stop Valves	9. Pipe break in steam lines that whip around and failure of turbine building windows, doors, and structure because of pipe movement and steam impact (pressure). Parts of turbine building fly into switchyard area. Also, may break other nearby steam lines, thus depriving the main feedwater and turbine-driven emergency feedwater pumps.	No Propagation to Other Buildings except for Some Minor Steam Leakage into Fuel Handling Building through the Reactor Building Entry Change Area			Yes.	2×10^{-3} (6×10^{-3} steam line break) x (0.3 loss of offsite power, given steam)	(comparison) Loss of main steam, LOOP, and TT.	Whipping affects nonsafety-related equipment. Breaks out windows or blows out roof fan openings to other buildings (except for a portion of the intermediate building). Note that the B bus bar for auxiliary station transformer is exposed a shorter distance to this steam environment.
Missiles		10. Turbine-related elements.				Yes, in Section ____.		(comparison)	Turbine missiles are addressed in Section ____. Other sources of missile cannot penetrate walls into adjacent buildings. Only failure is loss of offsite power by damaging bus bars.
		11. Gas bottles, pump failure.	No Propagation to Other Buildings			Yes.	5×10^{-6} $20 \times 10^{-3} \times 10^{-2}$ xt (0.03 missile with enough energy and at correct angle)		

SCENARIO TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 4 of 5

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Explosion		12. Hydrogen, gas explosion.	No Propagation to Other Buildings			Yes.	10^{-4} (3×10^{-3} hydrogen explosion) x (.03 fire ensues and is extinguished after it is very severe).	(comparison) LOOP + TT.	No adjacent buildings are affected. Offsite power bus bars are assumed as failed. Must be a severe explosion and fire to fail both bus bars. Other vital cables are far and would not be affected.
Caustic Attack		13. Spill of caustic fluid from caustic storage tanks. If acid tank also fails, violent reactions can take place.	No Propagation to Other Buildings		Will be contained by the curbs around the tanks.	No, cannot fail any component important to safety.		A subset of fire scenarios.	
Falling Objects		14. Crane				No.			Crane is the only source. Cannot drop an offsite power bus bar because they are near the walls. It is judged that objects will not go through and damage switchgears.
Pipe Whip and Steam		15. Main Feed Piping Main feedwater pipe break between containment wall and check v[illegible] FW-12B. S[illegible] operator s[illegible] will f[illegible] backwards out of this hole. Emergency feedwater will come on and continue steaming.	Open Grating on the Floor Open Roll-up Door	Reactor Building Personnel Access Area Fuel Handling FH-FZ-2	Steam would not fail building walls, but may blow out windows.	Yes.	10^{-5} (pipe failure frequency)	(comparison)	Impact on offsite power bus bars. Impact on 1C-ESV and other cables that are nearby. No impact in FH-FZ-2 on important cables or equipment.
Flood		16. Flood of any severity.	Confined to Turbine Building			Yes.	10^{-2}	(comparison) Only turbine trip.	

SCENARIO TABLE (continued)

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Sheet 5 of 5

Source Type	Synopsis of the Source	Scenario				Considered frr Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Steam		17. Auxiliary steam line break.				No, a subset of turbine trip events since it may steam out some turbine-related MCC.			No impact on offsite power because bus bars are too far away.

IMPACT TABLE

Location Name: Turbine Building
Designator: TB-FA-1
Building: Turbine Building

Scenario Summary: Steam; Scenario g; Main Steam Line Break

System Cost	Components Affected by the Hazard
MS	One of four steam lines break.
MF	Proximity to steam line break location, steam environment, or pipe movement affects susceptible components of main feedwater pumps and auxiliaries.
LOOP	Steam buildup throughout the turbine building gets inside the bus bars, carrying power from auxiliary station transformers; also, the debris from building may fall on transformers or other offsite power-related components and cause shorts.
TT	Turbine will trip on loss of steam.

INTAKE SCREEN AND PUMP HOUSE

LOCATION INVENTORY CODIFICATION TABLE

Location Name: 1R Switchgear Area
Designator: TSPH-FZ-1
Building: Intake Screen and Pump House

Sheet 1 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NR										
RR		RR-P-1B			X	X			1	Reactor building emergency cooling river water pump.
EP				1R-480V Switchgear ESG	X				1-FHA-046	
EP				1A Control Center (CC-1A)	X	X			1-FHA-046	
					1T-480V Switchgear ESG 1B-CC-MCC					
RR			RR-V-1B		X	X			2, 4192 C-302-202	RR-P-1B discharge valve.
RR			RR-V-10B		X	X			2, 4192 C-302-202	RR-P-1B recirculation valve.
DR			DR-V-1A						Table 3.10-5 of FHA	
NR		NR-P-1A			X	X			Plant Visit	Changed labels.
NR			NR-V-1A		X	X			Plant Visit	Changed labels.
NR								NR-S-1A	Plant Visit	Changed labels.
DR		DR-P-1B			X	X			Plant Visit	Changed labels.
DR		DR-P-2B			X	X			Plant Visit	Changed labels.
DR								DR-S-1B	Plant Visit	Changed labels.

NOTE: NR-P-1C, NR-V-1C, & DR-P-1A (as reported in DWG 11E-168-02-002 and FHA) relabeled as NR-P-1A, NR-V-1A, and DR-P-1B.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: 1R Switchgear Area
Designator: ISPH-FZ-1
Building: Intake Screen and Pump House

Sheet 2 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
SR		SR-P-1B			X	X			Plant Visit	
SR			SR-V-1B						Plant Visit	
SR								SR-S-1B	Plant Visit	
SR		SR-P-1C			X	X			Plant Visit	Changed labels.
SR			SR-V-1C		X	X			Plant Visit	Changed labels.
SR								SR-S-1C	Plant Visit	Changed labels.
SW		SW-P-1B			X	X			Plant Visit	
SW		SW-P-2B			X	X			Plant Visit	Changed labels.
SW								SW-S-1B	Plant Visit	Changed labels.
FS		FS-P-2			X	X			Plant Visit	Electric motor fire pump.
NR					X	NR-P-1B		S-31		
NR						NR-V-3		S-31		
DR						DR-V-1B		S-31		
RR						RR-V-1A		S-31		
EP						EG-CCESSH-1B		S-31		

NOTE: NR-P-1C, NR-V-1C, & DR-P-1A (as reported in DWG 11E-168-02-002 and FHA) relabeled as NR-P-1A, NR-V-1A, and DR-P-1B.

SOURCE AND MITIGATION TABLE

Location Name: 1R Switchgear Area
Designator: 1SPH-FZ-1
Building: Intake Screen and Pump House

Sheet 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		Fire Hazards Report	Automatic Wet Pipe Sprinkler System	Fire Hazards Report	
	Pump Oil Systems		1-FHA-046	Portable Dry Chemical Extingishers (two)		
	Switchgear Cabinet		1-FHA-046	Location 1SPH-FZ-3: Portable CO_2 Extinguisher		
				Portable Water Extinguisher		
				Doors (fire rating A)		
				Walls - Non-Fire Rated Concrete		
				To be added: North Wall Upgraded to 3-Hour Fire Barrier		
				Class A Rollup Door in North Wall		

SOURCE AND MITIGATION TABLE (continued)

Location Name: 1R Switchgear Area
Designator: ISPH-FZ-1
Building: Intake Screen and Pump House

Sheet 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Flood	Pipe Section Fire Hose and Wet Pipe Sprinker/Deluge Systems					Switchgear cabinets protected by splash guards.

SCENARIO TABLE

Location Name: 1R Switchgear Area
Designator: ISPH-FZ-1
Building: Intake Screen and Pump House

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, Pump Oil Systems, or Switchgear Cabinet	1. Cable burning due to an electrical short or transient fuel. Localized in middle of the room.	Confined to Room Only		(see Source and Mitigation table)	Yes.	1×10^{-6} (10^{-3} x 0.1 geometric factor x 10^{-2} severity;	(comparison)	Fire fails barrier and reaches cables well above above the floor (see impact table).
		2. Localized near the east wall.				Yes.	3×10^{-6} (10^{-3}/year x 0.3 geometric factor x 10^{-2} severity)	(comparison)	Impact the same as scenario 1 except that control cables for 1T switchgear and power for 1R.
		3. Engulfing.	Open West Doors	ISPH-FZ-3		Yes.		No action; subset of scenario 1.	
			Closed Doors	Incapable of propagation.					
Flood and Spray	Pipe Section	4. Pipe break can flood place.	Localized			Yes.	10^{-4}	(comparison)	It is assumed that 3 feet, 6 inches water on the floor fails pumps and switchgear. Water would drain back into river through open manhole. Water buildup very unlikely to reach electrical cabinets or pumps. Spray may fail the adjacent pumps. It is assumed that break is in NR pipe and sprays over to RR and DR pumps.

SCENARIO TABLE (continued)

Location Name: IR Switchgear Area
Designator: ISPH-FZ-1
Building: Intake Screen and Pump House

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
		5. Pipebreak in large river water piping.	Open Doors	ISPH-FZ-2 ISPH-FZ-3	Equipment is on pedestals so that water level would have to reach at least 2 feet.	Yes.	10^{-5} (10^{-1} door leak severity factor x 10^{-4})	(comparison)	It is assumed that 3 feet, 6 inches water on the floor fails pumps and switchgear. Water would drain back into river through open manhole. Water buildup very unlikely to reach electrical cabinets or pumps. Spray may fail the adjacent pumps. It is assumed that break is in NR pipe and sprays over to RR and DR pumps.

IMPACT TABLE

Location Name: 1R Switchgear Area
Designator: ISPM-FZ-1
Building: Intake Screen and Pump House

Scenario Summary: Fire, Scenario 1; Severe Fire in the Middle of the Room

System Cost	Components Affected by the Hazard
1R and 1T Switchgears	Impacts cables near the ceiling control cables for 1R switchgear and power cables for 1P switchgear. Several other components are affected. Their impact is assumed to be the same as loss of both switchgears.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: 1T Switchgear Area
Designator: ISPH-FZ-2
Building: Intake Screen and Pump House

Sheet 1 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NR		NR-P1B			X	X			1	Nuclear Service River Water Pump.
				X		Control	1	NR-P-1C.		
NR			NR-V-1B							
NR			NR-V-2							
NR			NR-V-7							
NR			NR-V-3							
RR		RR-P1A			X	X			1	Reactor Building
RR			RR-V-1A							Emergency Cooling River Water Pump.
RR				1T-480V SWGR ESG	X				1-FHA-046	
RR				1B Control Center (CC-1B)	X	X			1-FHA-046	
EP					1R-480V SWGR 1A-480V MCC					
RR			RR-V-1A		X	X			2, 4192 C-302-202	RA-P-1A discharge valve.
RR			RR-V-10A		X	X			2, 4192 C-302-202	RR-P-1A recirculation valve.
NR			NR-V-1A		X	X			2, 4192 C-302-202	NR-P-1A discharge valve.
NR			NR-V-1B		X	X			2, 4192-C-302-202	NR-P-1B discharge valve.
NR			NR-V-3		X	X			2, 4192 C-302-202	Header valve.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: 1T Switchgear Area
Designator: TSPH-FZ-2
Building: Intake Screen and Pump House

Sheet 2 of 2

System, Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NR		NR-P-1C			X	X			Plant Visit	Changed labels.
NR			NR-V-1C		X	X			Plant Visit	Changed labels.
NR								NR-S-1C	Plant Visit	Changed labels.
DR		DR-P-1A			X	X			Plant Visit	Changed labels.
DR		DR-P-2A			X	X			Plant Visit	Changed labels.
DR			DR-V-1B					DR-S-1A	Plant Visit	Changed labels.
SR		SR-P-1A			X	X			Plant Visit	Changed labels.
SR			SR-V-1A		X	X			Plant Visit	Changed labels.
SR								SR-S-1A	Plant Visit	Changed labels.
SW		SW-P-1A			X	X			Plant Visit	Changed labels.
SW		SW-P-2A			X	X			Plant Visit	Changed labels.
SW								SW-S-1A	Plant Visit	Changed labels.
NR						NR-P-1A		5-31		
NR						NR-V-1A		5-31		
DR		DR-P-1B			X	X		5-31		
DR						DR-V-1A		5-31		
RR					RR-V-1B	X		5-31		
EG						EG-CCESSH-1A		5-31		

SOURCE AND MITIGATION TABLE

Location Name: 1T Switchgear Area
Designator: ISPH-FZ-2
Building: Intake Screen and Pump House

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
	Same as ISPH-FZ-1 except that there are no portable dry chemical extinguishers for mitigation.					

SCENARIO TABLE

Location Name: IT Switchgear Area
Designator: ISPH-FZ-2
Building: Intake Screen and Pump House

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, Pump Oil Systems, or Switchgear Cabinet	1. Cable burning due to an electrical short or transient fuel. Localized in middle of the room.	Confined to Room Only		(see Source and Mitigation table)	Yes.	1×10^{-6} (10^{-3} x 0.1 geometric factor x 10^{-2} severity)	(comparison)	Fire fails barrier and reaches to cables well above above the floor (see impact table).
		2. Localized near the east wall.				Yes.	3×10^{-6} (10^{-3}/year x 0.3 geometric factor 10^{-2} severity)	(comparison)	Impact the same as scenario 1 except that control cables for IT switchgear and power for IR.
		3. Engulfing.	Open West Doors	ISPH-FZ-3		Yes.		No action; subset of scenario 1.	
			Closed Doors	Incapable of propagation.					
Flood and Spray	Pipe Section	4. Pipe break can flood place.	Localized			Yes.	10^{-4}	(comparison)	It is assumed that 3 feet, 6 inches water on the floor fails pumps and switchgear. Water would drain back into river through open manhole. Water buildup very unlikely to reach electrical cabinets or pumps. Spray may fail the adjacent pumps. It is assumed that break is in NR pipe and sprays over to RR and DR pumps.

SCENARIO TABLE (continued)

Location Name: 1T Switchgear Area
Designator: ISPH-FZ-2
Building: Intake Screen and Pump House

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation Type	Paths of Propagation To	Mitigation Portion				
		5. Pipebreak in large river water piping	Open Doors	ISPH-FZ-2 ISPH-FZ-3	Equipment is on pedestals so that water level would have to reach at least 2 feet.	Yes.	10^{-5} $(10^{-1}$ door leak severity factor $\times 10^{-4})$	(comparison)	It is assumed that 3 feet, 6 inches water on the floor fails pumps and switchgear. Water would drain back into river through open manhole. Water buildup very unlikely to reach electrical cabinets or pumps. Spray may fail the adjacent pumps. It is assumed that break is in NR pipe and sprays over to RR and DR pumps.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Trash Rake and Screen Area
Designator: ISPH-FZ-3
Building: Intake Screen and Pump House

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
				No components of interest in this location.						

SOURCE AND MITIGATION TABLE

Location Name: Trash Rake and Screen Area
Designator: TSPH-FZ-3
Building: Intake Screen and Pump House

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls Class A Doors Automatic Wet Pipe Sprinkler System Portable CO_2 Extinguisher Portable H_2O Extinguisher Thermal Detectors in Exhaust Ductwork	Fire Hazards Report	In addition to the four basic walls, there are also two subdividing walls.

SCENARIO TABLE

Location Name: Trash Rake and Screen Area
Designator: ISPH-FZ-3
Building: Intake Screen and Pump House

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation				
			Type	To					
Fire and Smoke	Cabling	1. Cable burning due to an electrical short or transient fuel.							
		1a. Engulfing.	Open Area	ISPH-FZ-2		Yes.		No action; subset of ISPH-FZ -1 scenarios.	
		2. Engulfing.	Open East Door	ISPH-FZ-1		Yes.		No action; subset of ISPH-FZ-1 scenarios.	
			Closed Doors						Incapable of propagation.
		3. Localized.				No; impact insignificant.			Operation of screen wash mechanisms not crucial to safety in short term.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Diesel Fire Pump Room
Designator: TSPH-FA-2
Building: Intake Screen and Pump House

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
FS		FS-P3							1-FHA-046	Diesel fire pump.

FUEL HANDLING BUILDING

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Fuel Handling Building Basement
Designator: FH-FZ-1
Building: Fuel Handling Building

Sheet 1 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU			MV-P-2A		MU-P-1A				FHA	
MU			MV-P-2B		MU-P-1B				FHA	
MU			MV-P-2C		MU-P-1C				FHA	
MU					MU-V17				FHA	
DC					DC-P-1A					
IC					IC-P-1A					
IC					IC-P-1B					
NS					NS-P-1A				FHA	
NS					NS-P-1B				FHA	
RR					RR-P-1A					
RR					RR-P-1B					
[illegible]					DH-P-1A					
DH					DH-P-1B					
DR					DR-P-1A					
DR					DR-P-1B					
NR					NR-P-1A				FHA	
NR					NR-P-1B				FHA	
NR					NR-P-1C				FHA	
EP	A				480VAC ESV-CC-1A				FHA	
EP	B				480V-AC				FHA	
EP					ESV-CC-1B				FHA	
EP	A				480V-AC				FHA	
EP	B				SH-ES-SWGR-1R	X			FHA	
EP					SWGR-1T	X			FHA	
AH					AH-E-1A				FHA	
					AH-E-1B				FHA	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Fuel Handling Building Basement
Designator: FH-FZ-1
Building: Fuel Handling Building

Sheet 2 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU					MU-P-3A				5-31	
MU					MU-P-3B	X			5-31	
MU					MU-P-3C				5-31	
MU					MU-V-12	X			5-31	
MU						MU-V-14A			5-31	
MU						MU-V-14B			5-31	
MU						MU-V-16A			5-31	
MU						MU-V-16B			5-31	
AH					AH-E-1C	AH-E-1C			FHA	
Event Monitoring						All Channels			FHA	
MU						MU-V-16C			5-31	
MU						MU-V-16D			5-31	
MU						MU-V-18			5-31	
MU						MU-V-20			5-31	
MU							MU-V-32		5-31	
MU					MU-V-36	X			5-31	
MU						MU-V-37			5-31	
MU						MU-V-1A			5-31	
MU					X	MU-V-1B			5-31	
MU						MU-V-2A			5-31	
MU						MU-V-2B			5-31	
MU						MU-V-3			5-31	
MU						MU-V-4			5-31	
MU						MU-V-8			5-31	
MU						MU-V-6A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Fuel Handling Building Basement
Designator: FH-F7-1
Building: Fuel Handling Building

Sheet 3 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-V-6B			5-31	
EF							EF-V-30A		5-31	
EF							EF-V-30B		5-31	
EF							EF-V-30C		5-31	
EF							EF-V-30D		5-31	
EF					EF-V-52	X			5-31	
EF					EF-V-53	X			5-31	
EF					EF-V-54	X			5-31	
EF					EF-V-55	X			5-31	
FW					FW-V-5A	X			5-31	
FW						FW-V-5B			5-31	
FW					FW-V-92A	X			5-31	
FW						FW-V-92B			5-31	
MS					MS-V-8A	X			5-31	
MS					MS-V-8B	X			5-31	
MS						MS-V-2A			5-31	
MS						MS-V-2B			5-31	
DH					DH-V-1	X			5-31	
DH					DH-V-2	X			5-31	
DH					DH-V-3	X			5-31	
DH						DH-V-4A			5-31	
DH						DH-V-4B			5-31	
DH						DH-V-5A			5-31	
DH						DH-V-5B			5-31	
DH						DH-V-6A			5-31	
DH						DH-V-6B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Fuel Handling Building Basement
Designator: FH-FZ-1
Building: Fuel Handling Building

Sheet 4 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
BS						BS-V-3A			5-31	
BS						BS-V-3B			5-31	
BS						BS-V-2A			5-31	
BS						BS-V-2B			5-31	
DH						DH-V-75A			5-31	
DH						DH-V-75B			5-31	
DH						DH-V-76A			5-31	
DH						DH-V-76B			5-31	
IC						IC-V-1A			5-31	
IC					X	IC-V-1B			5-31	
IC						IC-V-2			5-31	
IC						IC-V-3			5-31	
IC						IC-V-4			5-31	
AH						AH-D-3B			5-31	
NR						NR-P-1C			5-31	
NR						NR-V-1A			5-31	
NR						NR-V-1B			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-3			5-31	
NR						NR-V-5			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-4B			5-31	
NR					X	NR-V-1B			5-31	
NR						NR-V-10A			5-31	
NR						NR-V-10B			5-31	
DR						DR-V-1A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Fuel Handling Building Basement
Designator: FH-FZ-1
Building: Fuel Handling Building

Sheet 5 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
DR						DR-V-1B			5-31	
RR						RR-V-1A			5-31	
RR						RR-V-1B			5-31	
RR						RR-V-1C			5-31	
RR					RR-V-5	X			5-31	
EG					EG-Y-1B	X			5-31	
EG							EG-CCESV-1C		5-31	

SOURCE AND MITIGATION TABLE

Location Name: Fuel Handling Building Basement
Designator: FH-FZ-1
Building: Fuel Handling Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling IC ESS Valves and Heating Control Center		1	Automatic Wet Pipe Sprinkler System Fire Hose Protection Ionization Fire Detector Portable Dry Chemical Fire Extinguisher Location AB-FZ-4 Fire Hose Protection	Fire Hazards Report	
Flood	Fire Hose Station, RCP Seal Injection Piping			Open Areas to A Building Elevation 281'		
Steam	Auxiliary Steam Line			Openings to Other Parts of Fuel Handling and Auxiliary Building		
Missile	Transient Sources					
Hydrogen Explosion	Hydrogen Lines			The Pipe Is in Use Only a Few Times per Week for a Few Minutes		

SCENARIO TABLE

Location Name: Fuel Handling Building Basement
Designator: FH-FZ-1
Building: Fuel Handling Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, Cabinets, or Transient Level	1. Cable burning due to electrical short or transient fuel; localized to center of east wall.				Yes.	10^{-5} (10^{-3} x 10^{-2} severity factor)	(comparison) A, B trains of MUPs and nuclear service lost and a nuclear river train.	
		2. Localized to center of the fire zone.				Yes.	10^{-5}	(comparison) A, B, and C trains of MUPs, a train of nuclear service, and many others.	
		3. Localized near FH-FA-7.				Yes.	3 x 10^{-6} (10^{-3} x 10^{-2} severity x 0.3 geometry)	(comparison) Loss of all nuclear reactor trains.	Loss of both switchgears in screen house and both building spray trains.
		4. Localized to the 480V-ESV-1C.				Yes.	10^{-3}	(comparison)	Localized to 480V-ESV-1C.
		5. Very large fire near east wall.	Open Areas	Elevation 305'-0" of Auxiliary and Fuel Handling Buildings		Yes.	3 x 10^{-6} (10^{-3} x 10^{-2} severity x 0.3 geometry)	(comparison)	
Steam	Auxiliary Steam Pipes	6. Engulfs first floor of fuel handling and auxiliary buildings.	Open	Elevation 281'-0" of Fuel Handling and Auxiliary Buildings (AB-FZ-5; AB-FZ-4)		Yes.	10^{-5}	(system) 1C-MCC-ESV	Only 480V-ESV-1C is affected. Steam concentration level in other parts of building insufficient to cause damage.
Flood	Fire Protection System Seal Injection Cooling Pipes	7. Pipe break.	Open Hatch Open	AB-FA-1 AB-FA-2 AB-FZ-4 AB-FZ-5 AB-FZ-1		Yes.	10^{-4} (many sources)	(system) DHR and reactor building.	Impacts DHR and reactor building spray pumps only.

SCENARIO TABLE (continued)

Location Name: Fuel Handling Building Basement
Designator: FH-FZ-1
Building: Fuel Handling Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Missile	Transient Sources	8. Transient sources.	Localized			Yes.	10^{-6} (10.0 $\times 10^{-2}$ $\times 10^{-3}$ $\times 10^{-2}$)	(comparison)	May fail cabinet 1C-MCC-ESV. May fail cables.
Explosion	Hydrogen	1. Hydrogen leak from the piping and explosion.				No.			Hydrogen pipes are not normally filled with hydrogen. Area around the pipe is very large, and a leak would be diluted very rapidly.

IMPACT TABLE

Location Name: Fuel Handling Building Basement
Designator: FM-FZ-1
Buidling: Fuel Handling

Scenario Summary: Fire, Scenario 4; Fire Localized to Cabinet ESV-480V-CC-1C

Systems Cost	Components Affected by the Hazard
ESV/C	Cabinet Fire Power Cables above or near the Cabinet
NS/B	NS-P-1B
AHYC	AH-E-1C
MU/B	MN-P-1B and Associated Valves
Instrumentation	Instrumentation

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Fuel Handling Building at Elevation 305'
Designator: FH-FZ-2
Building: Fuel Handling Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH						X		AH-D-39	FHA	Fails closed on loss of air, which is not significant.
MU	C	MU-P-1C							FHA	
MU			MU-V17						FHA	
DC	B	DC-P-1B							FHA	
IC	B	IC-P-1C							FHA	
NS	C	NS-P-1C							FHA	
RR	B	RR-P-1B							FHA	
DH	B	DH-P-1B							FHA	
DR	B	DR-P-1B							FHA	
NR	B	NR-P-1C								
EP	B			480V-ESV-MCC 1B					FHA	
EP	B			480V AC-SH-ES-CC-1T					FHA	
AH				AH-E-1B					FHA	
AH				AH-E-18A					FHA	
AH				AH-E-18B					FHA	
MU					MU-P-2C				5-31	
MU				MU-P-3C					5-31	
MU					MU-V-14B				5-31	
MU					MU-V-16C				5-31	
MU					MU-V-16D				5-31	
MU					MU-V-18				5-31	
MU					MU-V-217				5-31	
MU				MU-V-20	X				5-31	
MU						MU-V-32			5-31	
MU					MU-V-37				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Fuel Handling Building at Elevation 305'
Designator: FH-FZ-2
Building: Fuel Handling Building

Sheet 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1	Reinforced Concrete Walls E: 3-Hour Fire-Rated Adjecent to Control Tower S: 3-Hour Fire-Rated up to Fuel Handling Building Operating Floor Class A - Rated Rollup Fire Doors on North, South, and East Walls Rolling Concrete Missile Door on West Wall (railroad entrance) Steel Hatch Access Air Intake Tunnel Automatic Wet Pipe Sprinkler System Carbon Dioxide Fire Extinguisher Dry Chemical Fire Extinguisher Location: Turbine Building Fire Hose Protection	Fire Hazards Report	

SOURCE AND MITIGATION TABLE (continued)

Location Name: Fuel Handling Building at Elevation 305'
Designator: FH-FZ-2
Building: Fuel Handling Building

Sheet 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
				FH-FZ-1 Fire Hose Protection FH-FZ-3 Fire Hose Protection AB-FZ-6 Portable Dry Chemical Extinguisher CO_2 Fire Extinguisher		
Steam	Auxiliary Steam Piping			Open Areas		
Flood	Fire Protection Lines			Open Areas		
Missiles	Transient Sources			Walls		
Falling Objects	Crane			Floor Slab		

SCENARIO TABLE

Location Name: Fuel Handling Building at Elevation 305'
Designator: FH-FZ-2
Building: Fuel Handling Building

Source Type	Synopsis of the Source	Scenario: Source Portion	Scenario: Paths of Propagation: Type	Scenario: Paths of Propagation: To	Scenario: Mitigation Portion	Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
Fire and Smoke	Cabling	1. Cable burning due to an electrical short or transient fuel, localized.				Yes.			(system)
		2. Engulfing.	Open Stairwell	FH-FZ-1 (subsequently to Elevation 305'-0" and 281'-0" of auxiliary and fuel handling buildings)		Yes.	3×10^{-5} $(10^{-3}$ $\times 0.3$ geometric factor $\times 0.1$ severity)	(comparison)	Fire on vertical cables. Fire affects the vertical green and red cables on the end of the corridor next to the rollup door.
			North Door Open	Turbine Building (change area)					
Steam	Auxiliary Steam Piping	3. Pipe break.				No, it does not affect any important equipment.			Cables can sustain steam environment.
Flood	Fire Protection Piping	4. Pipe break.	Opening on the Floor	FH-FZ-6	The opening has a lip about 4 inches high. The main tendency would be for water to get into railroad area.	Yes.	1×10^{-5} (few pipe pieces)	(CB-HVAC)	
Missile	Transient Sources	5. Transient sources.	Localized			Yes.	10^{-6} $(10.0$ $\times 10^{-2}$ $\times 10^{-3}$ $\times 10^{-2})$	(no action) Subset of scenario 4.	May fail cables.
Falling Objects	Crane Failure	6. Crane failure.	Localized			No, crane seldom used during plant operation.			May impact only a few components.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Fuel Handling Building at Elevation 305'
Designator: FH-FZ-2
Building: Fuel Handling Building

Sheet 1 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-V-1A			5-31	
MU						MU-V-2B			5-31	
MU						MU-V-8			5-31	
MU						MU-V-6A			5-31	
MU						MU-V-6B			5-31	
MU						MU-V-11A			5-31	
MU						MU-V-11B			5-31	
EF					EF-V-53	X			5-31	
EF					EF-V-54				5-31	
FW						FW-V-5B			5-31	
FW						FW-V-92B			5-31	
MS						MS-V-8A			5-31	
MS						MS-V-8B			5-31	
MS							MS-V-4A		5-31	
MS							MS-V-4B		5-31	
DH					DH-V-1				5-31	
DH						DH-V-2			5-31	
DH						DH-V-4B			5-31	
DH						DH-V-5B			5-31	
DH						DH-V-6B			5-31	
BS						BS-V-3B			5-31	
BS						BS-V-2B			5-31	
DH						DH-V-75B			5-31	
DH						DH-V-76B			5-31	
IC					IC-P-1B				5-31	
IC						IC-V-1A			5-31	
IC						IC-V-2			5-31	
IC						IC-V-79A			5-31	
IC						IC-V-79B			5-31	
IC						IC-V-79C			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Fuel Handling Building at Elevation 305'
Designator: FH-FZ-2
Building: Fuel Handling Building

Sheet 2 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
IC						IC-V-79D			5-31	
AH						AH-E-1C			5-31	
AH					AH-E-19B				5-31	
AH					AH-P-8A AH-P-8B	X			5-31	
AH					AH-P-9A AH-P-9B	X			5-31	
AH						AH-D-3B			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-5			5-31	
NR						NR-V-4B			5-31	
NR						NR-V-6			5-31	
NR						NR-V-15A			5-31	
NR						NR-V-15B			5-31	
DR						DR-V-1B			5-31	
RR						RR-V-1B			5-31	
EG					EG-Y-1B				5-31	
EG					EG-CCESV-1B				5-31	
EG						EG-CCESSH-1B			5-31	

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Fuel Handling Building at Elevation 305'
Designator: FH-FZ-3
Building: Fuel Handling Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU					MU-P-2A				5-31	
MU					MU-P-36				5-31	
IC					IC-P-18				5-31	

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Fuel Handling Building at Elevation 305'
Designator: FH-FZ-4
Building: Fuel Handling Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
				No Components of Interest in This Location						

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building Patio Area
Designator: FH-FZ-5
Building: Fuel Handling Building

Sheet 1 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
DR						DR-P-1B				
NR						NR-P-1B NR-P-1C				
RR					RR-P-1B					
DHR/CRDM					DH-P-1B			Control Rod Drive Mechanism Power Supply Trip Breaker	1-FHA--35	Elevation 338' 6".
								Control Rod Drive Mechanism Induction	1-FHA-035	Elevation 338' 6".
								Control Rod Drive Mechanism Transformers	1-FHA-035	Elevation 338' [illegible].
AH					X	X		AH-E-94A		Control building hallway booster fans (Elevation 380' 0").
AH					X	X		AH-E-94B		Control building hallway booster fans (Elevation 380' 0").
AH					X	X		AH-E-93A		Control building hallway supply fans (Elevation 322' 0").
AH					X	X		AH-E-93B		Control building hallway supply fans (Elevation 322' 0").

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Patio Area
Designator: FH-FZ-5
Building: Fuel Handling Building

Sheet 2 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
IC						IC-V-1A			5-31	
IC						IC-V-2			5-31	
IC						IC-V-79A			5-31	
IC						IC-V-79B			5-31	
IC						IC-V-79C			5-31	
IC						IC-V-79D			5-31	
AH						X		AH-E-88A		Fail open on loss of air, which is not significant.
AH						X		AH-E-88B		Fail open on loss of air, which is not significant.
ES	C				X					At Elevation 331' 4". We assume 1C-480V ESF valve control center.
AH	B				AH-E-1B				Color Coded Drawings	It is assumed that power cables for the fans are in trays (Elevation 380').
AH						AH-E-1C			FHA	It is assumed that power cables for the fans are in trays (Elevation 380').
AH					AH-E-18A				FHA	It is assumed that power cables for the fans are in trays (Elevation 380').
AH					AH-E-18B				FHA	It is assumed that power cables for the fans are in trays (Elevation 380').
Instrument	A					X				
Instrument	B					X				
EP					480V ACSM-ES-CC-1T					
MU	C				MU-P-1C	MU-P-3C			FHA	
MU	B					MU-P-2B			FHA	
MU					MU-V-17				FHA	
MU						MU-P-3B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Patio Area
Designator: FH-FZ-5
Building: Fuel Handling Building

Sheet 3 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH						AH-P-8A AH-P-8B			5-31	
AH						AH-P-9A AH-P-9B			5-31	
AH						AH-D-28			5-31	
AH						AH-D-38			5-31	
AH						X	AH-D-39		5-31	
AH						AH-D-41A			5-31	
AH						AH-D-41B			5-31	
AH						AH-D-43A AH-D-44A			5-31	
MU						MU-V-14B			5-31	
MU						MU-V-16C			5-31	
MU						MU-V-16D			5-31	
MU						MU-V-18			5-31	
MU						MU-V-217			5-31	
MU							MU-V-32		5-31	
MU						MU-V-37			5-31	
MU						MU-V-1A			5-31	
MU						MU-V-1B			5-31	
MU						MU-V-3			5-31	
MU						MU-V-8			5-31	
MU						MU-V-6A			5-31	
MU						MU-V-6B			5-31	
MU						MU-V-11A			5-31	
MU						MU-V-11B			5-31	
EF						X	EF-V-30A		5-31	
EF							EF-V-30B		5-31	
EF							EF-V-30C		5-31	
EF							EF-V-30D		5-31	
EF						EF-V-53			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Patio Area
Designator: FH-FZ-5
Building: Fuel Handling Building

Sheet 4 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF						EF-V-54			5-31	
EF					X		X	EF-HSPS-A	5-31	
EF					X		X	EF-HSPS-B	5-31	
EF					X		X	EF-HSPS-C	5-31	
EF					X		X	EF-HSPS-D	5-31	
FW						FW-V-5A			5-31	
FW						FW-V-5B			5-31	
FW						FW-V-92A			5-31	
FW						FW-V-92B			5-31	
MS						MS-V-8A			5-31	
MS						MS-V-8B			5-31	
MS							MS-V-4A		5-31	
MS							MS-V-4B		5-31	
MS						MS-V-2B			5-31	
AS						AS-V-4			5-31	
DH						DH-V-1			5-31	
DH						DH-V-2			5-31	
DH						DH-V-3			5-31	
DH						DH-V-4B			5-31	
DH						DH-V-5B			5-31	
DH						DH-V-6B			5-31	
BS						BS-V-3B			5-31	
BS						BS-V-2B			5-31	
DH						DJ-V-75B			5-31	
DH						DH-V-76B			5-31	
AH					AH-E-19B				5-31	
AH						AH-D-43B AH-D-44B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Patio Area
Designator: FH--Z-5
Building: Fuel Handling Building

Sheet 5 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NS						NS-V-52C			5-31	
NS						NS-V-53C			5-31	
NR						NR-V-1B			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-5			5-31	
NR						NR-V-4B			5-31	
NR						NR-V-6			5-31	
NR						NR-V-15A			5-31	
NR						NR-V-15B			5-31	
DR						DR-V-1B			5-31	
RR						RR-V-1B			5-31	
RR						RR-V-3C			5-31	
RR						RR-V-5			5-31	
EG						EG-CCESSH-1B			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building Patio Area
Designator: FH-FZ-5
Building: Fuel Handling Building

Sheet 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1	Elevation 322'-0"; Three Class A Doors on East Wall	Fire Hazards Report	
	Control Building Heating Control Center			Elevation 338'-6"; Two Class A and One Class B Door on East Wall		
	1G 480V Switchgear Reactor Plant			Elevation 355'-0"; One Class A and One Class B Door on East Wall		
	1L 480V Switchgear Reactor Plant			Elevation 380'-0"; One Class B and Two Unrated Doors on East Wall		
	1A Reactor Plant Control Center			Two Fire Hose Stations in Each Level Except the 380'-0" Level with One Station		
	Control Rod Drive Mechanism Transformers			Elevation 322'-0" and 338'-6"; Portable Dry Chemical Extinguishers		
	Control Rod Drive Mechanism Induction			Elevation 355'-0"; Portable Dry Chemical Extinguisher CO_2 Extinguisher		
				Elevation 380'-0"; Portable Water Extinguisher		

SOURCE AND MITIGATION TABLE (continued)

Location Name: Control Building Patio Area
Designator: FH-FZ-5
Building: Fuel Handling Building

Sheet 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
				Dry Chemical Extinguisher		
Flood	Fire Protection			Grating Floor		
				Walls; Doors Normally Closed		
Steam	Auxiliary Steam Line			Grating Floor		
				Doors Normally Closed		
Falling Objects	Crane			Grating Floor Can Hold 200 Pounds Per Ft^2		
Missiles	Transient Sources			Grating Can Hold 200 Pounds Per Ft^2		
	Halon					
Pipewhip	Auxiliary Steam Line					

SCENARIO TABLE

Location Name: Control Building Patio Area
Designator: FH-FZ-5
Building: Fuel Handling Building

Sheet 1 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling*	1. Cable burning due to an electrical short or transient fuel. Fire on Elevation 380'-0".	Localized			Yes.	3×10^{-4} (10^{-3} $\times$ 0.3 for geometric factor)	(system)	AH-E-18A and 18B affected.
		2. Fire on Elevation 355'-0".				Yes.	3×10^{-4}	(system)	AH-E-18A and 18B affected.
		3. Fire on Elevation 338'-6".				Yes.	3×10^{-4}	(comparison)	AH-E-18A and 18B and event monitoring affected.
		4. Fire on Elevation 322'-0".				Yes.	10^{-5} (10^{-3} $\times 10^{-2}$ geometric factor)	(comparison) Nuclear river pump 1B and 1C lost.	All FH-FZ-5 cables affected.
		5. Fire on Elevation 322'-0".	Open East Doors for Smoke Propagation (additional doors open in areas mentioned would result in propagation throughout level)	CB-FA-2g CB-FA-2c		Yes.	10^{-5} (10^{-3} $\times 10^{-2}$ doors open)	(comparison)	Smoke does not fail cables. No impact in CB-FA-2g.
		6. Fire on Elevation 338'-6" or below.	Open East Doors (additional doors open in areas mentioned would result in propagation throughout level)	CB-FA-3c CB-FA-3d		Yes.	10^{-5} (10^{-3} $\times 10^{-2}$ see above)	(comparison)	No impact in CB-FA-3c. Smoke damage on cabinets only.

Even though the area covers four elevations, fire and smoke can spread fairly easily since each floor is only composed of steel grating.

SCENARIO TABLE (continued)

Location Name: Control Building Patio Area
Designator: FH-FZ-5
Building: Fuel Handling Building

Sheet 2 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
		7. Fire on Elevation 355'-0" or below.	Open East Door (additional doors open in areas mentioned would result in propagation throughout level)	CB-FA-4b		Yes.	10^{-5} (see above)	(comparison)	Smoke damage on cabinets only.
	480V Switchgear (1L,1G) Control Centers (1A, 1B reactor plant and control building heating) Control Rod Drive Mechanical Transformer Control Rod Drive Mechanical Induction	8. Fire on Elevation [illegible]'-0" or below.	Open East Doors (additional doors open in areas mentioned would result in propagation throughout level)	CB-FZ-5a CB-FZ-5b		No.			Items in CB-FA-5a and 5b not sensitive to smoke, except may suck smoke into other containment building areas.
Flood	Fire Protection Piping	9. Pipe break.	Grating	FH-FZ-2 FH-FZ-6		Yes.	10^{-4} (many sources)	(CB-HVAC)	
Steam and Pipe Whip	Auxiliary Steam	10. Pipe break.	Grating (steam) Localized (pipe whip)	FH-FZ-2 FH-FZ-6 FH-FZ-1 Change Room		No, judged that steam cannot fail exposed components, including chillers in FH-FZ-6; pipe whip cannot damage cables because steam pipe is far from cables.			

SCENARIO TABLE (continued)

Location Name: Control Building Patio Area
Designator: FH-FZ-5
Building: Fuel Handling Building

Sheet 3 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Falling Objects	Crane	11. Heavy object dropped from crane, breaking grating floor.				No, grating may stop drop, or object may go through open hatch and land on floor. Also, crane seldom used during plant operation.			Assumption - objects carried by crane on equipment from control building and not objects containing hazardous materials. Damage cable.
Missiles	Transient	12. Pressurized bottles.	Grating	FH-FZ-2		Yes.	(10.0 bottle in the area $\times 10^{-2}$ $\times 10^{-3}$ $\times 10^{-2}$)	(no action) Subset of scenario 4.	Only cables are damaged. May lead to fire protection pipe failure, but not considered as credible.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Chiller Room
Designator: FH-FZ-6
Building: Fuel Handling Building

Sheet 1 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH		AH-P-3A			X	X			1-FHA-034	Chilled water supply pumps.
AH		AH-P-3B			X	X			1-FHA-034	Chilled water supply pumps.
AH		AH-C-4A			X	X			1-FHA-034	Control building water chillers.
AH		AH-C-4B			X	X			1-FHA-034	Control building water chillers.
AH		AH-P-8A			X	X			1-FHA-034	Control tower instrument air compressors.
AH		AH-P-8B			X	X			1-FHA-034	Control tower instrument air compressors.
AH		AH-P-9A			X	X			1-FHA-034	Control tower instrument air compressors.
AH		AH-P-9B			X	X			1-FHA-034	Control tower instrument air compressors.
NS			NS-V-108A		X	X			C. Adams Letter, 6/19/84	Nuclear service water to control building ventilation.
NS			NS-V-108B		X	X			C. Adams Letter, 6/19/84	Nuclear service water to control building ventilation.
NS					NS-P-1A				Plant Visit	
MU					MU-P-1A				Plant Visit	Assumptions: These cables are in conduits.
DH					DH-P-1A				Plant Visit	Train B and C parallel to these conduits but underground. Conduits about 8 fe above the floor.
BS					BS-P-1A				Plant Visit	Many control and Instrumentation cables are probably in these conduits.
EP					480V ACESV-1A				Plant Visit	
IC					IC-P-1A				Plant Visit	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Chiller Room
Designator: FH-FZ-6
Building: Fuel Handling Building

Sheet 2 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH					AH-E-1A				Plant Visit	
MU					MU-P-1B				5-31	
MU						MU-P-2A			5-31	
MU					MU-P-3A				5-31	
MU						MU-V-14A			5-31	
MU						MU-V-16A			5-31	
MU						MU-V-16B			5-31	
MU						MU-V-36			5-31	
MU						MU-V-1B			5-31	
MU						MU-V-4			5-31	
EF							EF-V-30A		5-31	
EF					EF-V-52	X			5-31	
EF					EF-V-53				5-31	
EF					EF-V-54				5-31	
EF					EF-V-55	X			5-31	
MS					MS-V-8A	X			5-31	
MS					MS-V-8B	X			5-31	
MS						MS-V-2A			5-31	
MS						MS-V-2B			5-31	
DH						DH-V-4A			5-31	
DH						DH-V-5A			5-31	
DH						DH-V-6A			5-31	
BS						BS-V-3A			5-31	
BS						BS-V-2A			5-31	
DH						DH-V-75A			5-31	
DH						DH-V-76A			5-31	
IC						IC-V-3			5-31	Protected.
IC						IC-V-4			5-31	Protected.
NS					NS-P-1A				5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Chiller Room
Designator: FH-FZ-6
Building: Fuel Handling Building

Sheet 3 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NS					NS-P-1B				5-31	
NR						NR-P-1A			5-31	
NR						NR-V-1A			5-31	
NR						NR-V-3			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-1B			5-31	
NR						NR-V-10A			5-31	
NR						NR-V-10B			5-31	
DC					DC-P-1A				5-31	
DR						DR-P-1A			5-31	
DR						DR-V-1A			5-31	
RR					RR-P-1A				5-31	
RR					RR-V-1A				5-31	
EG					EG-CCESV-1C				5-31	
EG						EG-CCESSH-1A			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Chiller Room
Designator: FH-FZ-6
Building: Fuel Handling Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		Fire Hazards Report	Reinforced Concrete Walls		
	Lube Oil			Class A - Rated Door; East Wall		
				Location		
				Stairwell		
				Portable Dry Chemical Extinguisher		
Flood	Nuclear Service Piping and Fire Protection			Walls		
				Door		Floor area $84m^2$. Chiller pump motors are ~ 8 inches above the floor. Critical chiller pump volume $84 \times 8 \times \frac{2.54}{100} = 17.07m^3$.

SCENARIO TABLE

Location Name: Chiller Room
Designator: FH-FZ-6
Building: Fuel Handling Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling and Transient Fuel	1. Cable burning due to an electrical short or transient fuel.	Localized (fire) Openings (smoke)	FH-FZ-2 FH-FZ-5		Yes.	10^{-3}	(CB-HVAC)	Affects both chillers or pumps only. Cables are only partially affected.
		2. Large fire affecting cables.	Localized (fire) Openings (smoke)	FH-FZ-2 FH-FZ-5		Yes.	9×10^{-7} (10^{-3}/ year fire) x (0.3 geometric factor x (0.2 failure to suppress) x (0.03 severity) x (0.5 spurious actuation)	(comparison)	CB-HVAC affected and train A of all safety equipment; spurious closure of either IC-V-3 or IC-V-4.
		3. Large fire affecting cables.	Open East Door (access to stairwell)	Smoke in Stairwell		No, subset of scenario 2 and no impact by the smoke.	3×10^{-5} (3×10^{-4} large fire) x (0.1 door left open)		
Flood	Nuclear Service or Fire Protection	4. Pumps fail from 3-foot deep water in the room.	Door	Control Building Stairwell		Yes	2×10^{-5} (two pipe pieces)	(CB-HVAC)	Pumps for the chillers affected.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Enclosed Room within FH-FZ-1 (lubricant storage room)
Designator: FH-FZ-7
Building: Fuel Handling Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
				No Components of Interest in This Location						

SOURCE AND MITIGATION TABLE

Location Name: Enclosed Room within FH-FZ-1 (Lubricant Storage Room)
Designator: FH-FZ-7
Building: Fuel Handling Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Transient Fuel, Cable, and Lubricants		Fire Hazards Report	Reinforced Concrete Wall (one) Class A Door Automatic Wet Pipe Sprinkler System Ionization Fire Detector Location FH-FZ-1 Fire Hose Protection Portable Dry Chemical Extinguishers	Fire Hazards Report	

SCENARIO TABLE

Location Name: Enclosed Room within FH-FZ-1
Designator: FH-FA-7
Building: Fuel Handling Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Transient Fuel, Lubricants, or Cables	Ignition. Engulfing.	Open West Door	Whole Building					
			1. Localized			No; no impact.	3×10^{-4}		Smoke has to leave the room to get dispersed by HVAC.
			2. Door (normally closed)	FH-FZ-1		Yes.	10^{-4} 3×10^{-4} frequency) x (0.3 severity factor, or door is left open)	No action (subset of FH-FZ-1 fire scenarios).	Train A cables of ES outside the door.

INTERMEDIATE BUILDING

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Valve Gallery and Penetration Room
Designator: IB-FZ-1
Building: Intermediate Building

Sheet 1 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RR			RR-V-3A		X	X			2, FHA, 3.10-3	
RR			RR-V-3B		X	X			4192 C-302-610	Inlet valves to AH-E-1A, AH-E-1B, and AH-E-1C (MOVs) normally open.
RR			RR-V-3C		X	X			4192 C-302-610	Inlet valves to AH-E-1A, AH-E-1B, and AH-E-1C (MOVs) normally open.
RR			RR-V-4A		X	X			4192 C-302-610	Cooler outlet valves normally closed, MOVs.
RR			RR-V-4B		X	X			4192 C-302-610	Cooler outlet valves normally closed, MOVs.
RR			RR-V-4C		X	X			4192 C-302-610	Cooler outlet valves normally closed, MOVs.
RR			RR-V-4D		X	X			4192 C-302-610	Cooler outlet valves normally closed, MOVs.
RR			RR-V-5		X	X			4192 C-302-610	MOV.
RR			RR-V-6		X	X			4192 C-302-610	Air-operated, fail open type.
NS			NS-V-52A		X	X			4192 C-302-610	Fan motor cooling water valves, pneumatic valves.
NS			NS-V-52B		X	X			4192 C-302-610	Fan motor cooling water valves, pneumatic valves.
NS			NS-V-52C		X	X			4192 C-302-610	Fan motor cooling water valves, pneumatic valves.
NS			NS-V-53A		X	X			4192 C-302-610	Fan motor cooling water valves, pneumatic valves.
NS			NS-V-53B		X	X			4192 C-302-610	Fan motor cooling water valves, pneumatic valves.
NS			NS V-53C		X	X			4192 C-302-610	Fan motor cooling water valves, pneumatic valves.
MS			MS-V-22A						E-304-014	Emergency feedwater relief valves on steam supply.
MS			MS-V-022B						E-304-014	Emergency feedwater relief valves on steam supply.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Valve Gallery and Penetration Room
Designator: IB-FZ-1
Building: Intermediate Building

Sheet 2 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF					EF-V-53	X			5-31	
EF					EF-V-54	X			5-31	
FW					FW-V-5B	X			5-31	
FW					FW-V-92B	X			5-31	
EP					EG-Y-1B				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Valve Gallery and Penetration Room
Designator: IB-FZ-1
Building: Intermediate Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		Fire Hazard Report	Ionization Fire Detector	1-FHA-039	
Flood	Pipe Section		2	Location IB-FZ-5 (upstairs) Contains Portable CO^2 Extinguishers (two) Portable Water Extinguishers (two) Hose Protection (two)		

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SCENARIO TABLE

Location Name: Valve Gallery and Penetration Room
Designator: IB-FZ-1
Building: Intermediate Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	Cable burning due to an electrical short or transient fuel.			Conservatively, no credit to fire protection equipment (upper level) is given.				Smoke is assumed to not impact equipment although it may spread throughout building. Fire is assumed not to be able to get up the stairs.
		1. Localized.				Yes.	10^{-3}	(system)	
		2. Engulfing.	Openings in Walls	IB-FZ-4 IB-FZ-3 IB-FZ-2		No, very unlikely to propagate because 1. Ceiling of IB-FZ-1 is higher than IB-FZ-4. 2. Doors are normally closed. 3. The small cubicles near the reactor building that have door to IB-FZ-2 are empty and unattended.			
Flood	Pipe Section	Pipe break can flood place.							Reactor river is just standing water until cooling required.
		3. Substantial reactor river or nuclear service pipe break.	Floor Openings and Wall Openings	IB-FZ-3 IB-FZ-2 IB-FZ-4	First, the alligator pit (IB-FZ-8) would have to fill and overflow. Second, the equipment is on pedestals.	No, not enough water in nuclear service to be spiked to cause damage.			See attached page for volume calculations. Flow water on nuclear service line to reactor river would alarm the control room in case of reactor river pipe break.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Turbine-Driven Emergency Feedwater Pump Room
Designator: IB-FZ 2
Building: Intermediate Building

Sheet 1 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF		EP-P-1				X			1	Turbine-driven emergency feedwater pump.
EF			EF-V-1B		X	X			2	No emergency feedwater MOVs.
EF			EF-V-8B			X			C. Adams Letter, 6/19/84	Air-operated valve maintained in the failed open position.
MS			MS-V-4A			X			C. Adams Letter, 6/19/84	AOVs.
MS			MS-V-4B			X			C. Adams Letter, 6/19/84	AOVs.
MS			MS-V-13A			X			C. Adams Letter, 6/19/84	AOVs fail open: emergency feedwater, turbine-driven pump, and steam supply valves.
MS			MS-V-13B			X			C. Adams Letter, 6/19/84	AOVs fail open: emergency feedwater, turbine-driven pump, and steam supply valves.
MS			MS-V-10A		X	X			C. Adams Letter, 6/19/84	MOV DC-operated.
MS			MS-V-10B			X			C. Adams Letter, 6/19/84	MOV DC-operated.
MS			MS-V-6			X				AOV fail open type. Assumed based on P&ID inspection.
MS			MS-V-2A			X			FHA	MOV.
MS			MS-V-2B			X			FHA	MOV.
MS			MS-V-8A		X	X			FHA	
MS			MS-V-8B						FHA	
RR			RR-V-3A		X					Inlet MOVs to AH-E-1A, AH-E-1B, and AH-E-1C, normally open. Assumed, based on P&ID inspection.
RR			RR-V-3B		X					Inlet MOVs to AH-E-1A, AH-E-1B, and AH-E-1C, normally open. Assumed, based on P&ID inspection.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Turbine-Driven Emergency Feedwater Pump Room
Designator: IB-FZ-2
Building: Intermediate Building

Sheet 2 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RR			RR-V-3C		X				E-214-025, Revision 1	Inlet MOVs to AH-E-1A, AH-E-1B, and AH-E-1C normally open. Assumed, based on P&ID inspection.
RR			RR-V-4A		X				E-214-025, Revision 1	Oulet valves normally closed MOVs.
RR			RR-V-4B		X				E-214-025, Revision 1	Oulet valves normally closed MOVs.
RR			RR-V-4C		X				E-214-025, Revision 1	Oulet valves normally closed MOVs.
RR			RR-V-4D		X				E-214-025, Revision 1	Oulet valves normally closed MOVs.
RR			RR-V-5		X				E-214-025, Revision 1	MOV.
RR			RR-V-6		X				E-214-025, Revision 1	AOV, fuel outlet.
NS			NS-V-52A		X				E-214-025, Revision 1	Pneumatic valves, fan motor cooling.
NS			NS-V-52B		X				E-214-025, Revision 1	Pneumatic valves, fan motor cooling.
NS			NS-V-52C		X				E-214-025, Revision 1	Pneumatic valves, fan motor cooling.
NS			NS-V-53A		X				E-214-025, Revision 1	Pneumatic valves, fan motor cooling.
NS			NS-V-53B		X				E-214-025, Revision 1	Pneumatic valves, fan motor cooling.
NS			NS-V-53C		X				E-214-025, Revision 1	Pneumatic valves, fan motor cooling.
EF					EF-V-53	X			5-31	
EF					EF-V-54	X			5-31	
FW					FW-V-5B	X			5-31	
FW					FW-V-92B	X			5-31	
AS						AS-V-4			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Turbine-Driven Emergency Feedwater Pump Room
Designator: IB-FZ-2
Building: Intermediate Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Turbine Bearing Oil System		1-FHA-039	Ionization Fire Detector	Fire Hazard Report	
	Cabling		Fire Hazard Report	Location IB-FZ-5 (upstairs) Contains Portable CO^2 Extinguishers (two) Portable H_2O Extinguishers (two) Hose Protection (two)		
Steam	Steam Piping for the EFW Pump	Any Break Upstream of Top Steam Admission Valves	Fire Hazard Report			
Flood	Pipe Section EFW Piping	Any Break Upstream of Pump	2			
Missiles	EFW Turbine Pump			Walls and a Missile Shield Guarding Opening to IB-FZ-3	Plant Visit	
Pipe Whip	Steam Piping	Any Break Upstream of Top Steam Admission Valves				

SCENARIO TABLE

Location Name: Turbine-Driven Emergency Feedwater Pump Room
Designator: IB-FZ-2
Building: Intermediate Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Turbine Bearing Oil System Cabling	1. Oil leakage from turbine pump can ignite; damage turbine pump and electrical cables. Localized.			Conservatively, no credit to fire protection equipment (upper level) is given.	Yes	10^{-3}	(comparison)	Smoke is assumed not to impact equipment although it can spread throughout the building.
		2. Engulfing.	Opening in Walls	IB-FZ-4 IB-FZ-3 IB-FZ-1		No, very unlikely to propagate because 1. Ceiling of IB-FZ-1 higher than IB-FZ-4. 2. Doors are normally closed. 3. The small cubicles near the reactor building that have door to IB-FZ-1 are normally empty and unattended.			Not reasonable to assume the fire will get up the stairs.
Flood	Pipe Section EFW Piping CST Suction	3. Pipe break upstream of pump can flood place. Substantial.	Floor Openings and Wall Openings	IB-FZ-1 IB-FZ-3	First, the alligator pit (IB-FZ-8) would have to fill and overflow. Second, the equipment is on pedestals.	Yes	2×10^{-5} (three pipe sections)	(system,	Alligator pit can handle about 300,000 gallons; about the same as one CST.

SCENARIO TABLE (continued)

Location Name: Turbine-Driven Emergency Feedwater Pump Room
Designator: IB-FZ-2
Building: Intermediate Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Steam (no pipe whip)	Pipe Sect on Main Steam	4. A break in the main steam line to the turbine-driven pump can discharge very large amounts of steam into the room, creating a high humidity environment.				Yes.	2×10^{-5} (four to six pipe sections) x (0.5 no pipe whip)	(comparison)	
		Substantial.	Wall Openings and Gratings Extending the Height of the Building	Whole Building					
Missiles	Turbine of Pump	5. A missile can be generated by the auxiliary feedwater turbine.	Localized		Missile shield and zone walls serve to localize the impact.	Yes.		(no action) Impact the same as scenario 1 and of low frequency.	
Pipe Whip and Steam	Steam Pipe Sections	6. Break in main steam piping.	Localized to Zone, But in Zone, Could Get Cabling along Ceiling		Zone walls serve to localize the impact. Steam in all zones.	Yes.	2×10^{-5} (four to six pipe sections) x (0.5 pipe whip)	(comparison)	Assumed that RR valves are susceptible to the pipe whip. EF pumps are qualified to operate in this environment.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Motor-Driven Emergency Feedwater Pump Area
Designator: IB-FZ-3
Building: Intermediate Building

Sheet 1 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF		EF-P-2A			X	X			1	Motor-driven emergency feedwater pumps.
EF		EF-P-2B			X	X			1	Motor-driven emergency feedwater pumps.
EF			EF-V-1A		X	X			2	Normally open emergency feedwater MOVs.
EF			EF-V-2A		X	X			2	Normally open emergency feedwater MOVs.
EF			EF-V-2B		X	X			2	Normally open emergency feedwater MOVs.
EF			EF-V-30A			X			2	Normally open emergency feedwater throttle valves--fail open on loss of air and in mid-position on loss of control signal.
EF			EF-V-30B			X			2	No emergency feedwater throttle valves--fail open on loss of air and in mid-position on loss of control signal.
								IA-T-1A	1-FHA-039	Air receivers.
								IA-T-1B	1-FHA-039	Air receivers.
					X	X		IA-P-1A	1-FHA-039	Instrument air compressors.
					X	X		IA-P-1B	1-FHA-039	Instrument air compressors.
					X	X		IA-P-2B	4692-302-272	Backup instrument air compressor.
					X	X	X	RM-A2	1-FHA-039	The IA-P-2A is shown on 1-FHA-039, but since it contradicts with 1-FHA-002, the 2B compressor, it is assumed in this location containment atmosphere monitor.
NS			NS-V-55A			X			C. Adams Letter, 6/19/84	Air-controlled; fan ventilation cooling.
NS			NS-V-55B			X			C. Adams Letter, 6/19/84	Air-controlled; fan ventilation cooling.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Motor-Driven Emergency Feedwater Pump Area
Designator: IB-FZ-3
Building: Intermediate Building

Sheet 2 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF			EF-V-30C						5-31	
EF			EF-V-30D						5-31	
EF			EF-V-52						5-31	
EF			EF-V-53						5-31	
EF			EF-V-54						5-31	
RR			RR-V-3A			X				Assumed, based on P&ID inspection. Normally open MOV, RR isolation.
RR			RR-V-3B			X				Assumed, based on P&ID inspection. Normally open MOV, RR isolation.
RR			RR-V-3C			X				Assumed, based on P&ID inspection. Normally open MOV, RR isolation.
RR			RR-V-4A			X			E-214-025, Revision 1	Normally closed MOV, RB isolation.
RR			RR-V-4B			X			E-214-025, Revision 1	Normally closed MOV, RB isolation.
RR			RR-V-4C			X			E-214-025, Revision 1	Normally closed MOV, RB isolation.
RR			RR-V-4D			X			E-214-025, Revision 1	Normally closed MOV, RB isolation.
RR			RR-V-5			X			E-214-025, Revision 1	Normally closed MOV, RB isolation.
RR			RR-V-6			X			E-214-025, Revision 1	Normally closed MOV, RB isolation.
NS			NS-V-52A			X			E-214-025, Revision 1	Normally open, pneumatic RB fan cooler valve.
NS			NS-V-52B			X			E-214-025, Revision 1	Normally open, pneumatic RB fan cooler valve.
NS			NS-V-52C			X			E-214-025, Revision 1	Normally open, pneumatic RB fan cooler valve.
NS			NS-V-53A			X			E-214-025, Revision 1	Normally open, pneumatic RB fan cooler valve.
NS			NS-V-53B			X			E-214-025, Revision 1	Normally open, pneumatic RB fan cooler valve.
NS			NS-V-53C			X			E-214-025, Revision 1	Normally open, pneumatic RB fan cooler valve.
EF			EF-V-55		X	X			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Motor-Driven Emergency Feedwater Pump Area
Designator: IB-FZ-3
Building: Intermediate Building

Sheet 3 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
FW					FW-V-5B	X			5-31	
FW					FW-V-92B	X			5-31	
MS					MS-V-8A	X			5-31	
MS							MS-V-4B		5-31	
MS						MS-V-2A			5-31	
MS					MS-V-10A	X			5-31	
MS					MS-V-10B	X			5-31	
MS						MS-V-13A			5-31	
MS						X	MS-V-13B		5-31	
AH					X	X		AH-E-27A/AH-E-24A	5-31	
AH					X	X		AH-E-24B	5-31	

SOURCE AND MITIGATION TABLE

Location Name: Motor-Driven Emergency Feedwater Pump Area
Designator: IB-FZ-3
Building: Intermediate Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		Fire Hazard Report	Ionization Fire	1-FHA-039	
	Pump and Compressor Lube Oil					
Flood	Pipe Section (auxiliary feedwater) Nuclear Service Piping	Any Break Upstream of Pump			1,2 E-304-8	
Missiles	Air Compressor Components			Walls		
	Assumed that H_2 Analyzer Has an Associated H_2 Bottle					

SCENARIO TABLE

Location Name: Motor-Driven Emergency Feedwater Pump Area
Designator: IB-FZ-3
Building: Intermediate Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Pump and Compressor Lube Oil Cabling	Oil leakage can ignite. 1. Confined.	Proximity	Adjacent Pump	Conservatively, no credit to fire protection equipment (upper level) is given.	Yes.	10^{-3}	(comparison)	Smoke is assumed not to impact equipment although it can spread throughout building. Not reasonable to assume fire could get upstairs (see impact table).
		2. Engulfing.	Opening in Walls	IB-FZ-2 IB-FZ-1 IB-FZ-4		No, because: 1. Very large area. 2. Doors are normally closed. 3. Outside corridor has low fuel loading.			
Flood	Pipe Section (or nuclear service)	Pipe break upstream of pumps could flood place.							
		3. Substantial spray on emergency feedwater pumps.	Floor Openings and Wall Openings	IB-FZ-1 IB-FZ-2	First, the alligator pit (IB-FZ-8) would have to fill and overflow. Second, the equipment is on pedestals.	Yes.	10^{-4} (pipe break or leaks directed toward a target)	(system) Emergency feedwater pumps are affected.	Note: If emergency feedwater pipe had to fail, emergency feedwater system would be lost. The CSTs are about 300,000 gallons (within the capacity of the alligator pit).
Missiles	Air Compressor Components and H_2 Bottle	4. Missiles can be generated.	Localized to Air Compressor Quarters		Geometry of the walls segregating the equipment prevents any impact on the motor-driven emergency feedwater pumps.	Yes.	3×10^{-6} ($0.3 \times 10^{-2} \times \times 10^{-3}$)	(no action) Same impact as scenario 1 and with lower frequency.	Missile affects cables at both ends of the room.

IMPACT TABLE

Location Name: Motor-Driven Emergency Feedwater Pump Area
Designator: IB-FZ-3
Building: Intermediate Building

Scenario Summary: Five, Scenario 1

System Cost	Components Affected by the Hazard
EF/2A, EF/2B	Motor-driven pumps, associated power control cables, valves, and piping.
Fan Coolers	Reactor river valves for emergency function of the fan coolers; cable failures in normally closed valves RR-V-4A, RR-V-4B, RR-V-4C, and RR-V-4D.
--	Nuclear service valves are normally open; fire can only fail their cables, and MOVs fail as they are.
--	Main feedwater valves in the area do not impact main feedwater function.
--	Main steam valves in the area do not impact main steam function.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Remainder of Elevation 295'
Designator: IB-FZ-4
Building: Intermediate Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF								Piping from Both A and B CSTs	Plant Walkdown	Supply for emergency feedwater pumps.
EF							EF-V-30B		5-31	
EF							EF-V-30D		5-31	
EF					EF-V-53	X			5-31	
EF					EF-V-54	X			5-31	
FW					FW-V-5B	X			5-31	
FW					FW-V-92B	X			5-31	
EG					EG-V-1A				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Remainder of Elevation 295'
Designator: IB-F2-4
Building: Intermediate Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-039	Location IB-F2-5 (upstairs) Contains: Portable CO_2 Extinguishers (two) Portable H_2O Extinguishers (two) Hose Protection (two)	Fire Hazards Report	
Flood	Emergency Feedwater and Reactor River Water Piping		Plant Visit			

SCENARIO TABLE

Location Name: Remainder of Elevation 295'
Designator: IB-FZ-4
Building: Intermediate Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling/ H_2 Recombiner Control Panels/Pump Lube Oil	1. Cable burning due to an electrical short or transient fuel/oil leakage from sump pumps can ignite, engulfing the area (most likely the northwest corner, based on what is known).	Openings in Wall	IB-FZ-1 IB-FZ-2 IB-FZ-3		No, no major sources of fuel in the area; may have large transient fuel, but room empty and doors closed to other areas.			The smoke is not considered to impact the equipment in the building although it can travel throughout.
Flood	Piping	1. The reactor river or emergency feedwater piping breaks.	Gratings	Alligator Pit (IB-FZ-8)		No.			The capacity of the two CSTs is about 300,000 gallons x MFW capacity, which the alligator pit is designed to handle. The reactor river piping has standing water unless the coolers are needed.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Intermediate Building at Elevation 305'
Designator: IB-FZ-[illegible]
Building: Intermediate Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RR	All							RR Piping to Emergency Cooling Coils	Plant Visit	Believe that this piping only has standing water since system normally not operational and RR-isolated (based on conversation with M. Kazarians 11/1/84).
H_2	Recombiners									

SOURCE AND MITIGATION TABLE

Location Name: Intermediate Building at Elevation 305'
Designator: IB-FZ-5
Building: Intermediate Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Flood	Pipe Breaks (reactor river water, fire protection)		Plant Visit			
Fire	Transient Fuel					

SCENARIO TABLE

Location Name: Intermediate Building at Elevation 305'
Designator: IB-FZ-5
Building: Intermediate Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Flood	Piping	1. Reactor river pipe break.	Stairs	Alligator Pit (IB-FZ-8)		No.			Believe that only standing water is in piping due to isolation valves, so IB-FZ-8 could handle capacity.
		2. Feedwater pump pipe break.	Stairs	Alligator Pit (IB-FZ-8)		No.			Only if the diesel-driven feedwater pumps were to start and pump more than 300,000 gallons could this be a concern.
Fire		3. Localized fire.	Localized			No, impact is limited to nonsafety components.			
		4. Large fire.	Doorways and Stairs	IB-FZ-1 IB-FZ-2 IB-FZ-3 IB-FZ-4		No, very unlikely and impact similar to fires in the zones to which propagated.			

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Intermediate Building at Elevation 322'
Designator: IB-FZ-6
Building: Intermediate Building

Sheet 1 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MS			MS-V-17A						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-17B						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-17C						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-17D						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-18A						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-18B						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-18C						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-18D						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-19A						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-19B						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-19C						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-19D						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-20A						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-20B						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-20C						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-20D						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-21A						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-21B						E-304-014	Main steam atmospheric relief valves.
MS			MS-V-1A						Plant Visit	Main steam isolation valves (the controls for the valves poke through the ceiling into IB-FZ-7).

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LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Intermediate Building at Elevation 322'
Designator: IB-FZ-6
Building: Intermediate Building

Sheet 2 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MS			MS-V-1B						Plant Visit	Main steam isolation valves (the controls for the valves poke through the ceiling into IB-FZ-7).
MS			MS-V-1C						Plant Visit	Main steam isolation valves (the controls for the valves poke through the ceiling into IB-FZ-7).
MS			MS-V-1D						Plant Visit	Main steam isolation valves (the controls for the valves poke through the ceiling into IB-FZ-7).
FW			FW-V-5B		X	X			5-31	
FW			FW-V-92B		X	X			5-31	
MS					MS-V-8A				5-31	
MS					MS-V-8B	X			5-31	
MS							MS-V-4A		5-31	
MS						MS-V-2B			5-31	
MS						MS-V-10A			5-31	
MS						MS-V-13A			5-31	
AS						AS-V-4			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Intermediate Building at Elevation 322'
Designator: IB-FZ-6
Building: Intermediate Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-041	Reinforced Concrete Walls	Fire Hazards Report	
	Fans (AH-E-73, AH-E-6A, and AH-E-6B)			Fire Hose Protection		
	Electrical Pumps (AH-P-6A and AH-P-6B)			Dry Chemical Extinguisher		
	Industrial Cooler Circulating Pumps (AH-P-2A, AH-P-1B)					
Flood	Pipe Section (high pressure MFW and low pressure fire protection)		1-FHA-041			
Steam	Pipe Section (high pressure main steam) (auxiliary steam)		Plant Visit E-304-014			
Pipe Whip	Mainsteam/Feedwater Piping		Plant Visit			
Missiles	H_2 Analyzer Bottles		Plant Visit			

SCENARIO TABLE

Location Name: Intermediate Building at Elevation 322'
Designator: IB-FZ-6
Building: Intermediate Building

Sheet 1 of .

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	Cable burning due to electrical short or transient fuel.							
		1. Engulfing fire confined to area.				No, impact on the equipment not important.			Smoke not considered to affect the equipment.
		2. Engulfing.	Open Stairwell and Gratings	Whole Building except for a Few Areas Isolated by Doors		No, very unlikely; doors isolate emergency feedwater pump and valve rooms.			Only affects equipment not interested in; not feasible for it to travel down two floors to where other equipment resides.
Flood, Steam, and Pipe Whip	Pipe Section	3. Main feed pipe break could flood place and steam whole building.	Stairs (Northwest) corner) and Open Door to IB-FZ-3 Downstairs Gratings at All Levels	IB-FZ-3	1. The alligator pit (IB-FZ-8) would have to fill and overflow. 2. Equipment is on pedestals.	Yes.	10^{-4}	(comparison)	Alligator pit designed to handle an MFW pipe break. Water would not collect on intermediate building floor. All three emergency feedwater pumps can survive through steam environment. Pipe movement may fail the steam supply line to turbine-driven emergency feedwater pump.
Steam and Pipe Whip	Main Steam Piping	4. Pipe break on any line.	Gratings Stairwells	Entire Building except for of a Few Rooms		Yes.	10^{-4}	(comparison)	Emergency feedwater pumps have passed environmental qualifications test for this scenario; report 500424. Air compressors are assumed failed. Cables will survive the steam environment. Some cables near the break point may be severely damaged.

SCENARIO TABLE (continued)

Location Name: Intermediate Building at Elevation 322'
Designator: IB-FZ-6
Building: Intermediate Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Missiles	H_2 Analyzer Bottles	5. H_2 bottle is dropped and explodes.		If It Heads to Ceiling, Can Get Auxiliary Steam Piping (to TDP) Scenario then Looks like Steam Break		Yes.		(no action) Subset of main feedwater line break or main steam line break.	Valves on high pressure piping are not considered source of missiles.
	Compressed Air Cylinders	6. Dropped and explodes.		If It Heads to Ceiling, Can Get Steam and Feed-Water Piping (far wall has feed-water pump piping). Scenario then Looks Like Steam or Flood Break		Yes.		(no action) Same as above.	

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Intermediate Building at Elevation 355'
Designator: IB-FZ-7
Building: Intermediate Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MS						MS-V-8B			5-31	
MS						MS-V-2B			5-31	
MS						MS-V-10A			5-31	
MS						MS-V-13A			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Intermediate Building at Elevation 355'
Designator: IB-FZ-7
Building: Intermediate Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Flood	Fire Hose Station				1-FHA-042	Very little exposed pipe at this level.

SCENARIO TABLE

Location Name: Intermediate Building at Elevation 355'
Designator: IB-F7-/
Building: Intermediate Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Flood	Fire Hose Station	1. Pipe break.	Stairwells at Each Level	Alligator Pit		No, very likely to be discovered before serious damage.			Alligator pit is designed to handle capacity of a feedwater line break (about 300,000 gallons), so only if fire protection diesel pumps go on, does an infinite source exist that could get emergency feedwater pumps; would take long time due to small size pipe available at this level.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Alligator Pit Around the Containment
Designator: IB-FZ-8
Building: Intermediate Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF							EF-V-30A		5-31	
EF							EF-V-30C		5-31	
EF					EF-V-52	X			5-31	
EF					EF-V-55	X			5-31	
FW					FW-V-5A	X			5-31	
FW					FW-V-5B	X			5-31	
FW					FW-V-92A	X			5-31	
FW					FW-V-92B	X			5-31	

LOCATION INVENTORY CODIFICATION TABLE

Location Name:
Designator: IB-FZ 3A
Building: Intermediate Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
				No Components of Interest in this Location						

DIESEL GENERATOR BUILDING

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Diesel Generator A Building Area
Designator: DG-FA-1
Building: Diesel Generator Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EG				X					1-FHA-044	Diesel generator relay cabinet 1.
EG								EG-P-1A	1-FHA-044	Air compressor.
EG								EG-T-1A-1	1-FHA-044	Air receiver.
EG								EG-T-1A-2	1-FHA-044	Air receiver.
DF		DF-P-1A							1-FHA-044	Fuel pump.
DF		DF-P-1B							1-FHA-044	Fuel pump.
DF								DF-T-2A	1-FHA-044	Diesel fuel day tank.
EG								EG-V-1A	1-FHA-044	Diesel generator Unit A.
AH								AH-E-29A	1-FHA-044	Air supply Unit A.
EP				ESD-SGES-1D		X			5-31	
EP					X			EG-Y-1A	5-31	
EH								EH-DPESDG-1Q	5-31	

SOURCE AND MITIGATION TABLE

Location Name: Diesel Generator A Building Area
Designator: DG-FA-1
Building: Diesel Generator Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Lube Oil		1, 1-FHA-044 1-FHA-045	Automatic Wet Pipe Sprinkler System	1, 1-FHA-044 1-FHA-044	
	Fuel Oil (day tank DF-T-2A and fuel piping) Transient Material		1, 1-FHA-044 1-FHA-045	Deluge Water Spray System	1, 1-FHA-044	
			1, 1-FHA-044 1-FHA-045	Dry Chemical Fire Extinguisher	1, 1-FHA-044	Two in DG-FA-1
			1, 1-FHA-044 1-FHA-045	Fire Hose Station	1, 1-FHA-044	
			1, 1-FHA-044 1-FHA-045	Yard Hydrants	1, 1-FHA-044	Three available for additional hose protection.
			1, 1-FHA-044 1-FHA-045	Thermal Fire Detectors	1, 1-FHA-044	
			1, 1-FHA-044 1-FHA-045	Rupture Alarm for Diesel Fuel Day Tank	1, 1-FHA-044	
			1, 1-FHA-044 1-FHA-045	Walls, Doors	1, 1-FHA-044	
Missiles	Diesel Missiles		1-FHA-044 1-FHA-045	Walls	1-FHA-044 1-FHA-045	
Explosion	Diesel Explosion		1-FHA-044 1-FHA-045	Walls	1-FHA-044 1-FHA-045	
	Fuel Oil Explosion		1-FHA-044 1-FHA-045	Walls	1-FHA-044 1-FHA-045	
Flood	Deluge System Wet Pipe Sprinkler System		1-FHA-044 1-FHA-045	Walls and Doors	1-FHA-044 1-FHA-045	
Steam	Cooling Water of the Engine while Engine Is Running		1-FHA-044 1-FHA-045	Ventilation	1-FHA-044 1-FHA-045	It is judged that the worst leak cannot generate sufficiently dense steam environment for damaging equipment.

SCENARIO TABLE

Location Name: Diesel Generator A Building Area
Designator: DG-FA-1
Building: Diesel Generator Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Lube Oil Fuel Oil	1. Localized.			(See Source and Mitigation table.)	Yes.	2×10^{-2} ry^{-1}	(system)	
	Transient Material	2. Doorway.		DG-FA-2	(See Source and Mitigation table.)	No, fire has to travel a long distance.			Fire in the two relay cabinet rooms.
Missiles	Diesel Missiles	3. Confined to DG-FA-1.			(See Source and Mitigation table.)	No, part of diesel generator failure frequency.			Very unlikely to break through walls and doors to DG-FA-2.
Explosion	Diesel Explosion Fuel Oil Explosion	4. Wall failure or double door failure.		DG-FA-2	(See Source and Mitigation table.)	No, very unlikely.			Very unlikely event because walls are tornado resistant.
Flood	Deluge System	5. Doorway.		DG-FA-2	(See Source and Mitigation table.)	No, very unlikely, the gap under the doors must be left clogged for this event.			Flood severe enough to get over the curb into relay cabinet area. Unlikely because door at other end has large opening under it to let water out.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Diesel Generator B Building Area
Designator: DG-FA-2
Building: Diesel Generator Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EG				X					1-FHA-044	Diesel generator relay cabinet 2.
EG								EG-P-1B	1-FHA-044	Air compressor.
EG								EG-T-1B-1	1-FHA-044	Air receiver.
EG								EG-T-1B-2	1-FHA-044	Air receiver.
DF		DF-P-1C							1-FHA-044	Fuel pump.
DF		DF-P-1D							1-FHA-044	Fuel pump.
DF								DF-T-2B	1-FHA-044	Diesel fuel day tank.
EG								EG-V-1B	1-FHA-044	Diesel generator Unit B.
AH								AH-E-29B	1-FHA-044	Air supply Unit B.
AH								X	1-FHA-044	Backup emergency air cylinders.
EP				ESD-SGES-1E		X			5-31	
EP				X	X			EG-Y-1B	5-31	
EH								EH-DPESDG-1Q	5-31	

SOURCE AND MITIGATION TABLE

Location Name: Diesel Generator B Building Area
Designator: DG-FA-2
Building: Diesel Generator Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Lube Oil		1, 1-FHA-044 1-FHA-045	Automatic Wet Pipe Sprinkler System	1, 1-FHA-044 1-FHA-045	
	Fuel Oil		1, 1-FHA-044 1-FHA-045		1, 1-FHA-044 1-FHA-045	
	Transient Material		1, 1-FHA-044 1-FHA-045		1, 1-FHA-044 1-FHA-045	
			1, 1-FHA-044 1-FHA-045	Deluge Water Spray System	1, 1-FHA-044 1-FHA-045	
			1, 1-FHA-044 1-FHA-045	Dry Chemical Fire Extinguishers	1, 1-FHA-044 1-FHA-045	Two in DG-FA-2.
			1, 1-FHA-044 1-FHA-045	Fire Hose Station	1, 1-FHA-044 1-FHA-045	
			1, 1-FHA-044 1-FHA-045	Yard Hydrants	1, 1-FHA-044 1-FHA-045	Three available for additional hose protection.
			1, 1-FHA-044 1-FHA-045	Thermal Fire Detectors	1, 1-FHA-044 1-FHA-045	
			1, 1-FHA-044 1-FHA-045	Rupture Alarm for Diesel Fuel Day Tank	1, 1-FHA-044 1-FHA-045	
			1, 1-FHA-044 1-FHA-045	Walls	1, 1-FHA-044 1-FHA-045	
			1, 1-FHA-044 1-FHA-045	Doors	1, 1-FHA-044 1-FHA-045	
Missiles	Diesel Missiles		1-FHA-044 1-FHA-045	Walls	1-FHA-044	
	Backup Emergency Air Cylinders		1-FHA-044 1-FHA-045	Walls	1-FHA-044	
Explosion	Diesel Explosion		1-FHA-044 1-FHA-045	Walls	1-FHA-044	
	Fuel Oil Explosion		1-FHA-044 1-FHA-045	Walls	1-FHA-044	

SOURCE AND MITIGATION TABLE (continued)

Location Name: Diesel Generator B Building Area
Designator: DG-FA-2
Building: Diesel Generator Building

Page 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Flood	Deluge System Wet Pipe Sprinkler System		1-FHA-044 1-FHA-045	Walls and Door	1-FHA-044	
Steam	Cooling Water of the Engine while It Is Running		1-FHA-044 1-FHA-045	Walls and Door	1-FHA-044	Judged to be insignificant for any damage.

SCENARIO TABLE

Location Name: Diesel Generator B Building Area
Designator: DG-FA-2
Building: Diesel Generator Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Lube Oil	1. Localized.				Yes.	2×10^{-2} ry^{-1}	(system)	
	Fuel Oil Transient Material	2. Doorway.		DG-FA-1		No, very unlikely. A large open area must be enveloped in fire.			Fire in the two relay cabinet rooms.
Missiles	Diesel Missiles Backup Emergency Air Cylinders	3. Confined to DG-FA-2.				No, part of DG failure frequency.			Very unlikely to break through walls because of thick wall design and bottles harnessed by bars and chain.
Explosion	Diesel Explosion Fuel Oil Explosion	5. Wall failure or double door failure.		DG-FA-1		No, very unlikely. All concrete walls.			
Flood	Deluge System	6. Doorway.		DG-FA-2		No, very unlikely. The gap under the doors must be clogged.			Flood severe enough to get over the curb into relay cabinet area.

CONTROL BUILDING

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building Health and Physics Lab Area
Designator: CB-FA-1
Building: Control Building

Sheet 1 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU	A				MU-P-1A	X			1	
MU	A				MU-P-3A	X			1	
MU	B				MU-P-1B				1	
MU	C				MU-P-3C				1	
Emergency FW	A				EF-P-2A				1	
Emergency FW	B				EF-P-2B				1	
DC					DC-P-1A				1	
DC	B				DC-P-1B				1	
IC	A				IC-P-1A				1	
IC	B				IC-P-1B				1	
NS	A				NS-P-1A				1	
NS	B				NS-P-1B				1	
NS	C				NS-P-1C				1	
RR	A				RR-P-1A				1	
DH	A				DH-P-1A				1	
DR	A					DR-P-1A			1	
NR	A					NR-P-1A			1	
Electrical	A				4160V ES SWGR-1D				Table 3.11-16 of FHA	
	B				4160V ES SWGR-1E				Table 3.11-16 of FHA	
	A				460V AC ESV CC-1A				Table 3.11-16 of FHA	
	B				460V AC ESV CC-1B				Table 3.11-16 of FHA	
	A				460V AC Screen House ES-SWGR-1R	X			Table 3.11-16 of FHA	
MU						MU-P-2A			5-31	
MU						MU-V-12			5-31	
MU						MU-V-14A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Health and Physics Lab Area
Designator: CB-FA-1
Building: Control Building

Sheet 2 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-V-16A			5-31	
MU						MU-V-16B			5-31	
MU						MU-V-20			5-31	
MU						MU-V-36			5-31	
MU						MU-V-18			5-31	
EP	A				460V AC ES CC-1A				Table 3.11-16 of FHA	
EP	B				460V AC ES CC-1B				Table 3.11-16 of FHA	
EP	A				125/250V DC ES-1A				Table 3.11-16 of FHA	
EP	B				125/250V DC ES-1B				Table 3.11-16 of FHA	
EP	A				125/250V DC ES-DG-1P				Table 3.11-16 of FHA	
EP	B				125/250V DC ES-DG-1Q				Table 3.11-16 of FHA	
EP					125/250V DC ES-1E				Table 3.11-16 of FHA	
EP					125/250V DC ES-1F				Table 3.11-16 of FHA	
EP	A and B				120V AC Vital Distribution Panels				Table 3.11-17 of FHA	
EP	A				VBA				Table 3.11-17 of FHA	
EP	B				VBB				Table 3.11-17 of FHA	
EP	A				VBC				Table 3.11-17 of FHA	
EP	B				VBD				Table 3.11-17 of FHA	
EP					Battery Chargers				Table 3.11-17 of FHA	
EP	A				1A				Table 3.11-17 of FHA	
EP	B				1B				Table 3.11-17 of FHA	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Health and Physics Lab Area
Designator: CB-FA-1
Building: Control Building

Sheet 3 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EP	A				1C				Table 3.11-17 of FHA	
EP	B				1D				Table 3.11-17 of FHA	
EP	A				1E				Table 3.11-17 of FHA	
EP	B				1F				Table 3.11-17 of FHA	
Inverters	A				1A				Table 3.11-17 of FHA	
	B				1B				Table 3.11-17 of FHA	
	A				1C				Table 3.11-17 of FHA	
	B				1D				Table 3.11-17 of FHA	
					1E				Table 3.11-17 of FHA	
Battery	A				Battery-1A Charger				Table 3.11-17 of FHA	
	A				Battery-1C Charger				Table 3.11-17 of FHA	
	B				Battery-1B Charger				Table 3.11-18 of FHA	
	B				Battery-1D Charger				Table 3.11-18 of FHA	
AH	A				AH-E-1A					
AH	B				AH-E-1B					
AH	A				AH-E-18A					
AH	B				AH-E-18B					
MU						MU-V-4			5-31	
EF					EF-V-52	X			5-31	
EF					EF-V-53				5-31	
EF					EF-V-53				5-31	
EF					EF-V-54				5-31	
EF					EF-V-55	X			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Health and Physics Lab Area
Designator: CB-FA-1
Building: Control Building

Sheet 4 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
FW						FW-V-5A			5-31	
FW						FW-V-92A			5-31	
MS					MS-V-8A	X			5-31	
MS					MS-V-8B	X			5-31	
MS						MS-V-2A			5-31	
MS						MS-V-2B			5-31	
MS					MS-V-10A	X			5-31	
MS					MS-V-10B	X			5-31	
MS						X	MS-V-13B		5-31	
DH						DH-V-4A			5-31	
DH						DH-V-5A			5-31	
DH						DH-V-6A			5-31	
BS						BS-V-2A			5-31	
DH						DH-V-75A			5-31	
DH						DH-V-76A			5-31	
IC						IC-V-3			5-31	
IC						IC-V-4			5-31	
IC						IC-V-79A			5-31	
IC						IC-V-79C			5-31	
AH					AH-E-19B				5-31	
AH					AH-D-27A	X			5-31	
AH					AH-D-24A	X			5-31	
AH					AH-E-24B	X			5-31	
AH					AH-P-8A				5-31	
AH					AH-P-8B				5-31	
AH					AH-P-9A				5-31	
AH					AH-P-9B				5-31	
AH					AH-D-101				5-31	
AH					AH-D-102					
NS						NS-V-52A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Health and Physics Lab Area
Designator: CB-FA-1
Building: Control Building

Sheet 5 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NS						NS-V-52B			5-31	
NS						NS-V-53A			5-31	
NS						NS-V-53B			5-31	
NR						NR-V-1A			5-31	
NR						NR-V-3			5-31	
NR						NR-V-5			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-1B			5-31	
NR						NR-V-10A			5-31	
NR						NR-V-10B			5-31	
RR						RR-V-1A			5-31	
RR						RR-V-3A			5-31	
RR						RR-V-3B			5-31	
RR					RR-V-4A	X			5-31	
RR					RR-V-4B	X			5-31	
RR					RR-V-4C	X			5-31	
RR					RR-V-4D	X			5-31	
EG					X	EG-Y-1A			5-31	
EG					X	EG-Y-1B			5-31	
EP					EE-SGES	X			5-31	
EP					EE-SGES-1S				5-31	
EP					EG-ESV-1A				5-31	
EP					EG-ESV-1B				5-31	
EP					EG-ESV-1C				5-31	
EP						ESSH-1A			5-31	
EP					EG-DP-ATA				5-31	
EP					EG-DP-ATB				5-31	
EP					EG-DP-VBA				5-31	
EP					EG-DP-VBB				5-31	
EP					EG-DP-YBC				5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Health and Physics Lab Area
Designator: CB-FA-1
Building: Control Building

Sheet 6 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EP					EG-DP-VBD				5-31	
EP					EH-INV-1A				5-31	
EP					EH-INV-1B				5-31	
EP					EH-INV-1C				5-31	
EP					EH-INV-1D				5-31	
EP					EH-INV-1E				5-31	
EP					EH-DP-1A				5-31	
EP					EH-DP-1B				5-31	
EP					EH-DP-1M				5-31	
EP					EH-DPES-1E				5-31	
EP					EH-DPES-1F				5-31	
EP					EH-DPESDG-1P				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building Health Physics and Lab Area
Designator: CB-FA-1
Building: Control Building

Sheet 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
1. Fire	Cabling above False Ceiling Health Physics, Related Materials, and Equipment		Fire Hazards Report	Reinforced Concrete Walls Class A Doors to FH-FZ-2 and North Control Building Stairwell Automatic Wet Pipe Sprinkler System (yellow false ceiling) Fire Suppression Equipment at Location FH-FZ-2 Portable Dry Chemical Extinguisher Planning to Add Ionization Detection above the False Ceiling	Fire Hazards Report	The area is used for health physics related activities.
2. Smoke	See Fire Sources			Walls and Doors Mentioned in (1) HVAC Ducts to FH-FZ-2 HVAC Ducts to Upper Parts of Control Building HVAC Heat Exchanger System to Exhaust Smoke Out		

SOURCE AND MITIGATION TABLE (continued)

Location Name: Control Building Health Physics and Lab Area
Designator: C -F[illegible]-1
Building: Control Building

Sheet 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
3. Flood	Laundry and Bathroom Facilities			None		
	Laboratory Sinks			Water-Tight Doors to FH-FZ-2 and North Control Building Stairwell Walls		
4. Water Spray						Not considered because critical items in the area are cables and no other cascading effects can be identified.
5. Falling Objects						Not considered because critical items in the room are above the false ceiling.
6. Explosion	Acetylene or Propane Gas Release and Explosion in the Labs			Walls, Door, and False Ceiling; Large Floor Area		
7. Missiles	Transient Sources					

SCENARIO TABLE

Location Name: Control Building Health Physics and Lab Area
Designator: CB-FA-1
Building: Control Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire	See (1) in Source Table	1. Severe enough to damage cables above false ceiling.				Yes.	3×10^{-6} (10^{-3} in the area) x (0.1 above false ceiling) x (severity and location .03)	(comparison)	All fire scenarios also may include scenario 5. Smoke impact is not important since smoke sensitive equipment not in the area.
		2. Fire large enough to propagate outside.	Open West Door	FH-FZ-2		No.	10^{-6} (10^{-3} in the area) x (0.1 near the door) x (10^{-2} severe and affects cable in both areas)	No further analysis. Small frequency and subset of FH-FZ-2 fires since no serious failures may occur in CB-FA-1 from this fire since vital cables above concrete ceiling, not the doorway.	If any door is open off the stairwell, then the fire could spread to that level.
		3. Fire large enough to propagate outside.	Open North Door	Stairwell		No; fire in stairwell does not have any impact on equipment.			
		4. Fire large enough to put hot gases into HVAC system (not an explosion).	HVAC Ducts	FH-FZ-2 Other Points of Control Building		No; subset of the preceding scenarios.			If pressure of hot gases is not large, the hot gases will end up (see Drawing C-302-842) in the auxiliary building exhaust fan system.
Smoke	See (1) in Source Table	5. Smoke to fuel handling building and control building.	Doorways	FH-FZ-2 CB-FA-1		No.			Impact on cables not important.
Flood	See (3) in Source Table	6. Flood is large enough to travel to FH-FZ-6.	Door Grating from FH-FZ-2	FH-FZ-2 FH-FZ-6		Yes.	10^{-5} (10^{-2} flood) x (10^{-3} not detected)	(CB-HVAC)	$17 m^3$ in FH-FZ-6 would damage chiller pumps (see Source Table) for FH-FZ-6.
			Door	North Stairwell					If flood spills equally in two directions (FH-FZ-2 and FH-FZ-6), it will take more than an hour at 100 gpm to damage chiller pumps.
			Door from Stairwell	FH-FZ-6					

SCENARIO TABLE (continued)

Location Name: Control Building Health Physics and Lab Area
Designator: CB-FA-1
Building: Control Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Explosion	See (6) in Source Table	7. Acetylene or propane line leak in secondary plant sampling room.	Door and HVAC Ducts	All of CB-FA-1		Yes.	3×10^{-5} ry^{-1} $(3 \times 10^{-4}$ for explosion to occur) $\times$ (0.1 for severity)	No action (same impact as scenario 1, but smaller frequency).	Fails false ceiling - sets out a large fire that puts hot gases under the ceiling.
Missiles	Not considered as likely events in the area.								

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building 1P Switchgear Room
Designator: CB-FA-2a
Building: Control Building

Sheet 1 of 4

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RR	A				RR-P-1A	X				
DH	A				DH-P-1A	X				
NR	A					NR-P-1A				
EP				MG Set	X				1-FHA-035	
EP	A			1A 480V AC ENG SFCC	X	X			1-FHA-035	
EP	A			1P 480V SWGR ENG SFGD	X				1-FHA-035	
AH						X		AH-D-28		Fails closed on loss of air, which is insignificant.
AH						X		AH-D-31G		Fails closed on loss of air, which is significant.
EF	A				EF-P-2A				1	
DC	A					DC-P-1A			1	
DC	A				DC-P-1A				1, 5-31	
IC	A					IC-P-1A			1	
IC	A				IC-P-1A				1	
MU	A				MU-P-1A				1	
MU	A					MU-P-2A				
MU	A				MU-P-3A	MU-P-3A				
MU	B				MU-P-1B	X				
NS	A				X	NS-P-1A			1	
NS	B					NS-P-1B			1	
NS					NS-P-1A				1	
Electrical	A				4160V ES SWGR-1D				Table 3.11-16 of FHA	
	A				480V AC Screen House SWGR 1R	X			Table 3.11-16 of FHA	
EP	A				480V AC ESVCC-1A				Table 3.11-16 of FHA	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 1P Switchgear Room
Designator: CB-FA-2a
Building: Control Building

Sheet 2 of 4

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EP	A				125/250V DC ES-DG Distribution Panel 1P				Table 3.11-16 of FHA	
EP	A				125V AC Vital Distribution Panel VBA				Table 3.11-16 of FHA	
EP					120V AC VBC				Table 3.11-16 of FHA	
AH	A				AH-E-1A	AH-E-1A			Table 3.11-22 of FHA	
AH	A				AH-E-18A	AH-E-18A		Supply Duct for the Fans Dedicated to This Floor (AH-E95A,B)	F-311-892	
DR	A					DR-P-1A			Table 3.11-28 of FHA	
MU						MU-V-14A			5-31	
MU						MU-V-16A			5-31	
MU						MU-V-16B			5-31	
MU						MU-V-36			5-31	
MU						MU-V-3			5-31	
MU						MU-V-4			5-31	
EF					EF-V-2A	X			5-31	
EF						EF-V-30A			5-31	
EF						EF-V-30C			5-31	
EF					EF-V-52	X			5-31	
EF					EF-V-55	X			5-31	
AH					X	AH-D-27A			5-31	
AH					X	AH-D-24A			5-31	
AH					AH-P-8A				5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 1P Switchgear Room
Designator: CB-FA-2a
Building: Control Building

Sheet 3 of 4

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH					AH-P-8B				5-31	
AH						X		AH-D-28	5-31	
AH						X		AH-D-30G	5-31	
AH						X		AH-D-31G	5-31	
AH						AH-D-41A			5-31	
AH					AH-D-101	X			5-31	
NS						NS-P-1C			5-31	
NS						NS-V-52A			5-31	
NS						MS-V-53A			5-31	
NR						NR-V-1A			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-3			5-31	
NR						NR-V-5			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-18			5-31	
NR						NR-V-10A			5-31	
NR						NR-V-10B			5-31	
DR						DR-V-1A			5-31	
FW						FW-V-5A			5-31	
FW						FW-V-92A			5-31	
MS						MS-V-8A			5-31	
MS						MS-V-8B			5-31	
MS							MS-V-4A		5-31	
MS						MS-V-2A			5-31	
MS						MS-V-2B			5-31	
MS						MS-V-10A			5-31	
MS						MS-V-13A	X		5-31	
DH						DH-V-4A			5-31	
DH						DH-V-5A			5-31	
DH						DH-V-6A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 1P Switchgear Room
Designator: CB-FA-2a
Building: Control Building

Sheet 4 of 4

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
BS						BS-V-3A			5-31	
BS						BS-V-2A			5-31	
DH						DH-V-75A			5-31	
DH						DH-V-76A			5-31	
IC						IC-P-1B			5-31	
IC						IC-V-3			5-31	
RR							RR-V-1A		5-31	
RR							RR-V-3A		5-31	
RR						RR-V-4A	X		5-31	
RR						RR-V-4C	X		5-31	
EP						X	EG-Y-1A		5-31	
EP						ESV-1A			5-31	
EP						ESV-1C			5-31	
EP						EG-CCESSH-1A			5-31	
EP						EH-INV-1A			5-31	
EP						EH-INV-1C	X		5-31	
EP						EH-INV-1E			5-31	
EP						Battery Charger 1A			5-31	
EP						Battery Charger 1C			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building 1P Switchgear Room
Designator: CB-FA-2a
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Class A Doors	Fire Hazards Report	
	Control Center (1A 480V ENG.SFCC)			Reinforced Concrete Walls (three) and Metal Panel (one)		
	Switchgear (1P 480V Switchgear ENG SFG0)			HVAC Duct Smoke Detectors		
	Motor Generator Set			Location FH-FZ-5		
				Fire Hose Protection		
				Portable CO_2 Extinguishers		
				Stair Tower		
				Portable CO_2 Extinguisher		
				Dry Chemical Extinguisher		
Missiles	Motor Generator Set Catastrophic Failure			Walls		
	Transient Sources					

SCENARIO TABLE

Location Name: Control Building 1P Switchgear Room
Designator: CB-FA-2a
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling Switchgear or Transient Fuel	1a. Confined.	Proximity	Adjacent Equipment		Yes, but needs independent events.	1×10^{-3} (3 x 10^{-3}/yr fire) x (0.3 spurious signal)	(comparison)	Smoke could get throughout control building and fuel handling building via ventilation if ventilation is not stopped, or fire dampers fail to close.
		1b. Large fire.	HVAC Ducting	CB-FA-2b		No.	$< 10^{-5}$		Fire does not fail the intake ducts of HVAC system. (See impact table.)
		2. Engulfing.	Closed Doors	Does Not Propagate		No, a subset of scenario 1.	3×10^{-4} ry^{-1}		Outlet damper closes, inlet damper fails to close, smoke backs up, spills into other zone.
		3. Engulfing.	Open West Door for Fire Fighting	CB-FA-2b		Yes.	9.0×10^{-5} (3 x 10^{-3} x (0.3 geometric factor) x (0.5 nonsuppression) x (0.2 severity factor)	(comparison)	Fire affect cables above switchgear.
		4. Engulfing.	Open South Door	CB-FA-2d		Yes.	1.5×10^{4}	(comparison)	Impact almost the same as scenario 1a because cabinets in CB-FA-2d are not readily susceptible to smoke.
Missiles	Catastrophic Failure of a Motor Generator Set, or Transient Sources	5. Missile hits the HVAC ducts and damages them.				No, unlikely event.	(Failure to run - 1.8×10^{-3} missile generation $\sim 10^{-2}$ $\sim 10^{-5}$)		Intake duct of HVAC system is damaged.

NOTE: FHA claims fire dampers at every opening. HVAC drawings indicate no dampers between CB-FA-2a and CB-FA-2b. Smoke could involve both rooms if there are no fire dampers.

IMPACT TABLE

Location Name: 1P Switchgear Room
Designator: CB-FA-2a
Building: Control Building

Scenario Summary: Fire, Scenario 1a

System Cost	Components Affected by the Hazard
NR/All	Spurious closure of NR-V-5; also, other NR-related equipment is affected.
MU/A and MU/B	MU-P-1A and MU-P-1B power and control cable and MU-V-14A, MU-V-16A, and MU-V-16B.
EF/2A	EF-P-2A power cable.
DC/A	DC-P-1A control cable.
IC/All	IC-P-1A, IC-P-1B, and IC-V-3 control cable; power cable for IC-P-1A.
NS/All	Power to NS-P-1a; control to NS-P-1B and 1C.
	RR valves RR-V-1A, RR-V-3A, RR-V-4A, and RR-V-4C are normally open, and the fire would fail them as they are.
RR/A	Power cable for RR-P-1A.
DH/A	Power cable for DH-P-1A; control for DH-V-4A.
DR/A	Control cable for DR-P-1A, DR-V-1A (normally open).
AH/1A	Power and control cable for AH-E-1A.
Train A of Electric Power	Power cables to 1D, 1R, 1A, and 1P switchgear.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building 1S Switchgear Room
Designator: CB-FA-2b
Building: Control Building

Sheet 1 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU					MU-P-1C	MU-P-1A			5-31	MU-P-1A cable to be protected.
MU					MU-P-1B	MU-P-1B			5-31	
MU					X	MU-P-2C			Table 3.11-19 of FHA	
MU					MU-P-3C	MU-P-3C			Table 3.11-20 of FHA	
EP				1B 480V ENG SFCC	X	X			1-FHA-035	
EP					SGES-1P				1-FHA-035	To be protected
EP	A			1S 480V SWGR ENG SFGD	X				1-FHA-035	
EP	B				4160V ES SWGR 1E					
EP	B				480V SH ES SWGR 1T					
EP	B				480V ESV CC-1B					
EP					125/250V DC ES DG Distribution Panel 1Q				Table 3.11-16	
EP					120V AC Vital Distribution Punel VBB				Table 3.11-17	
EP					120V AC Vital Distribution Panel VBD				Table 3.11-17	
AH					X	X		AH-E-95A	E-311-842	Booster fan control building ventilation for second floor.
AH					X	X		AH-E-95B	E-311-842	Booster fan control building ventilation for second floor.
AH						X		AH-D-101		Fail closed on loss of air, which is significant for second floor.
AH						X		AH-D-102		Fail closed on loss of air, which is significant for second floor.
AH						X		AH-D-30F		Fail closed on loss of air.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 1S Switchgear Room
Designator: CB-FA-2b
Building: Control Building

Sheet 2 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH						X		AH-D-30G		Fail closed on loss of air.
AH						X		AH-D-31E		Fail closed on loss of air.
AH						AH-E-1B			Table 3.11-22	
AH					AH-E-18B	AH-E-18B				
AH					Electrical Train C				Cable Tray Drawings	
EP					125V DC Vital Distribution Panel VBD				Table 3.11-17	
DC					DC-P-1B	X			5-31	
DC						DC-P-1A			5-31	Will be protected.
IC					IC-P-1B	X			1	
IC						IC-P-1A			5-31	Will be removed from this location.
NS					NS-P-1B				1	
NS						NS-P-1A			5-31	
NS						NS-P-1B			5-31	
NS						NS-P-1C			5-31	
RR					RR-P-1B				1	
DH					DH-P-1B				1	
DR						DR-P-1B			Table 3.11-28	
DR						DR-P-1A			5-31	To be protected.
DR						DR-V-1B			5-31	
RR						RR-P-1A			5-31	
RR						RR-V-1B			5-31	
RR						RR-V-3B			5-31	
RR						RR-V-4A			5-31	
RR					RR-V-4B				5-31	
RR						RR-V-4C			5-31	
RR					RR-V-4D	X			5-31	
EP						EG-Y-1A			5-31	To be protected.
EP					X	EG-Y-1B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 1S Switchgear Room
Designator: CB-FA-2b
Building: Control Building

Sheet 3 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EP					X	X	EE-SGES-1P		5-31	
EP					ES-V-16				5-31	
EP					ES-V-1C				5-31	
EP					EG-CCESV-1C				5-31	
EP						EG-CCESSH-1B			5-31	
EP					EH-INV-1B	X			5-31	
EP					EH-INV-1D				5-31	
EP					EH-BC-1B				5-31	
EP					EH-BC-1D				5-31	
AH					AH-E-24B	X			5-31	
AH					AH-P-9A		X		5-31	
AH					AH-P-9B		X		5-31	
AH						AH-D-28			5-31	Fail closed on loss of power, single element cutset for CB-HVAC fault tree.
AH						AH-D-30C			5-31	
AH						AH-D-31C			5-31	
AH						AH-D-30E			5-31	
AH						AH-D-31E			5-31	
AH						X		AH-D-30F	5-31	
AH						X		AH-D-31F	5-31	
AH						X		AH-D-30G AH-D-31G	5-31	
AH						AH-D-41A			5-31	Fail closed on loss of power; main supply isolation dampers.
AH						AH-D-41B			5-31	Fail closed on loss of power; main supply isolation dampers.
AH					X	X		AH-D-101	5-31	Fail closed on loss of power; main supply isolation dampers.
AH					X	X		AH-D-102	5-31	
NS						NS-V-52B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 1S Switchgear Room
Designator: CB-FA-2b
Building: Control Building

Sheet 4 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet*	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
KS						NS-V-53B			5-31	
NR						NR-P-1A			5-31	
NR						NR-P-1C			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-5			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-4B			5-31	
NR						NR-V-1B			5-31	
NR						NR-V-15B			5-31	
MS						X	MS-V 13B		5-31	
DH						DH-P-1A			5-31	
DH						DH-V-4A			5-31	Injection MOV.
DH						DH-V-4B			5-31	Injection MOV.
DH						DH-V-5A			5-31	BWST Suction MOV.
DH						DH-V-5B			5-31	BWST Suction MOV.
DH						DH-V-6B			5-31	
BS						BS-V-3A			5-31	
BS						BS-V-3B			5-31	
BS						BS-V-2B			5-31	
DH						DH-V-75A			5-31	
DH						DH-V-76A			5-31	
IC						IC-V-1B			5-31	
IC						IC-V-2			5-31	
IC						IC-V-3			5-31	
IC						IC-V-4			5-31	
AH						AH-E-1A			5-31	
AH						AH-E-18A			5-31	To be protected.
AN					X	AH-E-19B			5-31	
MU						MU-V-12			5-31	
MU						MU-V-14A			5-31	To be protected.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 1S Switchgear Room
Designator: CB-FA-2b
Building: Control Building

Sheet 5 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-V-14B			5-31	
MU						MU-V-16A			5-31	
MU						MU-V-16B			5-31	To be protected.
MU						MU-V-16C			5-31	
MU						MU-V-16D			5-31	
MU						MU-V-18			5-31	
MU					X	MU-V-20			5-31	
MU						MU-V-36			5-31	
MU						MU-V-37			5-31	
MU						MU-V-1B			5-31	
MU						MU-V-2A			5-31	
MU						MU-V-2B			5-31	
MU						MU-V-3			5-31	
EF					EF-P-2B				5-31	
EF						EF-V-2B			5-31	
EF						X	EF-V-30B		5-31	
EF							EF-V-30D		5-31	
EF					EF-V-53	X			5-31	
EF					EF-V-54	X			5-31	
MS						MSV-8A			5-31	
MS						MSV-8B			5-31	
MS							MSV-4A		5-31	
MS						MSV-2B			5-31	
MS						MSV-10B			5-31	
EP					EH-BC-1B				5-31	
EP					EH-BC-1D				5-31	
EP					EH-DP-1M				5-31	
MU						MU-P-3B			5-31	
MU						MU-P-3C			5-31	
EF					EF-P-2B				5-31	
EF						EF-V-2B			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building 1S Switchgear Room
Designator: CB-FA-2b
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls (two) and Metal Panel (one)	Fire Hazards Report	
	Control Center (1B 480V ENG SFCC or AC transient switchgear for ICVIVS)			Class A Doors		
				HVAC Duct Smoke Detectors		
	Switchgear (1S 480V switchgear ENG SFGD)			Location FH-FZ-5		
				Fire Hose Protection		
				Portable CO_2 Extinguisher		
				Stairwell		
				Portable CO_2 Extinguisher		
				Dry Chemical Extinguisher		
Missile	Transient Sources					

SCENARIO TABLE

Location Name: Control Building 15 Switchgear Room
Designator: CB-FA-2b
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Electrical cabinet or cables.	1a. Confined to electrical cabinet and above.	Proximity	Adjacent Cables		Yes.	2×10^{-5} (3×10^{-3} fire in the cabinet)x (0.2 failure to suppress) x (0.03 severity)	(comparison)	May involve entire room via smoke, heat. Cabinet is the main contributor to fire occurrence. Severity mainly because cables outside the cabinet have to fail and cables for some train A equipment will be protected by barriers. Smoke could get throughout control building, especially second floor and fuel handling building, via ventilation. May hamper fire fighting efforts. (See impact table.)
		1b. Large fire.	HVAC Ducting	CB-FA-2a		No.	$< 10^{-5}$		Smoke, two dampers fail.
		2. Engulfing.	Closed Doors	No Propagation to Other Zones		Yes.	3×10^{-5}	No action (subset of scenario 1a).	
		3. Engulfing.	Open East Door	CB-FA-2a		Yes.	3×10^{-6} $\times 10^{-1}$ 3×10^{-4} x (0.1 open east door to fight fire; not most likely path) x (0.1 smoke damage to cabinets)	(comparison)	Smoke damage to 480V switchgear is deemed to be unlikely (estimated 0.1 for severity factor to cause damage)
		4. Engulfing.	Open West Door	CB-FA-2c		No (fire growth very unlikely).	$< 10^{-5}$		Unlikely fire growth, little smoke effect (mainly cables).
Missiles	Not considered as likely.								

IMPACT TABLE

Location NAme: Control Building 1S Switchgear Room
Designator: CB-FA-2b
Building: Control Building

Scenario Summary: Fire; Scenario 1a; Fire Fails Cables and Switchgear within This Room

Sheet 1 of 2

System Cost	Components Affected by the Hazard
MU/All	Power cables for pumps MU-P-1A and MU-P-1B affected, control cable for pump MU-P-1C affected; the power cable for MU-P-1A will be protected.
ESV/B, ESV/1E, ESV/1T, ESV/1S	480V load centers ESV-1B, ESV-1S, ESV-1E, and ESV-1T.
ESV/C	480V ESV-1C.
Control Building HVAC	AH-D-101, AH-D-102, AH-E-95A, AH-E-95B, AH-D-41A, AH-D-41B, and AH-D-28 (isolation damper on single air duct).
DC/All	Power and control cables of DC-P-1B; control cable of DC-P-1A (will be protected).
IC/All	Cables for both pumps in the area (control cable for IC-P-1A will be protected).
NS/All	Power and control cables for NS-P-1B; control for NS-P-1A (to be protected); control for NS-P-1C; some additional NS values in the area.
RR/All	Control cable for RR-P-1A (to be protected; power cable for RR-P-1B.
DH/All	DHR pumps DH-P-1A and DH-P-1B or valves DH-V-4A and DH-V-4B.
DR/All	DR-P-1A control cable (to be protected); DR-P-1B control cables.

IMPACT TABLE

Location Name: Control Building 1S Switchgear Room
Designator: CB-FA-2b
Building: Control Building

Scenario Summary: Fire; Scenario 1a; Fire Fails Cables and Switchgear within This Room

Sheet 2 of 2

System Cost	Components Affected by the Hazard
NR/All	Hot short in control cable for NR-V-5. Also several valves and two pumps from this system are affected.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building Technical Support Center Area
Designator: CB-FA-2c
Building: Control Building

Sheet 1 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU					MU-P-1C				1	
RR					RR-P-1B					
DH					DH-P-1B					
DR						DR-P-1B			1	
NR						NR-P-1C				
NR			RCP	X PWR Monitor Rack A		X			1-FHA-035	
NR				RCP PWR Monitor Rack B	X		X		1-FHA-035	
AH						X		AH-D-30E		Fails closed on loss of air into the spare.
DC Buses				1M						(ME-11)
EP					480V SH [illegible]-SWGR 1T	X				
EP				AC Transfer Switch for 1C Valves MCC						
MU						MU-P-1B			5-31	
MU						MU-P-3B			5-31	
MU						MU-P-3C			5-31	
MU						MU-V-12			5-31	
MU						MU-V-14B			5-31	Normally closed MOV; BWST section.
MU						MU-V-16C			5-31	C-pump discharge; normally closed MOV.
MU						MU-V-16D			5-31	
MU						MU-V-18			5-31	
MU					X	MU-V-20			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Technical Support Center Area
Designator: CB-FA-2c
Building: Control Building

Sheet 2 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-V-36			5-31	
MU						MU-V-37			5-31	
MU						MU-V-1B			5-31	
MU						MU-V-2A			5-31	
MU						MU-V-2B			5-31	
MU						MU-V-3			5-31	
EF						EF-V-30A			5-31	
EF						X	EF-V-30B		5-31	
EF							EF-V-30D		5-31	
EF						EF-V-53			5-31	
EF						EF-V-54			5-31	
MS						MS-V-8A			5-31	
MS						MS-V-8B			5-31	
MS							MS-V-4A		5-31	
DH						DH-V-4B			5-31	
DH						DH-V-5B			5-31	
DH						DH-V-6B			5-31	
BS						BS-V-3B			5-31	
BS						BS-V-2A			5-31	
BS						BS-V-2B			5-31	
AH						AH-D-28			5-31	
AH							AH-D-30E		5-31	
AH							AH-D-31E		5-31	
AH						AH-D-3B			5-31	
AH						AH-D-41B			5-31	
AH						AH-E-1C			5-31	
AK						AH-E-18B			5-31	
AH						AH-E-19B			5-31	
AH						AH-P-9A			5-31	
AH						AH-P-9B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Technical Support Center Area
Designator: CB-FA-2c
Building: Control Building

Sheet 3 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NS						NS-P-1C			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-4B			5-31	
NR						NR-V-1B			5-31	
NR						NR-V-10B			5-31	
NR						NR-V-15B			5-31	
DC						DC-P-1B			5-31	
DR						DR-V-1B			5-31	
RR						RR-V-1B			5-31	
EP							EG-SEC-1C		5-31	
EP					EG-CCESV-1C				5-31	
EP						EG-CCESSH-1B			5-31	
EP						EC-INV-1B			5-31	
EP						EH-INV-1D			5-31	
EP					X	X		EH-DP-1M	5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building Technical Support Center Area
Designator: CB-FA-2c
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete (three) and Metal Panel (one)	Fire Hazards Report	
	Reactor Coolant Pump PWR Monitor Racks A and B			Class A - Rated Doors (four) and Class B Rated Door (one)		
	Decay Coolant Panel 1M			HVAC Duct Smoke Detectors		
	Loose Parts Monitoring Panel			Location FH-FZ-5		
				Fire Hose Protection		
				Portable CO_2 Extinguisher		
	Decay Coolant Transfer Switch for 1M			Stairwell		
				Portable CO_2 Extinguisher		
				Dry Chemical Extinguisher		
				Technical Support Center Surrounded By Automatic Halon Fire Suppression System Actuated by Ionization Fire Detector Located inside Support Center Area		
Missile	Transient Sources					

SCENARIO TABLE

Location Name: Control Building Technical Support Center Area
Designator: CB-FA-2c
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	1a. In situ or transient fuel confined.	Proximity	Adjacent		Yes	1.5×10^{-5} (10^{-3}/yr fire) (0.5 non-suppression) x (0.1 severity) x (0.3 geometric factor)	(comparison)	Smoke could get throughout control building and fuel handling building via ventilation and hamper fire fighting. (See impact table.)
		1b. Large fire.	HVAC	CB-FA-2b		Yes.	$< 10^{-5}$	No action (very unlikely for additional damage).	Smoke; two dampers fail.
		2. Engulfing.	Closed Doors	Incapable of Propagation		Yes.	10^{-4}	No action (subset of scenario 1).	
		3. Engulfing.	Open East Door	CB-FA-2b		Yes.	$< 10^{-5}$	No action (very unlikely for flames or hot gas damage equipment in CB-FA-2b).	
		4. Engulfing.	Open North Door	Stairwell		No.	$< 10^{-5}$		Nothing to damage.
		5. Engulfing.	Open West Door	FH-FZ-5		Yes.	$< 10^{-5}$	No action (smoke damage of little significance).	Smoke dilution; rising.
		6. Engulfing.	Open South Door	CB-FA-2e		Yes.	$< 10^{-5}$	(comparison)	
Missiles	Not considered as likely.								

IMPACT TABLE

Location Name: Technical Support Center Area
Designator: CB-FA-2c
Building: Control Building

Scenario Summary: Fire; Scenario 1a

System Cost	Components Affected by the Hazard
MU/B and MU/C	Control cables for normally closed BWST suction valve MU-V-14B; control cables for discharge valves for pump C, MU-V-16C, and MU-V-16D (normally closed); power and control cables for MU-P-1B and MU-P-1C.
EP/1T	Power cable for 1T-switchgear for screen house. Inverters B and D.
Train B	Train B of RR, DH, DR, NR.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building, East, Battery Charger Area
Designator: CB-FA-2d
Building: Control Building

Sheet 1 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
IC	A					IC-P-1A			5-31	To be protected.
IC	B					IC-P-1B			5-31	
EP	A			Inverter 1A	X				1, 1-FHA-035	
EP				Inverter 1C	X				1, 1-FHA-035	
EP				Inverter 1E	X				1, 1-FHA-035	
EP				AC Distribution Panels (120V) VBA and VBC	X				1, 1-FHA-035	
EP				Battery Chargers 1A, 1C, 1E Conflict with Table 3.11-17 (says chargers 1A, 1B, 1E)	X				1, 1-FHA-035	
EP				DC Main Distribution Panel 1A					1-FHA-035	
EP				DG distribution Panel (1P) Table 3.11-17 Also Shows 1E						Not shown on electrical system description and FHA-035.
EP					Battery 1A/1C (Table 3.11-17)					
AH	B					X		AH-D-30D		Fail closed on loss of air, which is significant.
AH	A									
AH	B					X		AH-D-31C AH-D-31D		Fail closed on loss of air, which is significant.
AH	A					AH-E-1A				
AH	B				AH-E-1B	AH-E-1B			3.11-22	
AH	A					AH-E-18A			3.11-22	

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LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building, East, Battery Charger Area
Designator: CB-FA-2d
Building: Control Building

Sheet 2 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH	B					AH-E-18B			3.11-22	To be protected.
DR	B					DR-P-1B				To be protected.
NS	A					NS-P-1A			5-31	
NS	B					NS-P-1B			5-31	
NS	C					NS-P-1C			5-31	To be protected.
NR	A					NR-P-1A				
NR	C					NR-P-1C				
DC	A					DC-P-1A				
DC	B					DC-P-1B				To be protected.
MU	C					MU-P-2C			3.11-19	
MU	A					MU-P-3A			3.11-19	
MU	C					MU-P-3C			3.11-20	To be protected.
MU						MU-P-1A				
MU						MU-P-3B				
MU						MU-V-12				
MU						MU-V-14A				
MU						MU-V-14B				To be protected.
MU						MU-V-16A				
MU						MU-V-16B				
MU						MU-V-16C				To be protected.
MU						MU-V-16D				
MU						MU-V-18				
MU						MU-V-217				
MU						MU-V-20				
MU						MU-V-37				To be protected.
MU						MU-V-1B				
MU						MU-V-8				
MU						MU-V-6A				
MU						MU-V-6B				
MU						MU-V-11A				
MU						MU-V-11B				

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building, East, Battery Charger Area
Designator: CB-FA-2d
Building: Control Building

Sheet 3 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
Instrumentation	A and B						More Than 50% of Instrumentation			
ICS						X	X			
EF						EF-V-2A			5-31	
EF						EF-V-2B			5-31	
EF						EF-V-30A			5-31	
EF							EF-V-30C		5-31	
EF						EF-V-52			5-31	
EF						EF-V-53			5-31	
EF						EF-V-54			5-31	
EF						EF-V-55			5-31	
EF					EF-HSPS-A				5-31	
[illegible]					EF-HSPS-C				5-31	
FW						FW-V-5A			5-31	
FW						FW-V-92A			5-31	
MS						MS-V-8A			5-31	
MS						MS-V-8B			5-31	
MS					MS-V-4A				5-31	
MS					MS-V-4B				5-31	
MS					MS-V-10A				5-31	
MS						MS-V-10B			5-31	
MS						MS-V-13A			5-31	
MS						MS-V-13B			5-31	
DH						DH-V-4A			5-31	
DH						DH-V-4B			5-31	
DH						DH-V-5A			5-31	
DH						DH-V-5B			5-31	
DH						DH-V-6B			5-31	
BS						BS-V-3A			5-31	
BS						BS-V-3B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building, East, Battery Charger Area
Designator: CB-FA-2d
Building: Control Building

Sheet 4 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
BS						BS-V-2B			5-31	
IC						IC-V-1A			5-31	
IC						IC-V-1B			5-31	
IC						IC-V-3			5-31	
IC						IC-V-4			5-31	To be rerouted.
AH						AH-P-9A			5-31	Both cables to be rerouted
AH						AH-P-9B			5-31	Both cables to be rerouted.
AH						AH-D-30A			5-31	
AH						AH-D-31A			5-31	
AH						X	AH-D-30C		5-31	
AH						X	AH-D-31C		5-31	
AH						X	AH-D-30D		5-31	
AH						X	AH-D-31D		5-31	
AH						AH-D-41B			5-31	
AH						AH-D-102			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-4B			5-31	
NR						NR-V-6			5-31	
NR						NR-V-18			5-31	To be protected.
NR						NR-V-15A			5-31	To be protected.
NR						NR-V-15B			5-31	To be protected.
DR						DR-P-1A			5-31	
DR						DR-V-1B			5-31	To be protected.
RR						RR-V-1B			5-31	
RR						RR-V-3A			5-31	
RR						RR-V-3B			5-31	
RR						RR-V-4A			5-31	
RR						RR-V-4B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building, East, Battery Charger Area
Designator: CB-FA-2d
Building: Control Building

Sheet 5 of 5

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RR						RR-V-4C			5-31	
RR						RR-V-4D			5-31	
EP						ED-SGES-1D			5-31	Assume to be protected.
EP						ED-SGES-1E			5-31	
EP						EG-Y-1A			5-31	
EP						EE-SGES-1P			5-31	Assume to be protected.
EP						EE-SGES-1S			5-31	Assume to be protected.
EP						EE-SGESSH-1R			5-31	Assume to be protected.
EP						EE-SGESSH-1T			5-31	Assume to be protected.
EP						EG-SEC-1C			5-31	
EP						EG-CCESSH-1B			5-31	
EP					X		EG-DP-ATA		5-31	
EP					X		EG-DP-ATB		5-31	
EP					X		EG-DP-VBA		5-31	
EP					X		EG-DP-VBC		5-31	
EP					X		EH-INV-1A		5-31	
EP					X	X	EH-INV-1C		5-31	
EP					EH-INV-1D				5-31	
EP					X		EH-INV-1E		5-31	
EP					X		EH-BC-1A		5-31	
EP					X		EH-BC-1C		5-31	
EP					X		EH-DP-1A		5-31	
EP					X	X	EH-DP-1M		5-31	
EP					X		EH-DPES-1E		5-31	
EP							EH-DPESDG-1P		5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building East Battery Charger Area
Designator: CB-FA-2d
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls (two) and Metal Panels (two)	Fire Hazard Report	
	Inverters (1A, 1C, and 1E)					
	AC Distribution Panels (VBA and VBC)			Class A - Rated Doors		
	Battery Chargers (1A, 1B, 1E)			HVAC Duct Smoke Detectors		
	DC Main Distribution Panel			Portable Dry Chemical Extinguisher		
	Diesel Generator Distribution Panel					
Missile	Transient Sources					

SCENARIO TABLE

Location Name: Control Building East Battery Charger Area
Designator: CB-FA-2d
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	1 Confined. Cable or cabinet burning due to an electrical short or transient fuel.	Proximity	Adjacent Equipment		Yes.	5 x 10^{-6} (3 x 10^{-3}/year fire) x (0.3 geometry) x (0.2 failure to suppress) x (0.03 severity)	(comparison)	Smoke could get throughout control building and fuel handling building via ventilation. See impact table.
		2. Engulfing.	Closed Doors	Incapable of Propagating Outside		Yes.	< 10^{-5}	No action (subset of scenario 1).	
		3. Engulfing.	Open West Door	CB-FA-2e		Yes.	2 x 10^{-6} (3 x 10^{-3}/year) x (0.01 severity factor for propagation through open door) x (0.3 geometric factor) x (0.2 failure to suppress)	(comparison)	Loss of all instruments. Loss of all inverters.
		4. Engulfing.	Open North Door	CB-FA-2a		Yes.	2 x 10^{-6} (3.0 x 10^{-3}/ year) x (0.03 severity factor) x (0.3 door is open and smoke damage) x (0.3 geometric factor) x (0.2 failure to suppress)	(comparison)	Smoke propagation (door opened to fight fire).
		5. Engulfing.	Open South Door	CB-FA-2f		Yes.	< 10^{-5}	No action (additional failures not important).	Little effect of smoke.
Missiles									Not considered as likely.

IMPACT TABLE

Location Name: East Battery Charger Area
Designator: CB-FA-2d
Building: Control Building

Scenario Summary: Fire; Scenario 1

System Cost	Components Affected by the Hazard
NS/All	Control cables for all three pumps; cable for NS-P-1C to be protected.
NR/A, NR/C	Control cables for NR-P-1A and NR-P-1C and several NR valves.
DC/All	Control cables for DC pumps; DC-P-1B cable to be protected.
MU/All	Control cables for all MU pumps and several valves; train C-related cables to be protected.
IC/All	Control cables for IC-P-1A and IC-P-1B (to be protected) and IC-V-3 and IC-V-4; cables for IC-P-1B and IC-V-4 to be protected.
EP/All	Control cables to several vital buses.
DC/A	Charger and inverter for train A DC loads.
CB/HVAC	Control cables for several components; cable for AH-E-18B to be protected.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building, West, Battery Charger Area
Designator: CB-FA-2e
Building: Control Building

Sheet 1 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EP				Inverter 1B	X				1-FHA-035	
EP				Inverter 1D	X				1-FHA-035	
EP				DC Distribution Panel 1B	X				1, 1-FHA-035	
EP				DG Distribution Panel 1Q	X				1	Not shown in electric system description and FHA-035.
EP				DC Main Distribution Panel 1B	X				1, 1-FHA-035	
EP				125V/250V DC ES Distribution Panel 1F					3.11-16	
EP				Battery Chargers 1C, 1D, 1F	X				1, 1-FHA-035	
EP				120V AC Vital Distribution Panels VBB and VBD					3.11-18	
EP					Batteries B/D				3.11-18	
AH						X		AH-D-30E		Fails closed on loss of air in the technical support center.
MU						MU-P-3A			3.11-19	
						MU-P-3B			5-31	
MU						MU-V-217			5-31	
MU						MU-V-1A			5-31	
MU						MU-V-8			5-31	
MU						MU-V-6A			5-31	
MU						MU-V-6B			5-31	
MU						MU-V-11A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building, West, Battery Charger Area
Designator: CB-FA-2e
Building: Control Building

Sheet 2 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-V-11B			5-31	
EF							EF-V-30B		5-31	
EF							EF-V-30C		5-31	
EF					EF-HSPS-A				5-31	
EF					EF-HSPS-B				5-31	
NR						NR-V-5			5-31	
NR						NR-V-6			5-31	
NR						NR-V-15A			5-31	
NR						NR-V-15B			5-31	
EP						ED-SGES-1E			5-31	
EP						EE-SGES-1S			5-31	
EP						EE-SGESSH-1T			5-31	
EP						EG-SEC-1C			5-31	
EP					EG-DP-ATB				5-31	
EP					X		EG-DP-VBB		5-31	
EP					X		EP-DP-VBD		5-31	
EP					X	X	EH-INV-1B		5-31	
EP					X	X	EH-INV-1D		5-31	
EP					X		EH-BC-1B		5-31	
EP					X		EH-BC-1D		5-31	
EF					EF-HSPS-C				5-31	
EF					EF-HSPS-D				5-31	
MS						MS-V-8A			5-31	
MS							MS-V-4B		5-31	
MS					MS-V-10B				5-31	
IC						IC-P-1B			5-31	
IC						IC-V-1A			5-31	
IC						IC-V-79A			5-31	
IC						IC-V-79B			5-31	
IC						IC-V-79C			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building, West, Battery Charger Area
Designator: CB-FA-2e
Building: Control Building

Sheet 3 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
IC						IC-V-79D			5-31	
AH							AH-P-8A		5-31	To be protected.
AH							AH-P-8B		5-31	To be protected.
AH								AH-P-9A	5-31	
AH								AH-P-9B	5-31	
AH						AH-D-2B			5-31	
AH					AH-D-30A				5-31	
AH					AH-D-31A				5-31	
AH							AH-D-30C, AH-D-31C		5-31	
AH						AH-D-38			5-31	
AH						AH-D-39			5-31	
AH						AH-D-43B			5-31	
AH						AH-D-44B			5-31	
EP					X			EH-DP-1B	5-31	
EP						X	EH-DP-1M		5-31	
EP							EH-DPES-1F		5-31	
EP					EH-DPESDG-1Q				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building, West, Battery Charger Area
Designator: CB-FA-2e
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls (two) and Metal Panels (two)	Fire Hazards Report	
	Inverters (1B and 1D)					
	DC Distribution Panel			Class A Doors		
	Diesel Generator Distribution Panel			HVAC duct Smoke Detectors		
	DC Main Distribution Panel			Portable Dry Chemical Extinguisher		
	Battery Chargers (1C, 1D, and 1F)					
	(AC distribution panels VBB & VBD7)					
Missile	Transient Sources					

SCENARIO TABLE

Location Name: Control Building, West, Battery Chorger Area
Designator: CB-FA-2e
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, electrical cabinets, or transient fuel.	1 Confined.	Proximity	Adjacent Equipment		Yes.	2×10^{-5} (3 x 10^{-3}/year fire) x (0.5 geometry) x (0.5 failure to suppress) x (0.1 severity) x (0.3 spurious signal in NR-V-5)	(comparison)	Smoke could get throughout control building and fuel handling building via ventilation. (See impact table.)
		2. Engulfing.	Closed Doors	Incapable of Propagation		Yes.	10^{-4}	No action (subset of scenario 1).	
		3. Engulfing.	Open East Door	CB-FA-2d		Yes.	5×10^{-5} (smoke)	No action (subset of CB-FA-2d, scenario 3).	Total loss of instrumentation.
		4. Engulfing.	Open North Door	CB-FA-2c		Yes.	$< 10^{-5}$ (small smoke effect)	No action (subset of CB-FA-2c, scenario 6).	More instrumentation is lost.
		5. Engulfing.	Open South Door	CB-FA-2g		Yes.	$< 10^{-5}$ (small smoke effect)	No action (additional failures not important).	
Missiles									Not considered as likely.

IMPACT TABLE

Location Name: West Battery Charger Area
Designator: CB-FA-2e
Building: Control Building

Scenario Summary: Fire; Scenario 1; Fail Cabinets and Cables

System Cost	Components Affected by the Hazard
DC Train B	Inverters and chargers related to DC train B in the area, along with associated power and control circuits.
	AH-D-30E - no significant impact on CB-HVAC.
MU/A and MU/B	MU-P-3A, MU-P-3B, MU-V-6A and MU-V-6B control cables; also several other MUPS valves.
	Emergency feedwater valves EF-V-30B and EF-V-30D fail closed; minor impact on EF system availability (1/2 of the injection valves lost).
NR/All	Spurious signal in NR-V-5 control cable (conditional frequency point estimate 0.3); other nuclear river valves in the area.
EP/B	Control cables to train B of electric buses.
ESV/C	Control cable to ESV-480V-CC-1C.
	Main steam valves effect not important.
IC/B	Control cable to IC-P-1B; the control cables to other valves may spuriously close the valve, but are very unlikely to disable IC.
	Several HVAC components are affected, but HVAC is failed because DC bus B is failed.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building, East, Battery Area
Designator: CB-FA-2F
Building: Control Building

Sheet 1 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
DC						DC-P-1B			1	Will be protected.
IC						IC-P-1B				Will be protected.
EP				Battery Rack 1A	X				1, 1-FHA-035	
EP				Battery Rack 1C	X				1, 1-FHA-035	
EP						125/250V DC ES Distribution Panel 1B			Table 3.11-16	
AH						X X		AH-D-30B AH-D-31A AH-D-31B		Fail closed on loss of air, which is significant.
AH						AH-E-18B			Table 3.11-22	Will be protected.
AH							Event Monitoring Trains A and B		Cable Tray Drawings	
MU						MU-P-2C MU-P-[illegible]			Table 3.11-19 Table 3.11-19	
MU						MU-P-3C			5-31	Will be protected.
MU						MU-V-12			5-31	
MU						MU-V-14B			5-31	Will be protected.
MU						MU-V-16C			5-31	Will be protected.
MU						MU-V-16D			5-31	
MU						MU-V-18			5-31	
MU						MU-V-217			5-31	
MU						MU-V-20			5-31	
MU						MU-V-37			5-31	Will be protected.
MU						MU-V-1B			5-31	
MU						MU-V-8			5-31	
MU						MU-V-6A			5-31	
MU						MU-V-6B			5-31	
MU						MU-V-11A			5-31	
MU						MU-V-11B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building, East, Battery Area
Designator: CB-FA-2f
Building: Control Building

Sheet 2 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF						EF-V-2B			5-31	ally open.
EF							EF-V-30A		5-31	Crosstie valve.
EF							EF-V-30C		5-31	
EF						EF-V-52			5-31	Normally open.
EF						EF-V-53			5-31	
EF						EF-V-54			5-31	
EF						EF-V-55			5-31	
FW						FW-V-5A			5-31	
FW						FW-V-92A			5-31	
MS						MS-V-8A			5-31	
MS						MS-V-8B			5-31	
[illegible]						MS-V-10B			5-31	
						X	MS-V-13B		5-31	
DH						DH-V-4B			5-31	
DH						DH-V-5B			5-31	
DH						DH-V-6B			5-31	
BS						BS-V-3B			5-31	Normally closed MOV.
BS						BS-V-2B			5-31	
IC						IC-P-1A			5-31	
IC						IC-V-1A			5-31	
IC						IC-V-1B			5-31	
IC						IC-V-4			5-31	To be rerouted.
AH						AH-E-1B			5-31	
AH						AH-P-9A AH-P-9B			5-31	Will be protected.
AH						AH-P-3B			5-31	Will be protected.
AH						X		AH-D-30A AH-D-31A	5-31	
AH						X		AH-D-30B AH-D-31B	5-31	
AH						AH-D-30C AH-D-31C			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building, East, Battery Area
Designator: CB-FA-2F
Building: Control Building

Sheet 3 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH						AH-D-41B			5-31	
AH						AH-D-102			5-31	
NS						NS-P-1C			5-31	Will be protected.
NR						NR-P-1C			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-4B			5-31	
NR						NR-V-1B			5-31	Will be protected.
NR						NR-V-15B			5-31	Will be protected.
DR						DR-P-1B			5-31	Will be protected.
DR						DR-V-1B			5-31	Will be protected.
RR						RR-V-1B			5-31	
RR						RR-V-3B			5-31	
RR						RR-V-4B			5-31	
RR						RR-V-4D			5-31	
EP						EE-SGES-1S			5-31	
EP						EE-SGESH-1T			5-31	
EP						EG-CCESSH-1B			5-31	
EP					EH-DP-1A				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building East Battery Area
Designator: CB-FA-2f
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls (two) and Metal Panels (two)	Fire Hazards Report	
	Battery Racks (1A and 1C)			Class A Doors Hydrogen Monitors in Exhaust Ventilation System HVAC Duct Ionization Dectection		
Missiles	Transient Sources					

SOURCE AND MITIGATION TABLE

Location Name: Control Building East Battery Area
Designator: CB-FA-2f
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, Battery, or Transient Fuel	1. Confined.	Proximity	Adjacent Battery Rack		Yes.	6×10^{-6} (10^{-3} per year fire) x (0.2 failure to suppress) x (0.03 severity)	(comparison)	Smoke could get throughout control building and fuel handling building via ventilation. Very severe fire must take place to fail protected cables. (See impact table.)
		2. Engulfing.	Closed Doors	Incapable of Propagation		Yes.	10^{-4}	No action (subset of scenario 1).	
		3. Engulfing.	Open North Door	CB-FA-2d		Yes.	5×10^{-5}	No action (subset of CB-FA-2d, scenario 5).	Smoke.
		4. Engulfing.	Open West Door	CB-FA-2g		Yes.	$< 10^{-5}$	No action (no important additional failures).	Direct flame or hot gas effect is very unlikely.
Missiles									Not considered as likely.

IMPACT TABLE

Location Name: Battery Area, East
Designator: CB-FA-2f
Building: Control Building

Scenario Summary: Fire; Scenario 1

Sheet 1 of 2

System Cost	Components Affected by the Hazard
EP/B	Train B of electric power, spurious signal in the control cables.
DC/B	Control cable for DC-P-1B (to be protected).
IC/All	Control cable for IC-P-1B (to be protected) and IC-P-1A.
DC Power1A, DC Power 1C	Battery racks and power cables 1A and 1C.
CB-HVAC, Partial Loss	Control cables for AH-E-18B (to be protected), AH-D-30B, AH-D-31A, AH-E-31B, AH-D-30C, AH-D-31C, AH-D-41A, and AH-D-102.
MU/A, MU/C	Control cables for MU-P-3A, MU-P-3C, MU-V-14B, and MU-V-16C (to be protected).
DM/B	Control cables for DH-V-4B, DH-V-5B, and DH-V-6B.
BS/B	MOV-BC-V-3B fails as-is (closed).
AH/B	Control cable for AH-E-1B.
	Control cable for EF-V-52, EF-V-53, EF-V-54 and EF-V-55.
	Spurious closure of two parallel valves very unlikely.
NS/C	Control cable for NS-P-1C (to be protected).
NR/C	Control cable for NR-P-1C (some other NR valve control cables in the area).

IMPACT TABLE (continued)

Sheet 2 of 2

System Cost	Components Affected by the Hazard
DR/B	Control cable for DR-P-1B. RR valves are normally open. Unlikely for fire to close all valves.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building, West Battery Area
Designator: CB-FA-2g
Building: Control Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EP				Battery Rack 1B	X				1, 1-FHA-035	
EP				Battery Rack 1D	X				1, 1-FHA-035	
EP					125/250V DC ES Distribution Panel 1B				3.11-16	
EP					EH-DP-1B				5-31	
AH						X		AH-D-30A AH-D-31A		Fails closed on loss of air, which is significant.
AH							Event Monitoring Trains A and B		Cable Tray Drawings	
EF						X	EF-V-30B		5-31	
EF							EF-V-30D		5-31	
BS						BS-V-2A			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building, West Battery Area
Designator: CB-FA-2g
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls (two) and Metal Panels (two)	Fire Hazards Report	
	Battery Racks (1B and 1D)			Class A Doors Hydrogen Monitors in Exhaust Ventilation System HVAC Duct Ionization Detection		
Missiles	Transient Sources					

SCENARIO TABLE

Location Name: Control Building, West Battery Area
Designator: CB-FA-2g
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further [illegible]	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, Battery, or Transient Fuel	1 Confined.	Proximity	Adjacent Battery Rack		Yes.	10^{-3}	(comparison)	Smoke could get throughout control building and fuel handling building via ventilation.
		2. Engulfing.	Closed Doors	Incapable of Propagation		Yes.	10^{-4}	No action (subset of scenario 1).	
		3. Engulfing.	Open North Door	CB-FA-2e		Yes.	5×10^{-5}	No action (subset of CB-FA-2e, scenario 5).	
		4. Engulfing.	Open East Door	CB-FA-2f		Yes.	$< 10^{-5}$	No action (subset of CB-FA-2f, scenario 4).	
		5. Engulfing.	Open West Door	FH-FZ-5		Yes.	$< 10^{-5}$	No action (additional failures not important).	Small smoke effect. Large dilution, smoke rising.
Missiles									Not considered as likely.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building 4,160V Switchgear 1D Area
Designator: CB-FA-3a
Building: Control Building

Sheet 1 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-P-1A			1	
MU						MU-P-1B			1	
MU						MU-P-2A			3.11-19	
EF						EF-P-2A			1	
EF					EF-P-2A				1	
AH						X		AH-D-43A		Fail closed on loss of air, which is significant.
AH								AH-D-43B		
AH						X		AH-D-44B		
RR						RR-P-1A			1	
RR					RR-P-1A				1	
DR						DR-P-1A			1	
DH						DH-P-1A			1	
EP				4,160V SWGR-1D	X	X			1, 1-FHA-035	
EP								Bus Bar to 1E SWGR from Auxiliary Transformer	Plant Visit	
EP	A				480V ESV CC-1A				3.11-16	
EP	A				480V ES SWGR-1P				3.11-16	
EP	A				480V SH ES SWGR-1R				3.11-16	
MU					MU-V-14A				5-31	
MU					MU-V-16A				5-31	
MU					MU-V-16B				5-31	
MU					MU-V-36				5-31	Will be rerouted.
MU					MU-V-3				5-31	
MU					MU-V-4				5-31	
EF						EF-V-30A			5-31	
EF						EF-V-30C			5-31	
FW					FW-P-1A				5-31	
FW					FW-P-1B				5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 4,160V Switchgear 1D Area
Designator: CB-FA-3a
Building: Control Building

Sheet 2 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
FW					FW-P-1A				5-31	
FW						FW-V-1B			5-31	
MS							MS-V-3A, MS-V-3B, MS-V-3C, MS-V-3C, MS-V-3E, MS-V-3F		5-31	
MS							MS-V-4A		5-31	
MS						MX-V-10A			5-31	
MS						MX-V-13A			5-31	
AS						AS-V-4			5-31	
DH						DH-V-4A			5-31	
DH						DH-V-5A			5-31	
DH						DH-V-6A			5-31	
BS						BS-V-3A			5-31	
IC						IC-V-3			5-31	Will be rerouted.
AH						AH-E-1A			5-31	
AH						AH-E-18A			5-31	
AH						AH-D-27A AH-D-24A			5-31	
AH							AH-D-43A, AH-D-44A		5-31	
AH						AH-D-41A			5-31	
AH								AH-D-43A AH-D-44D	5-31	
AH						X		AH-D-43B AH-D-44B	5-31	
AH						AH-D-43C AH-D-44C			5-31	
AH						AH-D-43D AH-D-44D			5-31	
AH						AH-D-101			5-31	
NS						NS-V-52A			5-31	
NS						NS-V-53A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 4,160V Switchgear 1D Area
Designator: CB-FA-3a
Building: Control Building

Sheet 3 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NR						NR-V-1A			5-31	
NR						NR-V-3			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-10A			5-31	
NR						NR-V-10B			5-31	
DR						DR-V-1A			5-31	
RR						RR-V-1A			5-31	
EP				ED-SGES-1D	X	X			5-31	
EP					ED-SGES-1E				5-31	
EP						EG-Y-1A			5-31	
EP						EG-Y-1B			5-31	
EP						EG-SEC-1C			5-31	Will be rerouted.
EP						EG-CCESSH-1A			5-31	
EP						EH-DP-1M			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building 4160V Switchgear 1D Area
Designator: CB-FA-3a
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls (three) and Metal Panel (one)	Fire Hazards Report	
	1D 4160V Switchgear					
				Class A Doors		
				HVAC Duct Smoke Detectors		
				Location CB-FA-3d		
				Portable Dry Chemical Extinguishers; Halon Extinguisher		
				Ionization Detector		
Missiles	Transient Sources					

Location Name: Control Building 4,160V Switchgear 1D Area
Designator: CB-FA-3a
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, Cabinets, or Transient Fuel	1. Localized.				Yes.	3×10^{-3}	No action; only one train of electric power is lost, and equivalent unavailability is ve y small.	Smoke could get throughout control building and fuel handling building via ventilation.
		2. Engulfing.		Does Not Propagate		Yes.	1.5×10^{-4} (3 x [illegible]/yr [illegible] fire) x (0.1 severity factor) x (0.5 non-suppression)	Analyzed in detail.	(See impact table.) Both offsite power connection bus bars affected by direct fire impingement.
		3. Engulfing.	Open West Door	CB-FA-3b		Yes.	3×10^{-6} (3×10^{-3} per year fire) x (0.1 door to CB-FA-3b used) x (0.01 fire fighting mishap)	(comparison)	Open door to fight fire; smoke damage to 1E cabinets is very unlikely; fire fighters will take special precaution when in CB-FA-3b; fire fighting mishap, such as dropping of fire hose or accidental sprayng of 1E cabinet.
		4. Engulfing.	Open South Door	CB-FA-3d		Yes.	1.5×10^{-4}	(comparison)	Open door to fight fire; smoke damage.
Missiles									Not considered as likely.

IMPACT TABLE

Location Name: 4,160V Switchgear 1D Area
Designator: CB-FA-3a
Buidling: Control Building

Scenario Summary: Fire; Scenario 1

System Cost	Components Affected by the Hazard
Electric Power Train A	Fire damage to 4,160V switchgear 1D.
LOOP	Offsite power connection (bus bar) to 4,160V switchgears 1D and 1E smokes; direct fire damage, or must be deenergized for fire fighting.
MU/A, MU/B	Control cables to MU-P-1A and MU-P-1B. Also, valves MU-V-16A, MU-V-16B, and MU-V-14A failed.
EF/2A	Power and control cables to EF-P-2A.
FW	Feedwater valve and pumps affected (both pumps).
MS/Partial	Steam dump into the condenser partially affected.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building 4,160V Switchgear 1E Area
Designator: CB-FA-3b
Building: Control Building

Sheet 1 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-P-1A			5-31	Will be protected.
MU						MU-P-1B			5-31	
MU						MU-P-1C			5-31	
EP	B			4,160V SWGR-1E	X				1, 1-FHA-035	
EP	B				ESV CC-1B (CS-7)					
EP	B				SWGR-1S					
EP	B				SWGR-1T					
AH						X		AH-D-43C AH-D-43D AH-D-44C AH-D-44D		Fail closed on a loss of air, which is significant.
EF						EF-P-2B			1, 5-31	Changed
EF					EF-P-2B				1, 5-31	
NS						NS-P-1C			5-31	
RR						RR-P-1B			1	
RR					RR-P-1B				1	
DH						DH-P-1B			5-31	Changed
MU						MU-V-14A			5-31	
MU						MU-V-14B			5-31	
MU						MU-V-16C			5-31	
MU						MU-V-16D			5-31	
MU						MU-V-36			5-31	
MU						MU-V-37			5-31	
MU						MU-V-2A			5-31	
MU						MU-V-2B			5-31	
MU						MU-V-3			5-31	
EF							EF-V-30A		5-31	
EF							EF-V-30C		5-31	
FW						FW-P-1A			5-31	
FW						FW-P-1B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 4,160V Switchgear 1E Area
Designator: CB-FA-3b
Building: Control Building

Sheet 2 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
FW						FW-V-1A			5-31	
FW						FW-V-1B			5-31	
MS								MS-V-3A MS-V-3B MS-V-3C MS-V-3D MS-V-3E MS-V-3F	5-31	
MS							MS-V-4A		5-31	
MS							MS-V-4B		5-31	
AS						AS-V-4			5-31	
DH						DH-V-4B			5-31	
DH						DH-V-5B			5-31	
BS						BS-V-3B			5-31	
BS						BS-V-2A			5-31	
BS						BS-V-2B			5-31	
DH						DH-V-75B			5-31	
DH						DH-V-76B			5-31	
IC						IC-V-2			5-31	
IC						IC-V-3			5-31	
IC						IC-V-4			5-31	
AH						AH-E-1B			5-31	
AH						AH-P-8A AH-P-8B			5-31	
AH						AH-D-39			5-31	
AH						AH-D-43B AH-D-44B			5-31	
AH						X	AH-D-43C AH-D-44C		5-31	
AH						X	AH-D-43D AH-D-44D		5-31	
NS						NS-V-52B			5-31	
NS						NS-V-53B			5-31	
NR						NR-V-4B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building 4,160V Switchgear 1E Area
Designator: CB-FA-3b
Building: Control Building

Sheet 3 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NR						NR-V-108			5-31	
DC						DC-P-1B			5-31	
DR						DR-P-1B			5-31	
RR						RR-V-4B			5-31	
RR						RR-V-4D			5-31	
EP					ED-SGES-1D				5-31	
EP				ED-SGES-1E	X	X			5-31	
EP						EG-Y-1A			5-31	
EP						EG-Y-1B			5-31	
EP						EE-SGES-1P			5-31	
EP					EE-SGES-1S	X			5-31	
EP						X	EE-SGESSH-1R		5-31	
EP					EE-SGESSH-1T	X			5-31	
EP						EG-SEC-1C			5-31	
EP						EH-DP-1M			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building 4,160V Switchgear 1E Area
Designator: CB-FA-3b
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling, 4,160V SWGR-1E		1-FHA-035	Reinforced Concrete Walls (two) and Metal Panels (two) Class A Doors HVAC Duct Smoke Detectors Ionization Detector	Fire Hazards Report	The southwest door is permanently locked.

SCENARIO TABLE

Location Name: Control Building 4,160V Switchgear 1E Area
Designator: CB-FA-3b
Building: Control Building

Source Type	Synopsis of the Source	Scenario: Source Portion	Scenario: Paths of Propagation: Type	Scenario: Paths of Propagation: To	Scenario: Mitigation Portion	Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
Fire and Smoke	Cabling, Cabinets or Transient Fuel	1. Localized.				Yes.	1×10^{-5} (3×10^{-3} per year fire) x (1.0 geometric factor) x (0.2 failure to suppress) x (0.05 severity) x (0.3 hot shorts)	(comparison)	Smoke is judged to affect electrical equipment and relay cabinets. Smoke could get throughout control building and fuel handling building via ventilation. It is assumed that conditional frequency of a hot short in one of the three IC valves is almost unity. (See impact table.)
		2. Engulfing.	Closed Doors	Incapable of Propagation		Yes.		No action (subset of scenario 1).	
		3. Engulfing.	Open East Door	CB-FA-3a		Yes.	5×10^{-7} (3×10^{-3} per year fire) x (0.05 door to CB-FA-3a open and left open) (0.01 fire fighting mishop)	(comparison)	It is unlikely for the door from CB-FA-3a to be opened because it is not the primary access path. Smoke can damage switchgear only under rare conditions. Fire fighters will take special precautions when only water hoses are in fire switchgear area.
		4. Engulfing.	Open West Door	CB-FA-3c		Yes.	2.7×10^{-4}	No action (no additional important failures).	
Missiles									Not considered as likely.

IMPACT TABLE

Location Name: 4,160V Switchgear 1E Area
Designator: CB-FA-3b
Building: Control Building

Scenario Summary: Fire; Scenario 1; Localized Fire Affecting Cables and Cabinet within This Zone

System Cost	Components Affected by the Hazard
Electric Power Train B	Switchgear 1E and cables to switchgears 1S and 1T.
MU/All	Control cables to all three MU pumps in the area; cable for MU-P-1A to be protected. Also, several MU valve cables.
EF/B	EF-P-2B power and control cable.
NS/C	NS-P-1C control cable.
RR/B	RR-P-1B power and control cable.
DH/B	DH-P-1B control cable.
IC/All	Hot shorts in at least one of control cables for IC-V-2. Valves IC-V-3 or IC-V-4 would not be affected because their associated breaker is opened.
DC/B	Control cable for DC-P-1B.
DR/B	Control cable for DR-P-1B.
CV	Damper failure because of loss of train B of electric power.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building ESAS Area
Designator: CB-FA-3c
Building: Control Building

Sheet 1 of 4

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU	A					MU-P-1A			1	
MU						MU-P-1B			5-31-8	
MU						MU-P-1C			5-31	
MU						MU-V-14A			5-31	
MU						MU-V-14B			5-31	
MU						MU-V-16A			5-31	
MU						MU-V-16B			5-31	
MU						MU-V-16C			5-31	
MU						MU-V-16D			5-31	
Actuation	A			Actuation "A" Cabs	X		X		1-FHA-035	
	B			Actuation "B" Cabs	X		X		1-FHA-035	
				Engineered Safeguard Relay Cabs	X		X		1-FHA-035	
AH	A					X		AH-D-44D		Fails closed on a loss of air, which is significant.
AH						AH-E-1A			5-31	
AH						AH-E-1B			5-31	
AH						AH-E-1C			5-31	
AH						AH-E-18A			5-31	
AH						AH-P-8A			5-31	
AH						AH-D-39			5-31	
AH						AH-D-41A			5-31	
AH						AH-D-43B AH-D-44B			5-31	
AH						AH-D-43C AH-D-44C				
AH						AH-D-43A, AH-D-44A			5-31	
AH						AH-D-43D AH-D-44D			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building ESAS Area
Designator: CB-FA-[illegible]
Building: Control Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
Event Monitoring	A and B				Events Monitoring Trains A and B				Cable Tray Drawings	
DH	A					DH-P-1A			5-31	
DH	B					DH-P-1B			5-31	
DH						DH-V-1			5-31	
DH						DH-V-2			5-31	
DH						DH-V-4A			5-31	Discharge isolation valves. Normally closed MOV.
DH						DH-V-4B			5-31	Discharge isolation valves. Normally closed MOV.
DH						DH-V-5A			5-31	Discharge isolation valves. Normally closed MOV.
DH						DH-V-5B			5-31	Discharge isolation valves. Normally closed MOV.
DH						DH-V-75A			5-31	Discharge isolation valves. Normally closed MOV.
DH						DH-V-75B			5-31	Discharge isolation valves. Normally closed MOV.
DH						DH-V-76A			5-31	Discharge isolation valves. Normally closed MOV.
DH						DH-V-76B			5-31	Discharge isolation valves. Normally closed MOV.
MU						MU-V-17			5-31	
MU						MU-V-18			5-31	
MU						MU-V-36			5-31	
MU						MU-V-37			5-31	
MU						MU-V-2A			5-31	
MU						MU-V-2B			5-31	
MU						MU-V-3			5-31	
EF							EF-V-30A		5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building ESAS Area
Designator: CB-FA-3c
Building: Control Building

Sheet 3 of 4

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF							EF-V-30C		5-31	
MS							MS-V-3A, MS-V-3B, MS-V-3C, MS-V-3D, MS-V-3E, MS-V-3F		5-31	
MS							MS-V-4A		5-31	
MS							MS-V-4B		5-31	
BS						BS-V-3A			5-31	
BS						BS-V-3B			5-31	
BS						BS-V-2A			5-31	
BS						BS-V-2B			5-31	
IC						IC-V-2			5-31	
IC						IC-V-3			5-31	
IC						IC-V-4			5-31	
NS						NS-P-1A			5-31	
NS						NS-P-1C			5-31	
NR						NR-P-1A			5-31	
NR						NR-P-1C			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-4B			5-31	
NR						NR-V-10B			5-31	
DC						DC-P-1A			5-31	
DC						DC-P-1B			5-[illegible]	
DR						DR-P-1A				
DR						DR-P-1B				
RR						RR-P-1A			5-31	
RR						RR-P-1B			5-31	
RR						RR-V-4A			5-31	
RR						RR-V-4B			5-31	
RR						RR-V-4C			5-31	
RR						RR-V-4D			5-31	
EP						EG-Y-1A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building ESA Area
Designator: CB-FA-3c
Building: Control Building

Sheet 4 of 4

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EP						EG-Y-1B			5-31	
EP						EG-SEC-1C			5-31	
EP						EE-SGES-1P			5-31	
EP						EE-SGES-1S			5-31	
EP						EE-SGESSH-1R			5-31	
EP						EE-SGESSH-1T			5-31	
EH						EH-DP-1M			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building ESAS Area
Designator: CB-FA-3c
Building: Control Building

Sheet 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling ESAS Actuation Cabinets (A, B) ESAS Relay Cabinets Transient Sources		1-FHA-035	Reinforced Concrete Walls (two) and Metal Panel (one) Class A Rated Doors (three) and Class B Door (one)* HVAC Duct Smoke Detectors Ionization Detection Manually Actuated Normally Dry Sprinkler System Equipped with Feasible Head Nozzles Location Stairwell Portable Dry Chemical Extinguisher, CO_2 Extinguisher CB-FA-3d Portable Dry Chemical Extinguishers, Halon Extinguisher FH-FZ-5 Portable CO_2 Extinguishers	Fire Hazards Report	

*The southeast door is permanently locked.

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SOURCE AND MITIGATION TABLE (continued)

Location Name: Control Building ESAS Area
Designator: CB-FA-3c
Building: Control Building

Sheet 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
				Fire Hose Protection		
				Splash Shields on the Top of the Cabinets	Plant Visit	
Missile	Transient Sources					

SCENARIO TABLE

Location Name: Control Building ESAS Area
Designator: CB-FA-3c
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	1. Cable burning due to an electrical short or transient fuel. Confined fire.	Proximity	Adjacent Cabinet		Yes.	10^{-4} (2×10^{-3} per year fire) x (1.0 geometric factor) x (0.5 failure to suppress) x (0.1 severity)	(comparison)	Smoke could get throughout control building and fuel handling building via ventilation. (See impact table.)
		2. Engulfing.	Closed Doors	Incapable of Propagation		Yes.	10^{-4}	No action (subset of scenario 1).	
		3. Engulfing.	Open East Door	CB-FA-3b		Yes.		No Action.	Smoke affects SWGR-1E when door is opened, very unlikely for switchgear to fail. Therefore, the overall impact is the same as scenario 1.
		4. Engulfing.	Open North Door (if another door on the stairwell is open, the fire could spread to another level)	Stairwell		No.	$< 10^{-5}$ (small effect)	No action (no important additional equipment is lost).	
		5. Engulfing.	Open West Door	FH-FZ-5		No.	$< 10^{-5}$	No action (no important additional equipment is lost).	Smoke dilution, rising.
		6. Engulfing.	Open South Door	CB-FA-3d		Yes.	5×10^{-5}	No action (subset of CB-FA-3d scenarios).	

IMPACT TABLE

Location Name: ESAS Area
Designator: CB-FA-3c
Building: Control Building

Scenario Summary: Fire; Scenario 1

System Cost	Components Affected by the Hazard
MU/All	Control cables for MU-P-1A, MU-P-1B, MU-P-1C, MU-V-14A, MU-V-16A, MU-V-16B, MU-V-16C, and MU-V-16D.
ESAS	Actuation cabinets.
AH/1A, AH/1B, AH/1C	Control cables for AH-E-1A, AH-E-1B, and AH-E-1C.
Instrumentation	Instrumentation cables of train A and train B.
Both Power Trains E	Control cables for switchgears 1P, 1S, 1R, and 1T (this event is recoverable).
DH/All	Control cables for DH-P-1A, DH-P-1B, and several DH valves.
Condenser Steam Dump	Cables for MS-V-3A, MS-V-3B, MS-V-3C, MS-V-3D, MS-V-3E, and MS-V-3F.
BS/All	Control cables for BS-V-3A and BS-V-3B.
IC/All	Spurious closure of IC-V-2 protected; valves IC-V-3, and IC-V-4 would not be affected because breaker is open.
NS/A and NS/C	Control cables for NS-P-1A and NS-P-1C.
	NR failure unlikely because of spurious closure of more than one valve.
DC/All	Control cables for DC-P-1A and DC-P-1B.
RR/All	Control cables for RR-P-1A and RR-P-1B.
	Partial loss of EF injection path.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building Relay Room Area
Designator: CB-FA-3d
Building: Control Building

Sheet 1 of 7

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
Electrical				Control Rod DR Power Cabs	X				1-FHA-035	
				Control Rod DR Control Cabs		X			1-FHA-035	
				Relay Cabinets XCL, XCC, and XCR					1-FHA-035	
				Relay Cabinets XPL and XPCR					1-FHA-035	
				Power Supply Cabinet PS-1	x				1-FHA-035	
					Events Monitoring Trains A and B					
MU						MU-P-1A MU-P-1B MU-P-1C				
EF						EF-P-2A EF-P-2B				
DC						DC-P-1A DC-P-1B				
IC						IC-P-1A IC-P-1B				
DR						DR-P-1A DR-P-1B				
RR						RR-P-1A RR-P-1B				
NR						NR-P-1A NR-P-1B NR-P-1C				
DH						DH-P-1A DH-P-1B				
NS						NS-P-1A				
NS						NS-P-1B				
NS						NS-P-1C				
MU						MU-P-2A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Relay Room Area
Designator: CB-FA-3d
Building: Control Building

Sheet 2 of 7

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-P-3A			5-31	
MU						MU-P-2B			5-31	
MU						MU-P-3B			5-31	
MU						MU-P-2C			5-31	
MU						MU-P-3C			5-31	
MU						MU-V-12			5-31	
MU						MU-V-14A			5-31	
MU						MU-V-14B			5-31	
MU						MU-V-16A			5-31	
MU						MU-V-16B			5-31	
MU						MU-V-16C			5-31	
MU						MU-V-16D			5-31	
MU						MU-V-17			5-31	
MU						MU-V-18			5-31	
MU						MU-V-217			5-31	
MU						MU-V-20			5-31	
MU						MU-V-32	X		5-31	
MU						MU-V-36			5-31	
MU						MU-V-37			5-31	
MU						MU-V-1A			5-31	
MU						MU-V-1B			5-31	
MU						MU-V-2A			5-31	
MU						MU-V-2B			5-31	
MU						MU-V-3			5-31	
MU						MU-V-4			5-31	
MU						MU-V-8			5-31	
MU						MU-V-6A			5-31	
MU						MU-V-6B			5-31	
MU						MU-V-11A			5-31	
MU						MU-V-11B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Relay Room Area
Designator: CB-FA-3d
Building: Control Building

Sheet 3 of 7

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EF						EF-V-2A			5-31	
EF						EF-V-2B			5-31	
EF							EF-V-30A		5-31	
EF						EF-V-30B			5-31	
EF							EF-V-30C		5-31	
EF							EF-V-20D		5-31	
EF						EF-V-52			5-31	
EF						EF-V-53			5-31	
EF						EF-V-54			5-31	
EF						EF-V-55			5-31	
EF							EF-HSPS-A		5-31	
EF							EF-HSPS-B		5-31	
FW						FW-P-1A			5-31	
FW						FW-P-1B			5-31	
FW						FW-V-1A			5-31	
FW						FW-V-1B			5-31	
FW						FW-V-5A			5-31	
FW						FW-V-5B			5-31	
FW						FW-V-92A			5-31	
FW						FW-V-92B			5-31	
MS						X	MS-V-3A, MS-V-3B, MS-V-3C, MS-V-3D, MS-V-3E, MS-V-3F		5-31	
MS						MS-V-8A			5-31	
MS						MS-V-8B			5-31	
MS					X		MS-V-4A		5-31	
MS					MS-V-4B				5-31	
MS						MS-V-2A			5-31	
MS						MS-V-			5-31	
MS						MS-V-10A			5-31	
MS						MS-V-10B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Relay Room Area
Designator: CB-FA-3d
Building: Control Building

Sheet 4 of 7

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MS						MS-V-13A			5-31	
MS						MS-V-13B			5-31	
AS						AS-V-4			5-31	
DH						DH-V-1			5-31	
DH						DH-V-2			5-31	
DH						DH-V-3			5-31	
DH						DH-V-4A			5-31	
DH						DH-V-4B			5-31	
DH						DH-V-5A			5-31	
DH						DH-V-5B			5-31	
DH						DH-V-6A			5-31	
DH						DH-V-6B			5-31	
BS						BS-V-3A			5-31	
BS						BS-V-3B			5-31	
BS						BS-V-2B			5-31	
DH						DH-V-75A			5-31	
DH						DH-V-75B			5-31	
DH						DH-V-76A			5-31	
DH						DH-V-76B			5-31	
IC						IC-V-1A			5-31	
IC						IC-V-1B			5-31	
IC						IC-V-2			5-31	
IC						IC-V-3			5-31	
IC						IC-V-4			5-31	
IC						IC-V-79A			5-31	
IC						IC-V-79B			5-31	
IC						IC-V-79C			5-31	
IC						IC-V-79D			5-31	
AH						AH-E-1A			5-31	
AH						AH-E-1B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Relay Room Area
Designator: CB-FA-3d
Building: Control Building

Sheet 5 of 7

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH						AH-E-1C			5-31	
AH						AH-E-18A			5-31	
AH						AH-E-18B			5-31	
AH						AH-D-27A AH-E-24A			5-31	
AH						AH-P-8A AH-P-8B	X		5-31	
AH						AH-P-9A AH-P-9B	X		5-31	
AH						AH-D-28			5-31	
AH						AH-D-36			5-31	
AH						AH-D-38			5-31	
AH						AH-D-39			5-31	
AH						AH-D-41A			5-31	
AH						AH-D-41B			5-31	
AH						AH-D-43A AH-D-44A			5-31	
AH						AH-D-101			5-31	
AH						AH-D-102			5-31	
NS						NS-V-52A			5-31	
NS						NS-V-52B			5-31	
NS						NS-V-52C			5-31	
NS						NS-V-53A			5-31	
NS						NS-V-53B			5-31	
NS						NS-V-53C			5-31	
NR						NR-V-1A			5-31	
NR						NR-V-1B			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-3			5-31	
NR						NR-V-5			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-4B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Relay Room Area
Designator: CB-FA-3d
Building: Control Building

Sheet 6 of 7

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NR						NR-V-6			5-31	
NR						NR-V-18			5-31	
NR						NR-V-10A			5-31	
NR						NR-V-10B			5-31	
NR						NR-V-15A			5-31	
NR						NR-V-15B			5-31	
DR						DR-V-1A			5-31	
DR						DR-V-1B			5-31	
RR						RR-V-1A			5-31	
RR						RR-V-1B			5-31	
RR						RR-V-3A			5-31	
RR						RR-V-3B			5-31	
RR						RR-V-3C			5-31	
RR						RR-V-4A			5-31	
RR						RR-V-4B			5-31	
RR						RR-V-4C			5-31	
RR						RR-V-4D			5-31	
RR						RR-V-5			5-31	
EP						ED-SGES-1D			5-31	
EP						ED-SGES-1E			5-31	
EP						EG-Y-1A			5-31	
EP						EG-Y-1B			5-31	
EP						EE-SGES-1P			5-31	
EP						EE-SGES-1S			5-31	
EP						EE-SGESSH-1R			5-31	
EP						EE-SGESSH-1T			5-31	
EP						EG-SEC-1C			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Relay Room Area
Designator: CB-FA-3d
Building: Control Building

Sheet 7 of 7

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
EP						EG-CCESSH-1A			5-31	
EP						EG-CCESSH-1B			5-31	
EP						EH-DP-1M	X		5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building Relay Room Area
Designator: CB-FA-31
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls	Fire Hazards Report	
	Relay Cabinets XCL, XCC, XCR, XPL and XPCR			Class A Doors		
				Marinite Boards between Redundant Cable Trays		
	Power Supply Cabinet PS-1			Low Pressure Carbon Dioxide System Actuated by Heat Detectors		
	Control Rod or Power and Control Cabinets			HVAC Duct Smoke Detectors		
	Nonnuclear and Integrated Control System Panels			Ionization Fire Detection		
	Analog Multiplexer			Portable Dry Chemical Extinguishers		
	Annunciator Logic Cabinet			Halon Extinguisher		
				FH-FZ-5		
				Portable CO_2 Extinguishers		
				Fire Hose Protection		

SCENARIO TABLE

Location Name: Control Building Relay Room Area
Designator: CB-FA-3d
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	1. Confined. Cable burning due to an electrical short or transient fuel.	Center of the Room	Adjacent Equipment		Yes.	2×10^{-6} (7×10^{-3}/yr fire (0.05 geometric factor) x (0.1 severity factor) x (0.3 non-suppression) x (0.2 operator error)	Comparison.	Operations can use the alternate shutdown system to recover from the fire effects.
		2. Engulfing.	Closed Doors			Yes.		No Action (subset of scenario 1).	
		3. Engulfing.	Open North Door	1. CB-FA-3a 2. CB-FA-3c		No.			Impact the same as CB-FA-3a or CB-FA-3c fire.
		4. Engulfing.	Open West Door	FH-FZ-5		No.			Smoke or fire would not have adverse effects on safety cables in FH-FZ-5 because of distance.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building Instrument Shop Area
Designator: CB-FA-4a
Building: Control Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
				No components of interest in this location.						

SOURCE AND MITIGATION TABLE

Location Name: Control Building Instrument Shop Area
Designator: CB-FA-4a
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling (computer equipment) Books in Library Transients		1-FHA-036	Reinforced Concrete Walls Class A Doors High Pressure Halon 1301 Suppression System for Computer Subfloor Area, Actuated by Ionization Detectors Fire Damper Separating Computer Areas of Locations CB-FA-4a and 4b Portable Dry Chemical Extinguisher Portable CO_2 Extinguisher Fire Door (can be dropped) across Window Separating Shift Superintendent's Office and Control Room	Fire Hazards Report	
Flood	Plumbing		1-FHA-036			

SCENARIO TABLE

Location Name: Control Building Instrument Shop Area
Designator: CB-FA-4a
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling (computer equipment)	Cable burning due to an electrical short or transient fuel.							
		1. Engulfing.	Open Fire Damper or Open East Door	CB-FA-46		No.	10^{-5} (large fire, smoke move through ducts)		Smoke could travel throughout control building and fuel handling building via ventilation.
Flood	Plumbing	2. A pipe break occurs.				No.	$< 10^{-5}$		First, it would have to get past two doors. Second, the equipment is off the ground. Third, it is such a well-traveled area, and the source not huge, so it would be spotted very soon.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building Control Room Area
Designator: CB-FA-4b
Building: Control Building

Sheet 1 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
				Nuclear Instrumentation and Reactor Protection Panels A, B, C, and D					1-FHA-035	
				Safety-Related Control Consoles and Panels					1-FHA-035	
					Event Monitoring Trains A and B				Cable Tray Drawings	
MU						MU-P-1A			5-31	
MU						MU-P-1B			5-31	
MU						Mu-P-1C			5-31	
MU						MU-P-2A			5-31	
MU						MU-P-3A			5-31	
MU						MU-P-2B			5-31	
MU						MU-P-3B			5-31	
MU						MU-P-2C			5-31	
MU						MU-P-3C			5-31	
MU						MU-V-12			5-31	
MU						MU-V-14A			5-31	
MU						MU-V-14B			5-31	
MU						MU-V-16A			5-31	
MU						MU-V-16B			5-31	
MU						MU-V-16C			5-31	
MU						MU-V-16D			5-31	
MU							MU-V-17		5-31	
MU						MU-V-18			5-31	
MU						MU-V-217			5-31	
MU						MU-V-20			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Control Room Area
Designator: CB-FA-4b
Building: Control Building

Sheet 2 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
MU						MU-V-32			5-31	
MU						MU-V-36			5-31	
MU						MU-V-37			5-31	
MU						MU-V-1A			5-31	
MU						MU-V-1B			5-31	
MU						MU-V-2A			5-31	
MU						MU-V-2B			5-31	
MU						MU-V-3			5-31	
MU						MU-V-4			5-31	
MU						MU-V-8			5-31	
MU						MU-V-6A			5-31	
MU						MU-V-6B			5-31	
MU						MU-V-11A			5-31	
MU						MU-V-11B			5-31	
EF						EF-P-2A			5-31	
EF						EF-P-2B			5-31	
EF						EF-V-2A			5-31	
EF						EF-V-2B			5-31	
EF							EF-V-30A		5-31	
EF						EF-V-30B			5-31	
EF							EF-V-30C		5-31	
EF							EF-V-30D		5-31	
EF						EF-V-52			5-31	
EF						EF-V-53			5-31	
EF						EF-V-54			5-31	
EF						EF-V-55			5-31	
EF							EF-HSPS-A		5-31	
EF							EF-HSPS-B		5-31	
FW						FW-P-1A			5-31	
FW						FW-P-1B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Control Room Area
Designator: CB-FA-4b
Building: Control Building

Sheet 3 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
FW						FW-V-1A			5-31	
FW						FW-V-1B			5-31	
FW						FW-V-5A			5-31	
FW						FW-V-5B			5-31	
FW						FW-V-92A			5-31	
FW						FW-V-92B			5-31	
MS						MS-V-8A			5-31	
MS						MS-V-8B			5-31	
MS					MS-V-4A		X		5-31	
MS					MS-V-4B		X		5-31	
MS						MS-V-2A			5-31	
MS						MS-V-2B			5-31	
MS						MS-V-10A			5-31	
MS						MS-V-10B			5-31	
MS						MS-V-13A			5-31	
MS						MS-V-13B	X		5-31	
AS						AS-V-4			5-31	
DH						DH-P-1A			5-31	
DH						DH-P-1B			5-31	
DH						DH-V-1			5-31	
DH						DH-V-3			5-31	
DH						DH-V-4A			5-31	
DH						DH-V-4B			5-31	
DH						DH-V-5A			5-31	
DH						DH-V-5B			5-31	
DH						DH-V-6A			5-31	
DH						DH-V-6B			5-31	
BS						BS-V-3A			5-31	
BS						BS-V-3B			5-31	
BS						BS-V-2B			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Control Room Area
Designator: CB-FA-4b
Building: Control Building

Sheet 4 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
IC						IC-P-1A			5-31	
IC						IC-P-1B			5-31	
IC						IC-V-1A			5-31	
IC						IC-V-1B			5-31	
IC						IC-V-2			5-31	
IC						IC-V-3			5-31	
IC						IC-V-4			5-31	
IC						IC-V-79A			5-31	
IC						IC-V-79B			5-31	
IC						IC-V-79C			5-31	
IC						IC-V-79D			5-31	
AH						AH-E-1A			5-31	
AH						AH-E-1B			5-31	
AH						AH-E-1C			5-31	
AH						AH-E-18A			5-31	
AH						AH-E-18B			5-31	
AH						AH-D-27A AH-E-24A			5-31	
AH						AH-P-8A AH-P-8B	X		5-31	
AH						AH-P-9A AH-P-9B	X		5-31	
AH						AH-D-28			5-31	
AH						AH-D-36			5-31	
AH						AH-D-38			5-31	
AH						AH-D-39			5-31	
AH						AH-D-41A			5-31	
AH						AH-D-41B			5-31	
AH						AH-D-101			5-31	
AH						AH-D-102			5-31	
NS						NS-P-1A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Control Room Area
Designator: CB-FA-4b
Building: Control Building

Sheet 5 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
NS						NS-P-1B			5-31	
NS						NS-P-1C			5-31	
NS						NS-V-52A			5-31	
NS						NS-V-52B			5-31	
NS						NS-V-52C			5-31	
NS						NS-V-53A			5-31	
NS						NS-V-53B			5-31	
NS						NS-V-53C			5-31	
NR						NR-P-1A			5-31	
NR						NR-P-1B			5-31	
NR						NR-V-1A			5-31	
NR						NR-V-1B			5-31	
NR						NR-V-1C			5-31	
NR						NR-V-3			5-31	
NR						NR-V-5			5-31	
NR						NR-V-4A			5-31	
NR						NR-V-4B			5-31	
NR						NR-V-6			5-31	
NR						NR-V-18			5-31	
NR						NR-V-10A			5-31	
NR						NR-V-10B			5-31	
NR						NR-V-15A			5-31	
NR						NR-V-15B			5-31	
DC						DC-P-1A			5-31	
DC						DC-P-1B			5-31	
DR						DR-P-1A			5-31	
DR						DR-P-1B			5-31	
DR						DR-V-1A			5-31	
DR						DR-V-1B			5-31	
RR						RR-P-1A			5-31	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Control Building Control Room Area
Designator: CB-FA-4b
Building: Control Building

Sheet 6 of 6

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RR						RR-P-1B			5-31	
RR						RR-V-1A			5-31	
RR						RR-V-1B			5-31	
RR						RR-V-3A			5-31	
RR						RR-V-3B			5-31	
RR						RR-V-3C			5-31	
RR						RR-V-4A			5-31	
RR						RR-V-4b			5-31	
RR						RR-V-4C			5-31	
RR						RR-V-4D			5-31	
RR						RR-V-5			5-31	
EP					X	ED-SGES-1D			5-31	
EP						ED-SGES-1E			5-31	
EP						EG-Y-1A			5-31	
EP						EG-Y-1B			5-31	
EP						EE-SGES-1P			5-31	
EP						EE-SGES-1S			5-31	
EP						EE-SGESSH-1R			5-31	
EP						EE-SGESSH-1T			5-31	
EP						EG-SEC-1C			5-31	
EP						EG-CCESSH-1A			5-31	
EP						EG-CCESSH-1B			5-31	
EP						EN-DP-1M			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building Control Room Area
Designator: CB-FA-4b
Building: Control Building

Sheet 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls	Fire Hazards Report	
	NVC Instructions and Reaction Protection Panels A, B, C, and D			Class A Doors		
	Computer Input and Output and Peripheral Cabinets Analog Local Input Logic Input and Output; Peripheral Input			High Pressure Halon 1301 Suppression for Computer Subfloor Area and Cable Trench Actuated by Ionization Detectors		
	Control Consoles and Panels			Ionization Fire Detection inside Safety-Related Control Consoles and Panels		
	RBB and RBA Transformers 1A and 1B					
	Computer Console Desk			Portable CO_2 Extinguishers		
				Portable Halon Extinguishers		
				Portable Water Extinguishers		
				Location		
				FH-FZ-5 Fire Hose Protection		

SOURCE AND MITIGATION TABLE (continued)

Location Name: Control Building Control Room Area
Designator: CB-FA-4b
Building: Control Building

Sheet 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
				Portable CO_2 Extinguishers Portable Dry Chemical Extinguishers		

SCENARIO TABLE

Location Name: Control Building Control Room Area
Designator: CB-FA-4b
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, electrical and electronic components, and transient fuel.	1. Control panel fire in panels CC and CR.	Confined to two panels			Yes.	3.0 x 10^{-6} (4.9^{-3}/yr fire) x (0.01 geometric factor) x (0.05 human error)	Comparison	Fire occurs in panel CC and CR and fails a large set of vital control circuits. Operators without alternate shutdown system to mitigate the fire. Human error rate is established for judgement.
		2. Fire confined to panels other than CC and CR.				No.			Impact limited to more vital systems or systems whose failure does not directly lead to core damage.
		3. Engulfing.	Open North Door Open West Door	CB-FA-4a FH-FZ-5		No.			Very unlikely and plant impact is not worse than scenario 1.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Stairwell, North of Control Tower
Designator: ________
Building: ______________

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
CW								Two Chilled Water Pipes Chillers in Basement to Fans in CA-FZ-5a CA-FZ-5b		

SOURCE AND MITIGATION TABLE

Location Name: Stairwell North of Control Tower
Designator: ________
Building: ________

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Flood	Pipe Break		Plant Visit	Stairs and Doors		

SCENARIO TABLE

Location Name: Stairwell North of Control Tower
Designator:
Building:

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Flood	Control Water Pipe	1. Pipe breaks.	Stairs and Open Door	FH-FZ-6		Yes.	10^{-4}	(CB-HVAC)	Dominated by flood sources in FH-FZ-6.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building, North, Heating and Ventilation Equipment Area
Designator: CB-FZ-5a
Building: Control Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH					X	X		AH-E-19A	1-FHA-036	Vent exhaust fans.
					X	X		AH-E-19B		
AH					X	X		AH-E-18A	1-FHA-036	Emergency vent supply fan A.
AH					X	X		AH-E-17A	1-FHA-036	Normal duty supply fan A.
AH						X		AH-D-87A		Fails closed on loss of air, which is significant.
AH								AH-D-87B		
AH						X		AH-D-36		
AH						X		AH-D-38		
AH						X		AH-D-35B		
AH						X		AH-D-37A		
AH						X		AH-D-41A		
AH						X		AH-D-32A		

SOURCE AND MITIGATION TABLE

Location Name: Control Building, North, Heating and Ventilation Equipment Area
Designator: CB-FZ-5a
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls	Fire Hazards Report	Smoke is not assumed to impact equipment.
	Fans			HVAC Duct Smoke Detectors		
	AH-F-3A Charcoal (from filters)			Charcoal Systems Thermal Fire Detectors		These are water nozzles inside the system, drains leading to the outside, and floor drains (plant visit).
	Heaters AH-C-5A			Automatic Deluge Water Spray System		
Flood	Chilled Water System Piping		Plant Visit	All the HVAC Units are on Pedestals; 6 Inches, except for 18A, 19B that are on 18 Inches; 19A on 4 Feet	Plant Visit	
	Fire Protection System Piping		Plant Visit	Also, Floors Have Drains near Filters (AH-F-3A)		

SCENARIO TABLE

Location Name: Control Building, North, Heating and Ventilation Equipment Area
Designator: CB-FA-5a
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling, etc.	Cable burning due to an electrical short or transient fuel.							Fire must fail several cables; need relatively severe fire to damage all cables of interest of redundant trains.
		1. Confined.	Proximity	Adjacent Equipment		Yes.	3×10^{-5} (10^{-3}/yr fire) x (0.3 geometric factor) x (0.5 failure to suppress) x (0.2 severity factor)	(CB-HVAC)	
		2. Engulfing.	Open West Door	FH-FZ-5 (upper portion)		Yes.	$< 10^{-5}$	No action.	No equipment of importance in upper FH-FZ-5.
Flood	Control Water or Feedwater Pipe	3. Pipe break.	Open West Door	FH-FZ-5		No, impact limited to a few components.	10^{-4}		Required failure of floor drains.
		4. Pipe break.	Confined to Area			No, impact limited to a few components.	10^{-3}		

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Control Building South H and V Equipment Area
Designater: CB-FZ-5b
Building: Control Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH					X	X		AH-E-18B	1-FHA-036	Emergency vent supply fan B.
AH					X	X		AH-E-17B	1-FHA-036	Normal duty supply fan B.
AH						X		AH-D-41B AH-D-32B		Fails closed on a loss of air, which is significant.
AH						AH-D-28			5-31	
AH						AH-D-39			5-31	

SOURCE AND MITIGATION TABLE

Location Name: Control Building, South, Heating and Ventilation Equipment Area
Designater: CB-FZ-5b
Building: Control Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Cabling		1-FHA-035	Reinforced Concrete Walls	Fire Hazards Report	Smoke is assumed not to impact equipment.
	Fans			HVAC Du Smoke Dectectors		
	AH-F-3B Charcoal (from filters)			Charcoal Systems Thermal Fire Detectors		
	Heater AH-C-5B					
				Automatic Actuation Deluge Water Spray System		
				Location FH-FZ-5 Fire Hose Protection		
Flood	Chilled Water System Piping		Plant Visit	Floors Have Drains near Filters (AH-F-3B)	Plant Visit	
	Fire Protection System Piping		Plant Visit	All the HVAC Units Are on Pedestals of at Least 6 Inches		

SCENARIO TABLE

Location Name: Control Building, South, Heating and Ventilation Equipment Area
Designator: CB-FA-5b
Building: Control Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire and Smoke	Cabling	Cable burning due to an electrical short or transient fuel.							
		1. Confined.	Proximity	Adjacent Equip-ment		Yes.	10^{-3}	(CB-HVAC)	
		2. Engulfing.	Open West Door	FH-FZ-5 (upper)		Yes.	10^{-4}	No action.	No important equipment in upper FH-FZ-5.
Flood	Control Water or Feedwater Pipe	3. Pipe break.	Open West Door	FH-FZ-5		No, impact limited.	10^{-4}		Requires floor drain failure.
		4. Pipe break.	Confined to Area			No, impact limited.	10^{-3}		

REACTOR BUILDING

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Reactor Building Outside Secondary Shield, North
Designator: RB-FZ-1a
Building: Reactor Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH					X			AH-E-A1	1	AH-E-1A (reactor building ventilation unit and cabling CG-21, 22).
AH					X			AH-E-1B	1	AH-E-1B (reactor building ventilation unit and cabling CH-14, 15).
AH					X			AH-E-1C	1	AH-E-1C (reactor building ventilation unit and cabling CS-99, 100).
RC							X		1	RC-3A-PT3 cables.
RC							X		1	RC-3A-PT4 cables.
RC							X		1	RC-4A-TE3 cables.
AH								AH-T-1	1-FHA-017	Water storage tank (not considered in fan cooler analysis).
AH								AH-E-4A	1-FHA-017	Fan.
AH								AH-E-3A	1-FHA-017	Fan.
					X				1-FHA-017	RG16A cable.
WDL		WDL-P-23							1-FHA-017	Steam generator drain pump.
WPL		WPL-P-16							1-FHA-017	Reactor drain pump.
					X				1-FHA-017	CG-23A cable.
								X	1-FHA-017	Chemical feed tank.
WDL								WDL-T3	1-FHA-017	Reactor coolant drain tank.
FW								FW-C-1A	1-FHA-017	FW-C-1A steam generator hot drain cooler.
FW								FW-C-1B	1-FHA-017	FW-C-1B steam generator hot drain cooler.
FW								Piping	E-304-081	
DH								Piping		
DH				DH-V-1	X				5-31	
DH				DH-V-2	X				5-31	
IC					IC-V [illegible]				5-31	
IC					IC-V-79C				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Reactor Building Outside Secondary Shield, North
Designator: RB-FZ-1a
Building: Reactor Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Electric Cables		1, 1-FHA-017 1-FHA-022	Halon Fire Stations Fire Hose Station	1 and 1-FHA-017	11 stations in RB-FZ-1a. 1 in RB-F2-1a
	Lube Oil Systems		1, 1-FHA-017 1-FHA-022	Portable Water Extinguisher	1 and 1-FHA-017	1 in RB-F2-1c
	Motors		1, 1-FHA-017 1-FHA-022	Ionization Fire Detection	1 and 1-FHA-017	
			1, 1-FHA-017 1-FHA-022	Ventilation	1 and 1-FHA-017	
			1, 1-FHA-017 1-FHA-022	Doors	1 and 1-FHA-017	
			1, 1-FHA-017 1-FHA-022	Walls	1 and 1-FHA-017	
			1, 1-FHA-017 1-FHA-022	Radiant Energy Heat Shields for Cables	1 and 1-FHA-017	
Flooding	Main Feedwater Pipe Break		1-FHA-017 through 1-FHA-022 E-304-081	Drain Pump in RB-F2-1c	Plant Visit	
	Decay Heat Pipe Break		Plant Visit	Drain Pump in RB-F2-1c		This pipe is normally isolated at two ends and contains small volume of water.
	Fire Hose System, Pipe Break or Initiation			Fire System Pump Under Normal Conditions		
Steam	Main Feedwater Pipe Break		E-304-081	RBS Reactor Building Emergency Cooling	E-304-713 1-FHA-017	
	Decay Heat Pipe Break		E-304-641	RBS Reactor Building Emergency Cooling	E-304-713 1-FHA-017	Decay heat piping is not pressurized because of isolation valves.
Pipe Whip	Main Feedwater		Plant Visit	Pipe Supports Walls	Plant Visit	

SCENARIO TABLE

Location Name: Reactor Building Outside Secondary Shield, North
Designator: RB-FZ-1a
Building: Reactor Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire	Motor Lube Oil Cabling	1. Localized.		AH-E-1B AH-E-1C AH-E-1A AE-1B AE-1C		Yes.	10^{-3}	(system)	
		Confined.			Wipes out room only.				
		2. Spreading via general openings.		RB-FZ-1b		Yes.	10^{-4} (10^{-3} fire)	(no action) x (10^{-1} severity factor)	In addition to the fan coolers, some instrumentation cables may be damaged.
		3. Spreading via general openings.		RB-FZ-1c		Yes.	10^{-4}	(no action)	In addition to the fan coolers, some instrumentation cables may be damaged.
		4. Spreading via general openings.		RB-FZ-1d		Yes.	10^{-4}	(no action)	In addition to the fan coolers, some instrumentation cables may be damaged.
		5. Spreading via general openings.		RB-FZ-1e		Yes.	10^{-4}	(no action)	In addition to the fan coolers, some instrumentation cables may be damaged.
		6. Spreading via general openings.		RB-FZ-2		No, very unlikely to propagate.			
Flood, Steam and Pipe Whip	Main Feedwater Pipe Break	7. Open (water).		RB-FZ-1b RB-FZ-1c RB-FZ-1d RB-FZ-1e RB-FZ-3 RB-FZ-2	The feedwater pumps will trip and will not empty hot well into containment.	Yes.	8×10^{-6} (pipe break frequency)	(comparison)	About 9 feet of water on the floor. Very unlikely.
		(steam)							Pipe whip may impact one feedwater unit and cables in RB-FZ-1a.
Smoke	Fire	8. Open. Localized.		RB-FZ-1b RB-FZ-1c RB-FZ-1d RB-FZ-1e RB-FZ-2 RB-FZ-3		No.	10^{-3}		Smoke does not have short-term effect on safety equipment.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Reactor Building Outside Secondary Shield, Southeast
Designator: RB-FZ-1b
Building: Reactor Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
SP							X		1	SP-1A-LT1 cables; sewage pumping (RE-7, A).
SP							X		1	SP-6A-PT1 cables; sewage pumping (RE-7, A).
SP							X		1	SP-6A-PT2 cables; sewage pumping (RE-7, A).
RC							X		1	RC-3A-PT3 cables.
RC							X		1	RC-3A-PT4 cables.
RC							X		1	RC-4A-TE2 cables.
RC							X		1	RC-4A-TE3 cables.
RC							X		1	RC-1-LT1 cables RFC-156A and RFC-71A.
RC							X		1	RC-1-LT2 cables.
RC							X		1	RC-1-LT3 cables.
RC					X				1-FHA-017	RG16A cable for RC-3A-PT3.
RC					X				1-FHA-017	RG17A cable for RC-3A-TE2.
RC					X				1-FHA-017	RE109A cable for SP-6A-PT1.
RC								TR-7	1-FHA-017	TR-7.
RC					X				1-FHA-017	RE156A for RC-1-LTI.
RC					X			J18	1-FHA-017	J18 junction box.
MV								Piping	Isometric Drawing	
DH					DH-V-1				5-31	
DH					DH-V-2				5-31	

SOURCE AND MITIGATION TABLE

Location Name: Reactor Building Outside Secondary Shield, Southeast
Designator: RB-FZ-1b
Building: Reactor Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Concrete Coating		1, 1-FHA-017 1-FHA-022	Fire Hose Station	1, 1-FHA-017 1-FHA-022	One station in RB-FZ-1b.
	Electric Cables		1, 1-FHA-017 1-FHA-022	Halon Fire Station	1, 1-FHA-017 1-FHA-022	One station.
			1, 1-FHA-017 1-FHA-022	Portable Water Extinguisher	1, 1-FHA-017 1-FHA-022	One in RB-FZ-1c.
			1, 1-FHA-017 1-FHA-022	Ventilation	1, 1-FHA-017 1-FHA-022	
			1, 1-FHA-017 1-FHA-022	Doors	1, 1-FHA-017 1-FHA-022	
			1, 1-FHA-017 1-FHA-022	Walls	1, 1-FHA-017 1-FHA-022	
			1, 1-FHA-017 1-FHA-022	Radiant Energy Heat Shields for Cables	1, 1-FHA-017 1-FHA-022	
			1, 1-FHA-017 1-FHA-022	Walls	1, 1-FHA-017 1-FHA-022	
Flooding	Makeup	Let down related piping	1, 1-FHA-017 1-FHA-022	Drain Sump in RB-FZ-1c	1, 1-FHA-017 1-FHA-022	A small piece of makeup pipe in the area. RCP seal-related pipes are dry under normal conditions.
Pipe Whip	Makeup		Plant Visit	Pipe Supports, Walls	Plant Visit	
Steam	Makeup		Plant Visit	RBS, Reactor Building Emergency Cooling	Plant Visit	

SCENARIO TABLE

Location Name: Reactor Building Outside Secondary Shield, Southeast
Designator: RB-FZ-1b
Building: Reactor Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency ((yr^{-1}))	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire	Cabling and Concrete Coating	1. Localized. Confined.		Cable Trays Room Only		Yes.	10^{-3}	(comparison)	One train.
		2. General openings.		RB-FZ-1a		Yes.	10^{-4} (10^{-3} fire) x (10^{-1} severity)	(no action) Assuming scenario 1 fails all instructions.	Additional failures.
		3. General openings.		RB-FZ-1c		No, very unlikely.			
		4. General openings.		RB-FZ-1d		No, very unlikely.			
		5. General openings.		RB-FZ-1e		No, very unlikely.			
		6. Stairway.		RB-FZ-2		No, very unlikely.			
Flood and Steam	Makeup Piping	7. Open (flood). Localized (pipe whip). Open (steam).		RB-FZ-1a RB-FZ-1c RB-FZ-1d RB-FZ-1e RB-FZ-3 The Rest of Reactor Building		Yes.	8×10^{-6} (pipe break frequency)	(system)	Only a few feet of water on the floor. BWST not affected. RCE seal with fail if an additional failure occurs.
Smoke	Fire	8. Opening.		The Rest of Reactor Building		No.	10^{-3}		No short term impact on important components.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Reactor Building Outside Secondary Shield, Southwest
Designator: RB-FZ-1c
Building: Reactor Building

Sheet 1 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
AH					X				1	AH-E-1A reactor building ventilation unit cables (CG-21, 22).
AH					X				1	AH-E-1B reactor building ventilation unit cables (CG-14, 15).
AH					X				1	AH-E-1C reactor building ventilation unit cables (CG-99, 100).
SP							X		1	SP-1A-LT1 cables, sewage.
SP							X		1	SP-1B-LT1 cables, sewage.
SP							X		1	SP-6A-PT1 cables, sewage.
SP							X		1	SP-6A-PT2 cables, sewage.
SP							X		1	SP-6B-PT1 cables, sewage.
RC							X		1	RC-3A-PT3 cables.
RC							X		1	RC-3A-PT4 cables.
RC							X		1	RC-3B-PT3 cables.
RC							X		1	RC-4A-TE2 cables.
RC							X		1	RC-4A-TE3 cables.
RC							X		1	RC-4B-TE2 cables (RG-61A).
RC							X		1	RC-4B-TE3 cables.
RC							X		1	RC-5A-TE2 cables.
RC							X		1	RC-5A-TE4 cables.
RC							X		1	RC-5B-TE2 cables.
RC							X		1	RC-5B-TE4 cables.
RC							X		1	RC-1-LT1 cables.
RC							X		1	RC-1-LT2 cables.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Reactor Building Outside Secondary Shield, Southwest
Designator: RB-FZ-1c
Building: Reactor Building

Sheet 2 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RC							X		1	RC-1-LT3 cables.
NI							X		1	NI-1 cables (RG-1A, RG-2a, and RG-4).
NI							X		1	NI-2 cables.
NI					X				1-FHA-017	RG1A cable for NI-1.
RC					X				1-FHA-017	RG17A cable for RC-4A-TE2.
SP					X				1-FHA-017	RE71A cable for SP-1A-LT1.
SP					X				1-FHA-017	RG16A cable for RC-3A-PT3 and RC-5B-PT8.
MU			MU-V-1A					MU-C-1A	E-304-661 1-FHA-017	Letdown cooler A.
			MU-V-1B					MU-C-1B	1-FHA-017 1-FHA-023	Letdown cooler B.
MU			MU-V-2A							
			MU-V-2B					TR-6	1-FHA-022, 1-FHA-017	TR-6.
AH								AH-E-3B	1-FHA-017	Fan.
DH			DH-V-2		X				1-FHA-017	Dropline isolation valve.
SP					X				1-FHA-017	RG 202A cable for RC-3A-PT3 and RC-BB-PT3.
SP					X				1-FHA-017	RE 177 cable for RC-5A-TE2, and others.
SP					X				1-FHA-017	RE 72A cable for SP-1B-LT1.
SP							X		1	SP-6B-PT2.
SP/RC								Tray 815	1-FHA-017	Cable tray.*
SP/RC								Tray 816	1-FHA-017	Cable tray.**
DH					DH-V-1	X			5-31	
IC			IC-V-1A			X			5-31	
IC			IC-V-1B		X	X			5-31	

* Includes cables for SP-1B-LT1, SP-6B-PT1, SP-6B-PT2, RC-5A-TE2, RC-5A-TE4, RC-5B-TE2, RC-5B-TE4, and RC-1-LT1.
**Includes cables for SP-1A-LT1, SP-6A-PT1, SP-6A-PT2, RC-1-LT2, and RC-1-LT3.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Reactor Building Outside Secondary Shield, Southwest
Designator: RB-FZ-1c
Building: Reactor Building

Sheet 3 of 3

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
IC			IC-V-2		X	X			5-31	
IC						IC-V-79A			5-31	
IC						IC-V-79B			5-31	
IC						IC-V-79C			5-31	
IC						IC-V-79D			5-31	
SP/RC								Penetration 204E	1-FHA-017	Penetration.
SP/RC								Penetration 205E	1-FHA-017	Penetration.
SP/RC								Penetration 313E	1-FHA-017	Penetration.
								Sump	1-FHA-017	Sump under letdown cooler.
MU								Piping	Isometric Drawings	
DH								Piping	Isometric Drawings	
PBS								Piping	Isometric Drawings	

SOURCE AND MITIGATION TABLE

Location Name: Reactor Building Outside Secondary Shield, Southwest
Designator: RB-72-1c
Building: Reactor Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Electric Cables		1, 1-FHA-017 1-FHA-022 1-FHA-023	Water Fire Extinguisher	1, 1-FHA-017 1-FHA-022 1-FHA-023	
	Concrete Coating		1, 1-FHA-017 1-FHA-022 1-FHA-023	Fire Hose Station	1, 1-FHA-017 1-FHA-022 1-FHA-023	
	DH-V-2 Motor		1, 1-FHA-017 1-FHA-022 1-FHA-023	Ionization Fire Detection	1, 1-FHA-017 1-FHA-022 1-FHA-023	
				Portable Halon Extinguishers	1, 1-FHA-017 1-FHA-022 1-FHA-023	Located outside personnel access hatch on Elevation 308'0" of the turbine building.
				DH-V2 Fire Protection	1, 1-FHA-017 1-FHA-022 1-FHA-023	Manually actuated dry pipe fire suppression system with a single closed head nozzle.
				Radiant Energy Heat Shields for Specified Cables	1, 1-FHA-017 1-FHA-022 1-FHA-023	See Reference 1.
				Walls	1, 1-FHA-017 1-FHA-022 1-FHA-023	
Flood	RBS Pipe Break		Plant Visit	Drain Sump	Plant Visit	
	Makeup Pipe Break		Plant Visit	Drain Sump	Plant Visit	
	Decay Heat Pipe Break		Plant Visit	Drain Sump	Plant Visit	
Steam	Makeup Pipe Break		Plant Visit	RBS Reactor Building Emergency Cooling	Plant Visit	Let down related piping.
	Decay Heat Piping		Plant Visit	RBS Reactor Building Emergency Cooling	Plant Visit	Check valves prevent the decay heat pipe in this region to be pressurized.
Pipe Whip	Makeup		Plant Visit	Pipe Supports	Plant Visit	
			Plant Visit	Walls	Plant Visit	

SCENARIO TABLE

Location Name: Reactor Building Outside Secondary Shield, Southwest
Designator: RB-FZ-1c
Building: Reactor Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire	Cables or Concrete Coating	1. Localized.			(See Source and Mitigation table.)	Yes.	10^{-3}	(comparison)	
		Confined to room only.			(See Source and Mitigation table.)				
	Decay Heat-V2 Motor								
		2. General openings.		RB-FZ-1a	(See Source and Mitigation table.)	Yes.	10^{-4}	(no action) A subset of scenario 1.	
		3. General openings.		RB-FZ-1b	(See Source and Mitigation table.)	No, very unlikely.			
		4. General openings.		RB-FZ-1d	(See Source and Mitigation table.)	No, very unlikely.			
		5. General openings.		RB-FZ-1e	(See Source and Mitigation table.)	No, very unlikely.			
		6. Stairway.		RB-FZ-2	(See Source and Mitigation table.)	No, very unlikely.			
Flood	Pipe Break in RBS	7. Open.		RB-FZ-1a RB-FZ-1b RB-FZ-1d RB-FZ-1e RB-FZ-3	(See Source and Mitigation table.)	No, impact.	2×10^{-5}		
Flood and Pipe Whip	Makeup	8. Open.		Same as Scenario 7	(See Source and Mitigation table.)	Yes.	3×10^{-6} (0.1 for pipe whip damage)	(comparison)	May cause loss of RBS from pipe whip.
Steam	Pipe Break Makeup	9. Open.		Rest of Reactor Building	(See Source and Mitigation table.)	No, very small steam impact.	8×10^{-6}		
Smoke	Fire	10. Opening.		All Reactor Building Zones	(See Source and Mitigation table.)	No.	10^{-3}		No short-term effects on the exposed components.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Reactor Building Inside Secondary Shield, East
Designator: RB-FZ-1d
Building: Reactor Building

Sheet 1 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RC					X				1	Pressurizer heater Group 8.
RC					X				1	Pressurizer heater Group 9.
RC							X		1	RC-4A-TE2 cables.
RC							X		1	RC-4A-TE3 cables.
RC							X		1	RC-5A-TE2 cables.
RC							X		1	RC-5A-TE4 cables.
RC								RC-H-1A	1-FHA-017	Steam generator A.
RC					X				1-FHA-017	RE 178A cable for RC-5A-TE2.
RC					X				1-FHA-017	RG 17A cable for RC-4A-TE2.
RC		RC-P-1A							1-FHA-017	Reactor coolant pump A.
RC		RC-P-1B							1-FHA-017	Reactor coolant pump B.
RC								RC-T2	1-FHA-017	Pressurizer.
RC			RC-RV-2			X				Pilot-operated relief valve (PORV).
RC			RC-V-2		X					PORV block valve.
RC			RC-V-1		X					Pressurizer spray line isolation.
RC			RC-V-3		X					Pressurizer spray line isolation.
RC			RC-V-4		X					Auxiliary pressurizer spray isolation from DHR.
RC			RC-RV-1A							Pressurizer safety relief valves.
RC			RC-RV-1B							Pressurizer safety relief valves.
MV								Seal Injection-Related Piping	Isometric Drawings	

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Reactor Building Inside Secondary Shield, East
Designator: RB-FZ-1d
Building: Reactor Building

Sheet 2 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
Main FW								Piping, Feedwater Injection into Once-Through-Steam-Generator	Isometric Drawings	
EF								Piping	Isometric Drawings	
MS								Piping	Isometric Drawings	
RCS								Piping	Isometric Drawings	

SOURCE AND MITIGATION TABLE

Location Name: Reactor Building Inside Secondary Shield, East
Designator: RB-FZ-1d
Building: Reactor Building

Page 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	RCP Motor and Lube Oil System		1, 1-FHA-017 through 1-FHA-022	RCP Oil Splash Guard and Reservoirs	1, 1-FHA-017 through 1-FHA-022	
				Walls		
	Electric Cables		1, 1-FHA-017 through 1-FHA-022	Fire Hose Stations	1, 1-FHA-017 through 1-FHA-022	One near the shield door on the north boundary outside RB-FZ-1d. One at the top of the shield wall.
				Ionization Fire Detection	1, 1-FHA-017 through 1-FHA-022	
				Halon Fire Extinguishers	1, 1-FHA-017 through 1-FHA-022	Located outside the personnel access hatch on Elevation 308'0" of the turbine building.
				Radiant Energy Heat Shields for Specified Cables	1, 1-FHA-017 through 1-FHA-022	See Reference 1.
Flood	Main Feedwater Pipe Break		1-FHA-017 through 1-FHA-022	Drain Sump	1-FHA-017 through 1-FHA-022	
	Emergency Feedwater Pipe Break		1-FHA-017 through 1-FHA-022	Drain Sump	1-FHA-017 through 1-FHA-022	
	Makeup Pipe Break		1-FHA-017 through 1-FHA-022	Drain Sump	1-FHA-017 through 1-FHA-022	
	RCS Pipe Break		1-FHA-017 through 1-FHA-022	Drain Sump	1-FHA-017 through 1-FHA-022	
Steam	Main Feedwater Pipe Break		1-FHA-017 through 1-FHA-022	RBS Reactor Building Emergency Cooling	1-FHA-017 through 1-FHA-022	
	Main Steam Pipe Break		1-FHA-017 through 1-FHA-022	RBS Reactor Building Emergency Cooling	1-FHA-017 through 1-FHA-022	

SOURCE AND MITIGATION TABLE

Location Name: Reactor Building Inside Secondary Shield, East
Designator: RB-FZ-1d
Building: Reactor Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
	Makeup Pipe Break		1-FHA-017 through 1-FHA-022	RBS Reactor Building Emergency Cooling	1-FHA-017 through 1-FHA-022	
	RCS Pipe Break		1-FHA-017 through 1-FHA-022	RBS Reactor Building Emergency Cooling	1-FHA-017 through 1-FHA-022	
Missile	RCP Missile Ejection		1-FHA-017 1-FHA-022	Walls	1-FHA-017 1-FHA-022	
	Pressurizer Missile		1-FHA-017 1-FHA-022	Pressurizer Missile Shield	1-FHA-017 1-FHA-022	
Pipe Whip	Main Feedwater		Plant Visit	Pipe Supports	Plant Visit	
	Emergency Feedwater		Plant Visit	Walls	Plant Visit	
	Main Steam		Plant Visit	Walls	Plant Visit	
	Makeup		Plant Visit	Walls	Plant Visit	
	RCS		Plant Visit	Walls	Plant Visit	

SCENARIO TABLE

Location Name: Reactor Building Inside Secondary Shield, East
Designator: RB-FZ-1d
Building: Reactor Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire	RCP Motor Lube Oil Cables, or Transient Fuel	1. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	Localized.		(See Source and Mitigation table.)	Yes.	10^{-2}	(Comparison)	Only cables may be affected.
		2. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-1a	(See Source and Mitigation table.)	No, very unlikely.			The fire has to be very severe and overcome long distances of low combustible loads to propagate.
		3. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-1b	(See Source and Mitigation table.)	No, very unlikely.			The fire has to be very severe and overcome long distances of low combustible loads to propagate.
		4. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-1c	(See Source and Mitigation table.)	No, very unlikely.			The fire has to be very severe and overcome long distances of low combustible loads to propagate.
		5. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-1e	(See Source and Mitigation table.)	No, very unlikely.			The fire has to be very severe and overcome long distances of low combustible loads to propagate.
		6. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-2	(See Source and Mitigation table.)	No, very unlikely.			The fire has to be very severe and overcome long distances of low combustible loads to propagate.

SCENARIO TABLE (continued)

Location Name: Reactor Building Inside Secondary Shield, East
Designator: RB-FZ-1d
Building: Reactor Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
		7. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-3	(See Source and Mitigation table.)	Yes.	10^{-3} (10^{-2} x 10^{-1})	(no action) Additional failure not important.	Smoke damage not important to safety.
Flood, Steam and Pipe Whip	Main Feedwater Piping Break	8. Main feedwater would initially flash until cooled to boiling point, then spill as water.	(steam) Openings. (flood) Openings.	Rest of Reactor Building Rest of Reactor Building	(See Source and Mitigation table.)	Yes.	8 x 10^{-6} (pipe break frequency)	(event tree)	Steam may impact the RCPS. Flood would cause about 9 feet of water on the floor; steam jet affects PORV cables; conservatively assume half of emergency feedwater and makeup are lost. Pipe whip may fail makeup or emergency feedwater piping.
Flood	Emergency Feedwater Piping	9. Pipe break may empty CST inside the containment.	Opening	Rest of Reactor Building	(See Source and Mitigation table.)	Yes.	10^{-6} (8 x 10^{-6} pipe break) x (0.1 emergency feedwater in operation	(no action) Same as emergency feedwater pipe break and no other failures.	About 9 feet of water on the floor.
Flood and Steam	Makeup	10. Pipe break.	Opening	Rest of Reactor Building	(See Source and Mitigation table.)	Yes.	2 x 10^{-5} (two makeup pipe sections)	(no action) Same as makeup pipe break and no other failure.	May degrade RCP seals. Pipe whip is judged to be of insufficient energy to cause any scenario damage.
Steam and Pipe Whip	Main Steam	11. Pipe break.	Opening (steam)	Rest of Reactor Building	(See Source and Mitigation table.)	Yes.	8 x 10^{-6}	(event tree)	
			Localized (pipe whip)						May impact emergency feedwater, main feedwater, and makeup piping.
Missiles	RCP or Pressurizer Missile	12.	Opening	RB-FZ-3	(See Source and Mitigation table.)	No, very unlikely.			May fail RL spray piping. May damage pressurizer.
Steam, Flood, and Pipe Whip	Reactor Coolant Piping	13. General openings. (pipe whip localized)		Rest of Reactor Building	(See Source and Mitigation table.)	No, additional failures not important.			
Smoke	Fire	14. General openings.		Rest of Reactor Building	(See Source and Mitigation table.)	No.	10^{-3}		Smoke does not have a short-term effect on safety equipment.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Reactor Building Inside Secondary Shield, West
Designator: RB-FZ-1e
Building: Reactor Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RC							X		1	RC-4B-TE2 cables (RG-61A).
RC							X		1	RC-4B-TE3 cables.
RC							X		1	RC-5A-TE2 cables (RE-178A and RE-177A).
RC							X		1	RC-5A-TE4 cables.
RC							X		1	RC-5B-TE2 cables (RE-182A and RE-177A).
RC							X		1	RC-5B-TE4 cables.
RC								RC-H-1B	1-FHA-017	Steam generator B.
RC					X			J17	1-FHA-017	J17 junction box.
RC					X				1-FHA-017	RE 182A for RC-5B-TE2.
RC					X				1-FHA-017	R661A for RC-4B-TE2.
DH			DH-V-1		X				1-FHA-017	Dropline isolation valve.
RC		RC-P-1C							1-FHA-017	Reactor coolant pump C.
RC		RC-P-1D							1-FHA-017	Reactor coolant pump D.
MS								Piping	Isometric Drawings	
FW								Piping	Isometric Drawings	
EF								Piping	Isometric Drawings	
MU								Piping	Isometric Drawings	
RCS								Piping	Isometric Drawings	

SOURCE AND MITIGATION TABLE

Location Name: Reactor Building Inside Secondary Shield, West
Designator: RB-FZ-1e
Building: Reactor Building

Page 1 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	RCP Motor and Lube Oil System		1, 1-FHA-017 through 1-FHA-022	RCP Oil Splash Guard and Reservoir	1, 1-FHA-017 through 1-FHA-022	
	Electric Cables		1, 1-FHA-017 through 1-FHA-022	Fire Hose Stations	1, 1-FHA-017 through 1-FHA-022	One near the shield door on the north boundary outside RB-FZ-1d. One at the top of the shield wall.
			1, 1-FHA-017 through 1-FHA-022	Ionization Fire Detection	1, 1-FHA-017 through 1-FHA-022	
			1, 1-FHA-017 through 1-FHA-022	Halon Fire Extinguishers	1, 1-FHA-017 through 1-FHA-022	Located outside the personnel access hatch on Elevation 308'0" of the turbine building.
			1, 1-FHA-017 through 1-FHA-022	Radiant Energy Heat Shields for Specified Cables	1, 1-FHA-017 through 1-FHA-022	See Reference 1.
			1, 1-FHA-017 through 1-FHA-022	DH-V1 Fire Protection	1, 1-FHA-017 through 1-FHA-022	Manually actuated fire suppression system with a single, closed head nozzle.
			1, 1-FHA-017 through 1-FHA-022	Walls	1, 1-FHA-017 through 1-FHA-022	
Flood	Main Feedwater Pipe Break		1-FHA-017 E-301-081 1-FHA-022	Drain Sump	1-FHA-017 E-301-081 1-FHA-022	
	Emergency Feedwater Pipe Break		1-FHA-017 E-301-081 1-FHA-022	Drain Sump	1-FHA-017 E-301-081 1-FHA-022	
	Makeup Pipe Break		1-FHA-017 E-301-081 1-FHA-022	Drain Sump	1-FHA-017 E-301-081 1-FHA-022	
	RCS Pipe Break		1-FHA-017 E-301-081 1-FHA-022	Drain Sump	1-FHA-017 E-301-081 1-FHA-022	

SOURCE AND MITIGATION TABLE

Location Name: Reactor Building Inside Secondary Shield, West
Designator: RB-FZ-1e
Building: Reactor Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Steam	Main Steam Pipe Break		1-FHA-017 E-301-081 1-FHA-022	RBS Reactor Building Emergency Cooling	1-FHA-017 E-301-081 1-FHA-022	
	Main Feedwater Pipe Break		1-FHA-017 E-301-081 1-FHA-022	RBS Reactor Building Emergency Cooling	1-FHA-017 E-301-081 1-FHA-022	
	Makeup Pipe Break		1-FHA-017 E-301-081 1-FHA-022	RBS Reactor Building Emergency Cooling	1-FHA-017 E-301-081 1-FHA-022	
	RCS Pipe Break		1-FHA-017 E-301-081 1-FHA-022	RBS Reactor Building Emergency Cooling	1-FHA-017 E-301-081 1-FHA-022	
Missile	RCP Missile Ejection		1-FHA-017 E-301-081 1-FHA-022	Walls	1-FHA-017 E-301-081 1-FHA-022	
Pipe Whip	Main Steam		Plant Visit	Pipe Supports	Plant Visit	
	Main Feedwater		Plant Visit	Walls	Plant Visit	
	Emergency Feedwater		Plant Visit		Plant Visit	
	Makeup		Plant Visit		Plant Visit	
	RCS		Plant Visit		Plant Visit	

SCENARIO TABLE

Location Name: Reactor Building Inside Secondary Shield, West
Designator: RB-FZ-1e
Building: Reactor Building

Sheet 1 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire	RCP Motor Lube Oil System, Cables, or Transient Fuel	1. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	Localized. Confined to room only.		(See Source and Mitigation table.)	Yes	10^{-3}	(comparison)	Only cables are damaged; fire is severe but mechanical equipment, such as pipes and valves, remain functional (valve motors would fail). No severe structural damage can be envisioned because concrete walls are very thick and no important structural parts are immediately above the RCPs.
		2. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-1a	(See Source and Mitigation table.)	No, very unlikely.			
		3. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-1b	(See Source and Mitigation table.)	No, very unlikely.			
		4. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-1c	(See Source and Mitigation table.)	No, very unlikely.			
		5. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-1d	(See Source and Mitigation table.)	No, very unlikely.			
		6. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-2	(See Source and Mitigation table.)	No, very unlikely.			

SCENARIO TABLE (continued)

Location Name: Reactor Building Inside Secondary Shield, West
Designator: RB-FZ-1e
Building: Reactor Building

Sheet 2 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
		7. RCP oil leaks out and ignites on hot surfaces, or other combustibles ignite from internal causes.	General openings.	RB-FZ-3	(See Source and Mitigation table.)	No, additional damages not important.			
Flood, Steam, and Pipe	Main Feedwater	8. Pipe break.	General openings.	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)		8×10^{-6}	(ET)	About 9 feet of water on the floor; steam jet may damage cables; assume that makeup and emergency feedwater supply in the area affected.
	Makeup	9. Pipe break.	General openings.	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)		2×10^{-5} (Two pipe sections)		
	RCS	10. Pipe break.	General openings.	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)		8×10^{-6}		
Steam and Pipe Whip	Main Steam	11. Pipe break.	General openings.	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)		8×10^{-6}	(ET)	
Missile	RCP Missile	12. Pipe break.	General openings.	RB-FZ-3 (pipe whip localized)	(See Source and Mitigation table.)		10^{-5}		
Flood and Pipe Whip	Emergency Feedwater	13. Pipe break.	General openings.	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)	No, very unlikely since pipe in standby.			About 9 feet of water on the floor.

SCENARIO TABLE (continued)

Location Name: Reactor Building Inside Secondary Shield, West
Designator: RB-FZ-1e
Building: Reactor Building

Sheet 3 of 3

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Smoke	Fire	14. See five sources.	General Openings	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)		10^{-3}		Smoke does not have a short-term effect on safety equipment.

LOCATION INVENTORY CATION TABLE

Location Name: Reactor Building Outside Secondary Shield
Designator: RB-FZ-2
Building: Reactor Building

Sheet 1 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RC					X				1	Pressurizer heater group 8.
RC					X				1	Pressurizer heater group 9.
SP							X		1	SP-6A-PT1 cables.
SP							X		1	SP-6A-PT2 cables.
SP							X		1	SP-6B-PT1 cables.
SP							X		1	SP-6B-PT2 cables.
RC							X		1	RC-3A-PT3 cables.
RC							X		1	RC-3A-PT4 cables.
RC							X		1	RC-3B-PT3 cables.
AH								AH-E-2A	1-FHA-018	Reactor compartment ventilation unit A.
AH								AH-E-2B	1-FHA-018	Reactor compartment ventilation unit B.
AH								SM-12-AH-F-12	1-FHA-018	Kidney filter plenum.
CF								CF-T-1A	1-FHA-018	Core flooding tank A.
CF								CF-T-1B	1-FHA-018	Core flooding tank B.
SP								Tray 800	1-FHA-018	Cable tray.*
SP								Tray 823	1-FHA-018	Cable tray.*
ES								J24	1-FHA-018	J24 junction loop.
RC					X				1-FHA-018	RG 16A cable for RC-3A-PT3 and RC-3B-PT3.
RC					X				1-FHA-018	RG 201A cable for RC-3A-PT3 and RC-3B-PT3.
SP					X				1-FHA-018	RE 109A cable for SP-6A-PT1.
SP					X				1-FHA-018	RE 110A cable for SP-6B-PT1.
SP/RC								Tray 815	1-FHA-018	Cable tray.**
SP/RC					X				1-FHA-018	RG 202A cable for RC-3A-PT3 and RC-3B-PT3.

* Includes cables for SP-6B-PT1 and SP-6B-PT2.
**Includes cables for SP-1B-LT1, SP-6B-PT1, SP-6B-PT2, RC-5A-TE2, RC-5A-TE4, RC-5B-TE2, RC-5B-TE4, and RC-1-LT1.

LOCATION INVENTORY CODIFICATION TABLE (continued)

Location Name: Reactor Building Outside Secondary Shield
Designator: RB-FZ-2
Building: Reactor Building

Sheet 2 of 2

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
IC			IC-V-2						C. Adams Letter 6/19/84	Intermediate cooling return isolation.
IC			IC-V-79A						C. Adams Letter 6/19/84	RCP-1A cooler inside containment outlet isolation.
IC			IC-V-79B						C. Adams Letter 6/19/84	RCP-1B cooler inside containment outlet isolation.
IC			IC-V-79C						C. Adams Letter 6/19/84	RCP-1C cooler inside containment outlet isolation.
IC			IC-V-79D						C. Adams Letter 6/19/84	RCP-1D cooler inside containment outlet isolation.
MS								Piping	Isometric Drawings	
FW								Piping	Isometric Drawings	
EF								Piping	Isometric Drawings	
MU								Piping	Isometric Drawings	
CF								Piping	Isometric Drawings	
BS								Piping	Isometric Drawings	
DH				DH-V-1		X			5-31	
DH				DH-V-2					5-31	

SOURCE AND MITIGATION TABLE

Location Name: Reactor Building Outside Secondary Shield
Designator: RB-FZ-2
Building: Reactor Building

Page 1 of 2

Source Type	Source Description			Mitigation of Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Charcoal in the Kidney Filter Plenum		1, 1-FHA-018 Plant Visit	Water Fire Extinguisher	1, 1-FHA-018 Plant Visit	One in RB-FZ-2.
	Electric Cables		1, 1-FHA-018 Plant Visit	Fire Hose Stations	1, 1-FHA-018 Plant Visit	Two in RB-FZ-2.
	Concrete Coating		1, 1-FHA-018 Plant Visit	Ionization Fire Detection	1, 1-FHA-018 Plant Visit	
				Self-Contained Automatic Deluge Water System	1, 1-FHA-018 Plant Visit	For charcoal in the kidney filter plenum.
				Radiant Energy Heat Shields for Specified Cables	1, 1-FHA-018 Plant Visit	See Reference 1.
Flood	Main Feedwater Pipe Break		1-FHA-018 1-FHA-022 Plant Visit	Drain Sump in RB-FZ-1c	1, 1-FHA-018, Plant Visit	
	Emergency Feedwater Pipe Break		1-FHA-018 1-FHA-022 Plant Visit	Drain Sump in RB-FZ-1c	1, 1-FHA-018, Plant Visit	
	Makeup Pipe Break		1-FHA-018 1-FHA-022 Plant Visit	Drain Sump in RB-FZ-1c	1, 1-FHA-018, Plant Visit	
	Core Flood Pipe/Tank Break		1-FHA-018 1-FHA-022 Plant Visit	Drain Sump in RB-FZ-1c	1, 1-FHA-018, Plant Visit	
	RBS Pipe Break or Inadvertent Actuation		1-FHA-018 1-FHA-022 Plant Visit	Drain Sump in RB-FZ-1c	1, 1-FHA-018, Plant Visit	
Steam	Main Steam Pipe Break		1-FHA-018 1-FHA-022 Plant Visit	RBS Reactor Building Emergency Cooling	Plant Visit	
	Main Feedwater Pipe Break		1-FHA-018 1-FHA-022 Plant Visit	RBS Reactor Building Emergency Cooling	Plant Visit	

SOURCE AND MITIGATION TABLE (continued)

Location Name: Reactor Building Outside Secondary Shield
Designator: RB-FZ-2
Building: Reactor Building

Page 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
	Emergency Feedwater Pipe Break		1-FHA-018 1-FHA-022 Plant Visit	RBS Reactor Building Emergency Cooling	Plant Visit	
	Makeup Pipe Break		1-FHA-018 1-FHA-022 Plant Visit	RBS Reactor Building Emergency Cooling	Plant Visit	
	Core Flood Pipe/Tank Break		1-FHA-018 1-FHA-022 Plant Visit	RBS Reactor Building Emergency Cooling	Plant Visit	
Pipe Whip	Main Steam		1-FHA-018 1-FHA-022 Plant Visit	Pipe Supports	Plant Visit	
	Main Feedwater		1-FHA-018 1-FHA-022 Plant Visit	Walls	Plant Visit	
	Emergency Feedwater		1-FHA-018 1-FHA-022 Plant Visit	Walls	Plant Visit	
	Makeup		1-FHA-018 1-FHA-022 Plant Visit	Walls	Plant Visit	
	Core/Flood		1-FHA-018 1-FHA-022 Plant Visit	Walls	Plant Visit	

SCENARIO TABLE

Location Name: Reactor Building Outside Secondary Shield
Designator: RB-FZ-2
Building: Reactor Building

Sheet 1 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire	Charcoal in the Kidney Filter Plenum, Cables, Concrete Coating, or Transient Fuels	1.	Localized.		(See Source and Mitigation table.)	No.	10^{-3}		Impact on plant safety minimal.
		2.	General openings.	RB-FZ-1a	(See Source and Mitigation table.)	No, very unlikely.			
		3.	General openings.	RB-FZ-1b	(See Source and Mitigation table.)	No, very unlikely.			
		4.	General openings.	RB-FZ-1c	(See Source and Mitigation table.)	No, very unlikely.			
		5.	General openings.	RB-FZ-1d	(See Source and Mitigation table.)	No, very unlikely.			
		6.	General Openings.	RB-FZ-1e	(See Source and Mitigation table.)	No, very unlikely.			
		7.	General openings.	Rb-FZ-3	(See Source and Mitigation table.)	No, additional failures not important.			
Flood	Reactor Building Spray	8.	General openings.	Rest of Reactor Building	(See Source and Mitigation table.)	No, exposed equipment can take spray.	10^{-2}		
Flood and Pipe Whip	Emergency Feedwater Core Flood	9.	General openings.	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)		2×10^{-5} 8×10^{-6} $\times 2$ (two pipe sections)		About 9 feet of water on the floor.
		10.	General openings.	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)	No, impact on plant safety minimal.	2×10^{-5} (two pipe sections)		

SCENARIO TABLE (continued)

Location Name: Reactor Building Outside Secondary Shield
Designator: RB-FZ-2
Building: Reactor Building

Sheet 2 of 2

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Steam and Pipe Whip	Main Steam	11. Pipe break.	General Openings.	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)	Yes.	2×10^{-5} (two pipe sections)	(event tree)	Steam jets or pipe movement may damage local cables. See impact table.
Steam, Flood, and Pipe Whip	Main Feedwater	12. Pipe break.	General openings.	Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)	Yes.	2×10^{-5}	(no action) Impact similar to scenario 11.	About 9 feet of water on the floor.
Smoke	Fire	13. Pipe break.	General openings.	Rest of Reactor Building	(See Source and Mitigation table.)	No.	10^{-3}		Impact of smoke on exposed equipment long-term only.

IMPACT TABLE

Location Name:	Reactor Building Outside Secondary Shield
Designator:	RB-FZ-2
Building:	Reactor Building
Scenario Summary:	Steam and Pipe Whip; Scenario 11; Pipe Break in Main Steam Line Piping; Steam Jets and Pipe Movement Impacts Pipes and Cables

Systems Lost	Components Affected by the Hazard
One OTSG Dry	Main steam line break, one pipe.
Instrumentation (large number of channels)	Instrumentation cable failed from steam jet and pipe movement.
IC/All	IC piping and IC-V-2 (single line feeding all four RCPs).
EF to One OTSG	One emergency feedwater pipe.
FW to One OTSG	Main feedwater pipe affected from steam pipe movement.
	Makeup pipe affected from steam pipe movements.

LOCATION INVENTORY CODIFICATION TABLE

Location Name: Reactor Building Inside and Outside Secondary Shield
Designator: RB-FZ-3
Building: Reactor Building

System/ Train	Train or Safety Division	Pump	Valve	Electrical Cabinet	Cables			Other Items	Reference	Remarks/Assumptions
					Power	Control	Instrumentation			
RC							X		1	RC-3A-PT3 cables.
RC							X		1	RC-3A-PT4 cables.
RC							X		1	RC-3B-PT3 cables.
NI							X		1	NI-1 cables.
NI							X		1	NI-2 cables.
RC								RC-T-1	1-FHA-017	Reactor vessel.
RC							X		1-FHA-020	RC-3A-PT2 cables.
FH								FH-A-1	1-FHA-023	Fuel handling bridge.
FH								FH-A-2	1-FHA-023	Fuel handling bridge.
FH								X	1-FHA-023	Incore instruction jib crane.
FH								X	1-FHA-023	CDR service jib crane.
FH								X	1-FHA-024	Reactor building crane.
RC					X				1-FHA-020	RG 201A cable for RC-3A-PT1.
RC					X				1-FHA-020	RG 202A cables.
RBS								Piping	Isometric Drawings	
RCS								Piping	Isometric Drawings	

SOURCE AND MITIGATION TABLE

Location Name. Reactor Building Inside and Outside Secondary Shield
Designator: RB-FZ-3
Building: Reactor Building

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Fire and Smoke	Electric Cables		1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	Fire Hose Stations	1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	Four in RB-FZ-3.
			1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	Ionization Fire Detection	1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	
			1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	Dry Chemical Fire Extinguisher	1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	One in RB-FZ-3.
Flood	RBS Pipe Break or Inadvertent Actuation		1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	Drain Sump in RB-FZ-1c	1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	
	RCS Pipe/Vessel Break		1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024		1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	
Steam	RCS Pipe/Vessel Break		1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	RBS Reactor Building Emergency Cooling	1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	
Missiles	CRDM Ejection		1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	CRDM Missile Shield	1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	See 1-FHA-022.
Falling Objects	Crane		1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	Walls CRDM Missile Shield	1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024	

SOURCE AND MITIGATION TABLE (continued)

Location Name: Reactor Building Inside and Outsidr Secondary Shield
Designator: RB-FZ-3
Building: Reactor Building

Page 2 of 2

Source Type	Source Description			Mitigation of the Source		Remarks
	Description	Assumptions	Reference	Mitigative Feature	Reference	
Pipe Whip	RCS Piping		1, 1-FHA-020 1-FHA-021 1-FHA-022 1-FHA-023 1-FHA-024 Plant Visit	Pipe Supports Walls	Plant Visit	

SCENARIO TABLE

Location Name: Reactor Building Inside and Outside Secondary Shield
Designator: RB-FZ-3
Building: Reactor Building

Source Type	Synopsis of the Source	Scenario				Considered for Further Analysis	Frequency (yr^{-1})	Summary of Quantification Results and Further Actions	Remarks
		Source Portion	Paths of Propagation		Mitigation Portion				
			Type	To					
Fire	Electric Cables	1. Localized.			(See Source and Mitigation table.)	No, impact not important.			
		Confined room only.			(See Source and Mitigation table.)				
		2. General openings.		RB-FZ-1a	(See Source and Mitigation table.)	No, very unlikely.			
		3. General openings.		RB-FZ-1b	(See Source and Mitigation table.)	No, very unlikely.			
		4. General openings.		RB-FZ-1c	(See Source and Mitigation table.)	No, very unlikely.			
		5. General openings.		RB-FZ-1d	(See Source and Mitigation table.)	No, very unlikely.			
		6. General openings.		RB-FZ-1e	(See Source and Mitigation table.)	No, very unlikely.			
		7. General openings.		RB-FZ-2	(See Source and Mitigation table.)	No, very unlikely.			
Flood	RBS	8. General openings.		Rest of Reactor Building	(See Source and Mitigation table.)	No, impact unimportant.	10^{-2}		
Steam, Flood, and Pipe Whip	RCS	9. General openings.		Rest of Reactor Building (pipe whip localized)	(See Source and Mitigation table.)	Yes.	8×10^{-6}	(no action) Considered as part of initiating events.	No pipe whip or other important failures.
Missiles	CRDM	10. Confined to RB-FZ-3.			(See Source and Mitigation table.)	No, impact not important.	10^{-5}		
Falling Objects	Crane	11.			(See Source and Mitigation table.)	No, crane operated only during cold shutdown.	Insignificant		Crane not operating during power operations.
Smoke	Fire	12. General openings.		Rest of Reactor Building	(See Source and Mitigation table.)	No, impact long-term and minimal.	10^{-3}		