Risk Assessment of Operational Events

Handbook

Volume 2 – External Events

Internal Fires – Internal Flooding – Seismic – Other External Events
Frequencies of Seismically-Induced LOOP Events

Revision 1.01

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SDP Phase 3 ● ASP ● MD 8.3

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

ACRONYMS

ac	alternating current
AFW	auxiliary feedwater
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor
BWR	boiling water reactor
CCDP	conditional core damage probability
CCW	component cooling water
CDF	core damage frequency
CDP	core damage probability
DBE	design basis earthquake
dc	direct current
EDG	emergency diesel generator
EFW	emergency feedwater
ESW	emergency service water
FEMA	Federal Emergency Management Agency
FLI	internal flooding
FP	fire protection
FT	fault tree
GEM (code)	Graphical Evaluation Module (code)
HCLPF	high confidence of low probability of failure
HEP	human error probability
HVAC	heating, ventilation and air conditioning
IE	initiating event
IEfreq	initiating event frequency
IMC 0309	Inspection Manual Chapter 0309
IPEEE	Individual Plant Examination of External Events
LER	licensee event report
LERF	large early release frequency
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LOSWS	loss of service water system
LPSI	low-pressure safety injection
MCR	main control room
MD 8.3	Management Directive 8.3
MFW	main feedwater
MLOCA	medium loss-of-coolant accident

NOAA	National Oceanic and Atmospheric Administration
NPP	nuclear power plant
NSW	nuclear service water
OBE	operating-basis earthquake
PCS	power conversion system
pga	peak ground acceleration
PMH	probable maximum hurricane
PMP	probable maximum precipitation
PORV	power-operated relief valve
PWR	pressurized water reactor
RASP	Risk Assessment of Operational Events Handbook
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RWST	refueling water storage tank
SA	spectral acceleration
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SBO	station blackout
SDP	Significance Determination Process
SG	steam generator
SLOCA	small loss-of-coolant accident
SMA	seismic margins analysis
SPAR (model)	Standardized Plant Analysis Risk (model)
SPRA	seismic probabilistic risk analysis (model)
SRA	senior reactor analyst
SRV	solenoid relief valve
SSC	structures, systems and components
SSE	safe shutdown earthquake
SW	service water
USGS	U.S. Geological Survey

1.0 Introduction

1.1 Objectives

The first objective of the Risk Assessment of Operational Events Handbook (sometimes known as "RASP Handbook" or "handbook") was to document methods and guidance that NRC staff could use to achieve more consistent results when performing risk assessments of operational events and licensee performance issues.

The second objective was to provide analysts and Standardized Plant Analysis Risk (SPAR) model developers with additional guidance to ensure that the SPAR models used in the risk analysis of operational events represent the as-built, as-operated plant to the extent needed to support the analyses.

This handbook represents best practices based on feedback and experience from the analyses of over 600 precursors in the Accident Sequence Precursor (ASP) Program (since 1969) and numerous Significance Determination Process (SDP) Phase 3 analyses (since 2000).

1.2 Scope of the Handbook

The scope of the handbook is provided below.

- ***Applications.*** The methods and processes described in the handbook can be primarily applied to risk assessments for Phase 3 of the SDP, the ASP Program, and event assessments under the NRC's Incident Investigation Program (in accordance with Management Directive 8.3). The guidance for the use of SPAR models and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software package can be applied in the risk analyses for other regulatory applications, such as the Generic Safety Issues Program and special risk studies of operational experience.

- ***Relationships to program requirements.*** This handbook is intended to provide guidance for implementing requirements contained in program-specific procedures, such as Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," and IMC 0309, "Reactive Inspection Decision Basis for Reactors." It is not the scope of this handbook to repeat program-specific requirements in the handbook, since these requirements may differ between applications and may change as programs evolve. Program-specific requirements supersede guidance in this handbook.

- ***Deviations from methods and guidance.*** Some unique events may require an enhancement of an existing method or development of new guidance. Deviations from methods and guidance in this handbook may be necessary for the analysis of atypical events. However, such deviations should be adequately documented in the analysis to allow for the ease of peer review. Changes in methodologies and guidance may be reflected in future revisions of this handbook.

1.3 Audience for the Handbook

The principal users of this handbook are senior reactor analysts (SRAs) and headquarters analysts involved with the risk analysis of operational events. It is assumed that the analysts using this handbook have received PRA training at the SRA qualification level. The analyst using this handbook should be familiar with the risk analysis of operational events, SAPHIRE software package, and key SPAR model assumptions and technical issues. Although, this handbook could be used as a training guide, it is assumed that the analyst either has completed the NRC course "Risk Assessment in Event Evaluation (Course Number P-302) or has related experience.

1.4 Handbook Content

The revised handbook includes three volumes, designed to address Internal Events (Volume 1), External Events (Volume 2), and SPAR Model Reviews (Volume 3). The scope of these volumes is as follows:

- ***Volume 1, Internal Events.*** Volume 1, "Internal Events," provides generic methods and processes to estimate the risk significance of initiating events (e.g., reactor trips, losses of offsite power) and degraded conditions (e.g., a failed high pressure injection pump, failed emergency power system) that have occurred at nuclear power plants.[1]

  Specifically, this volume provides guidance on the following analysis methods:

  - Exposure Time Determination and Modeling
  - Failure Determination and Modeling
  - Mission Time Modeling
  - Test and Maintenance Outage Modeling
  - Recovery of Failed Equipment Modeling
  - Multi-Unit Considerations Modeling

  In addition, the appendices provide further guidance on the following analysis topics:

  - Roadmap - Risk Analysis of Operational Events
  - Quick Reference Guide – SAPHIRE Version 7

  Although, the guidance in this volume of the handbook focuses on the analysis of internal events during at-power operations, the basic processes for the risk analysis of initiating events and degraded conditions can be applied to external events, as well as events occurring during low-power and shutdown operations. A future revision of the handbook will integrate all volumes of the handbook.

- ***Volume 2, External Events.*** Volume 2, "External Events," provides methods and guidance for the risk analysis of initiating events and conditions associated with external events. External events include internal flooding, internal fire, seismic, external flooding,

[1] In this handbook, "initiating event" and "degraded condition" are used to distinguish an incident involving a reactor trip demand from a loss of functionality during which no trip demand occurred. The terms "operational event" and "event," when used, refer to either an initiating event or a degraded condition.

external fire, high winds, tornado, hurricane, and others. This volume is intended to complement Volume 1 for Internal Events.

Specifically, this volume provides the following guidance:

- Internal Flood Modeling and Risk Quantification
- Internal Fire Modeling and Risk Quantification
- Seismic Event Modeling and Seismic Risk Quantification
- Other External Events Modeling and Risk Quantification

Volumes 1 and 2 update the staff guidance that was provided for trial use in 2005 and 2006, respectively.

- ***Volume 3, SPAR Model Reviews.*** Volume 3, "SPAR Model Reviews," provides analysts and SPAR model developers with additional guidance to ensure that the SPAR models used in the risk analysis of operational events represent the as-built, as-operated plant to the extent needed to support the analyses. This volume provides checklists that can be used following modifications to SPAR models that are used to perform risk analysis of operational events. These checklists were based on the PRA Review Manual (NUREG/CR-3485, Ref. 1-1), the ASME PRA Standard (ASME RA-S-2005, Ref. 1-2), Regulatory Guide 1.200 (Ref. 1-3), and experiences and lessons learned from the SDP and ASP analyses.

  In addition, this volume summarizes key assumptions in a SPAR model and unresolved technical issues that may produce large uncertainties in the analysis results. The importance of these assumptions or issues depends on the sequences and cut sets that were impacted by the operational event. Additionally, plant-specific assumptions and issues may play an even larger role in the analysis uncertainties.

1.5 Companion References to the Handbook

Guidance in the three volumes of the handbook often refers to other references, as applicable to the application. A bibliography of current technical references used in the risk analysis of operational events is provided in Volume 3, in which most of the documents are referenced in individual sections throughout the handbook.

Key companion references that are an extension to this handbook include:

- NUREG/CR-6268, Rev. 1, "Common-Cause Failure Database and Analysis System: Event Data Collection, Classification, and Coding" (Ref. 1-4)
- NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology" (Ref. 1-5)
- NUREG/CR-6883, "SPAR-H Human Reliability Analysis Method" (Ref. 1-6)
- Handbook for Phase 3 Fire Protection (FP) Significance Determination Process (SDP) Analysis (Ref. 1-7)
- Basic SAPHIRE training manual (Ref. 1-8)
- Advanced SAPHIRE training manual (Ref. 1-9)
- Plant-specific SPAR model manual

1.6 Future Updates to the Handbook

It is intended that this handbook will be updated on a periodic and as-needed basis, based on user comments and insights gained from “field application” of the document. New topics will also be added as needed, and the handbook can also be re-configured and/or reformatted based on user suggestions.

- ***Revision 2 plans.*** Current plans for Revision 2 of the handbook will include the following additional method guides and tutorials:

  Methods

  - Common-Cause Failure Determination and Modeling
  - SPAR-H Human Reliability Analysis Method
  - Parameter Estimation and Update Methods
  - Convolution of Failure to Run Parameters Method
  - Uncertainty Analysis Method
  - Simplified Expert Elicitation Method

  Tutorials and examples

  - Internal Events Modeling of Conditions and Initiating Events – Examples
  - Quick Reference Manual – SPAR Models
  - Tutorial - Common-Cause Failure Modeling
  - Tutorial - NRC's Risk Databases and Calculators

- ***Future volumes.*** Two additional volumes are planned in the near future:

  - Risk Analysis of Low-Power and Shutdown Events
  - Risk Analysis of Events Involving Containment-Related Events (LERF)

1.7 Questions, Comments, and Suggestions

Questions, comments, and suggestions should be directed to the following:

<u>Internal NRC staff and NRC contractors</u>:

- Volume 1, Internal Events – Don Marksberry, 301-415-6378, dgm2@nrc.gov
- Volume 2, External Events – Selim Sancaktar, 301-415-8184, sxs9@nrc.gov
- Volume 3, SPAR Model Reviews – Peter Appignani, 301-415-6857, pla@nrc.gov

<u>External NRC (e.g., public, licensees)</u>:

- All handbook volumes; Significant Determination Process – Paul Bonnett, 301-415-4107, fpb@nrc.gov

1.8 References

1-1. U.S. Nuclear Regulatory Commission, "PRA Review Manual," NUREG/CR-3485, September 1985.

1-2. American Society of Mechanical Engineers, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-S-2005, 2005.

1-3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, January 2007. http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/active/01-200/01-200r1.pdf

1-4. U.S. Nuclear Regulatory Commission, "Common-Cause Failure Database and Analysis System: Event Data Collection, Classification, and Coding," NUREG/CR-6268, Rev. 1, September 2007. http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6268/

1-5. U.S. Nuclear Regulatory Commission, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology," NUREG/CR-6850, September 2005. http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/

1-6. U.S. Nuclear Regulatory Commission, "The SPAR-H Human Reliability Analysis Method," NUREG/CR-6883, August 2005.

1-7. U.S. Nuclear Regulatory Commission, "Handbook for Phase 3 Fire Protection (FP) Significance Determination Process (SDP) Analysis," December 2005. (ADAMS Accession Number ML053620267)

1-8. Idaho National Laboratory, "SAPHIRE Basics - An Introduction to Probabilistic Risk Assessment via the Systems Analysis Program for Hands-On Integrated Reliability Evaluations (SAPHIRE) Software," January 2005 or current revision.

1-9. Idaho National Laboratory, "Advanced SAPHIRE - Modeling Methods for Probabilistic Risk Assessment via the Systems Analysis Program for Hands-On Integrated Reliability Evaluations (SAPHIRE) Software," March 2005 or current revision.

2.0 Internal Fire Modeling and Fire Risk Quantification

2.1 Objectives and Scope

- ***Objectives.*** This document provides methods and guidance for risk analysis of initiating events and conditions associated with internal plant fire scenarios. In addition, this handbook provides guidance for modeling scenarios related potential internal plant fire event initiators, and quantifying their sequence frequency estimates using SPAR models and SAPHIRE software. This volume of the handbook complements Volume 1 for internal events (Ref. 2-1).

- ***Scope.*** This handbook provides guidance for the analysis of the following types of operational events:

    - Conditions related to degraded fire protection structures, systems, and components (SSC) (e.g., fire suppression system, fire-rated barrier, smoke detection system).

    - Conditions related to degraded SSC other than fire protection SSCs in which associated baseline accident sequence frequencies are heavily influenced by postulated fire scenarios (e.g., risk-important cables running through the room of a redundant train).

    - Fire initiators where a reactor trip may or may not have been caused by the fire.

  Note that fire-induced initiating events may be best modeled using an internal events SPAR model in which an appropriate internal event initiator is set to TRUE (e.g., loss of offsite power, loss of main feedwater). Also, for those conditions related to degraded fire protection structures, systems, and components, an analysis using the fire protection SDP, as documented in IMC 0609 Appendix F (Ref. 2-5), would aid in the identification of fire scenario characteristics and fire effects.

- ***Alternative guidance.*** The following additional guides may be used in SDP Phase 3 and ASP analyses as an alternative to the guidance presented in this volume of the RASP Handbook:

    - *NUREG/CR-6850.* This volume of the RASP Handbook simplifies the detailed guidance provided in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology," (Ref. 2-2) for performing a fire risk analysis. In certain cases, a more detailed analysis as provided in NUREG/CR-6850 may be better suited for modeling fire scenarios in risk-important areas in the plant.

    - *Handbook for Phase 3 Fire Protection SDP Analysis.* Guidance provided in "Handbook for Phase 3 Fire Protection SDP Analysis" (Ref. 2-3) may be used as an alternative to this volume of the RASP Handbook. Reference 2-3 also simplifies the detailed guidance in NUREG/CR-6850. The analysis process in Ref. 2-3 essentially follows a back-end approach (i.e., analysis starts with the mitigation process and works through the detection process and fire frequency estimation).[2] This approach may be better suited when uncertainties associated with the fire source and frequencies may be larger than those associated with detection and mitigation probabilities. The need to justify a fire frequency in this case with its uncertainty may be reduced.

This handbook attempts to avoid repeating guidance in that reference for the front end fire analyses that are used to define fire scenarios. However, for convenience of the reader, some of the data from NUREG/CR-6850 (Ref. 2-2) is duplicated in this handbook.

2.2 Fire Scenario Definition and Quantification

A two-step process is discussed to model fire scenarios and quantify their core damage frequencies (CDFs):

1. Define fire scenarios that could lead to core damage, using applicable cases in Appendix 2A; calculate scenario frequencies. Definition of a fire scenario is discussed in Section 2.2.1.

2. Quantify the CDF of sequences resulting from these scenarios using a SPAR model and the SAPHIRE software. For this purpose, first the scenario-induced conditional core damage probability (CCDP) is calculated. Then this CCDP is multiplied by the scenario frequency calculated in Step 1 to obtain a fire sequence CDF. From a single fire ignition source or a single fire area fire, multiple scenarios may be derived, leading to multiple fire sequences whose CDFs need to be summed. Quantification of sequence CDF is discussed in Section 2.2.2.

2.2.1 Define Fire Scenarios

For the event (or plant condition) in question, one or more fire scenarios must be defined. These scenarios would consider ignition frequency, severity, non-suppression, spurious actuation, propagation to other fire areas, etc., but will not include plant safety and non-safety system responses to a postulated trip: this aspect of the fire-induced CDF sequence will be considered in by calculating the CCDP of the plant response to the fire scenario.

Note that a single ignition source (or a fire in an area) may produce multiple fire scenarios.

- ***Fire scenario cases.*** Depending upon the issue, the following cases are envisioned and are included in the scope:

  - Fires limited to one fire area.
  - Fires that can propagate into a second fire area (due to fire barrier failures).
  - Fires that can cause spurious actuations.
  - Main control room fires.
  - Containment fires.

[2] The analysis process taken in this volume of the RASP Handbook follows the front-end approach (i.e., analysis starts with fire frequency estimation and works through the detection and mitigation processes).

Fire scenarios for many of these cases assume specific configurations relative to the hazards, fire protection features and systems, and spatial considerations such as room size. For example, credit for fire suppression systems at the nominal value imply that the system is properly designed and installed for the hazard. Credit for separation to protect the redundant train in the case where fixed suppression is failed is highly dependent on the fire hazard and room size, which determines whether a hot gas layer can develop. Some probabilities assigned in the event trees for fire reflect specific configurations, which if changed, could affect the assigned probabilities significantly. As a result, the analyst should verify if the configuration which is being analyzed reflects the likelihood of failure of the fire protection feature and systems which are identified in the event trees for these fires.

A systematic method to define fire scenarios that fit into one of these cases, using simple event tree logic is given in Appendix 2A. Those fire scenarios that can lead to core damage are selected and their CDFs are quantified, as discussed in this section.

- ***Fire scenario frequency.*** The initiating event frequency (IEfreq) of a fire scenario can be simply defined as

  IEfreq = Ffi * SF * Pns, where

  Ffi= fire ignition frequency
  SF = Severity factor
  Pns = Non-suppression probability.

  Other scenario-specific factors can be introduced to the above equation, as warranted (e.g., probability of fire propagation to second train). See the example in Section 2.3.3 for such an additional factor introduced into the equation.

- ***Fire scenario summary table.*** Examine the event/condition characteristics and refer to the applicable appendices of this document accordingly. Select the fire scenarios that lead to core damage accident sequences and summarize those sequences in terms of a table, such as Table 2-1. The columns of this table are discussed below. Note that, each fire ignition event is treated as an initiating event that will be assigned an event tree.

  1. *Scenario name (initiating event ID).* This always starts with FRI- and is used both for the event tree and the initiating event names.

  2. *Scenario description.*

  3. *Scenario IEfreq.* This is calculated using models discussed in Appendices 2A-1 through 2A-5.

  4. *Equipment lost.* Equipment credited in the probabilistic risk assessment (PRA) that is lost due to fire are listed in this column.

  5. *Initiating event caused.* This is the initiating event caused by the fire. In most cases, it is one of the internal initiating event categories already defined (e.g., loss of main feedwater (LOMFW), reactor trip (TRANS), loss of offsite power (LOOP), loss-of-coolant accident (LOCA)). In some cases, such as in main

control room (MCR) fire, a new event tree model needs to be developed to model the operation of the plant from the remote shutdown panel. In that case, put the name of the new event tree in this column (scenarios 3 through 8 refer to such new event trees in the example below).

The guidance in the following table can be used to determine the initiating event caused by the fire for select cases.

Initiating Event Caused by the Fire for Select Cases.

If Fire Causes	Initiating Event Caused
No spurious opening of reactor coolant system (RCS) valves; No main control room evacuation; and No LOOP.	Transient
Turbine building fire that damages MFW or condenser system equipment	Transient with loss of MFW
Spurious opening of RCS system valve(s) (e.g., power-operated relief valve (PORV), solenoid relief valve)	LOCA (LOCA size depend on the number and size of valves)
Equipment damage (e.g., bus, transformer) leading to LOOP; Self induced LOOP by operators by fire procedures	Loss of Offsite Power
Reactor shutdown from remote shutdown panel after main control room evacuation	Make special event tree model

6. *Human error probabilities (HEPs) and other basic events affected.* List the basic events and operator actions that are affected by the fire (failed, degraded). This is in addition to equipment listed in Column 4. Considerations about operator actions are provided in Appendix 2G.

7. *New basic events (failures) introduced.* List any new basic events introduced (such as scenario initiating event frequencies) to model the scenarios.

Other columns may be introduced as needed.

Table 2-1 Example Summary of Fire Scenarios

	Name	Description	IE Frequency	Equipment Lost	Initiating Event Caused	HEPs / Other BEs Affected	New Basic Events (Failures) Introduced
	1	2	3	4	5	6	7
1	FRI-FI1	Auxiliary Building MCC 1-62J Room	2.63E-4	Valve BT 2B Valve MS 100B MCC 62J (all three affecting AFW)	TRANS	None / None	IE-FRI-FI1
3	FRI-FI3	4.16KV SWGR Room 16 buses 1 and 2, beneath cable tray 1AT9N	9.5E-05	MFW pumps A & B	LOMFW	None / None	IE-FRI-FI3
4	FRI-FI4	Diesel B Oil Fire	8.9E-3	RAT EDG B BUS 6	FRI-MCR-E-0-07	None / None **	IE-FRI-FI4 EPS-XHE-DSP AFW-XHE-DSP SWS-B1-B2-FAIL SWS-XHE-DSP
5	FRI-FI5	Fire in Relay room	6.78E-7	BUS 6 TAT Valve BT3A (AFW)	FRI-DSP	***	IE-FRI-FI5
6	FRI-FI6	Turbine Building AFW Pump A oil fire	6.45E-4	AFW MDP A BUS 5	FRI-MCR-E-0-07		IE-FRI-FI6
8	FRI-FI8	Fire Near buses 51 and 52	4.65E-05	BUS 5	FRI-MCR-E-0-07	None/ None	IE-FRI-FI8
9	FRI-FI10	Fire in MCR Bus 5 Switches Occurs	2.02E-04	Bus 5	FRI-MCR-E-0-07	None/ None	IE-FRI-FI10
10	FRI-FI11	Fire in MCR Bus 6 Switches Occurs	2.20E-04	Bus 6	FRI-MCR-E-0-07	None/ None	IE-FRI-FI11
12	FRI-FI13	Fire in Pressurizer PORV Switches	1.39E-04	Valve PR-2B and 1B are stuck open	SLOCA	*	IE-FRI-FI13

Notes:
* = New FTs: FAB-PR-2B-SO and BLEED-PR-2B-SO
** = New ET: FRI-MCR-E-0-07
*** = New ET: FRI-DSP

Scenarios 2, 7 and 11 are omitted from this table for presentation purposes.

2.2.2 Quantify Sequence CDFs

The CDF of each sequence can be calculated as a product of the scenario frequency and the CCDP given the scenario has occurred:

CDF = IEfreq * CCDP.

The scenario IEfreq is already calculated in the earlier step, using Appendices 2A-1 through 2A-5. The CCDP can be calculated by using the SAPHIRE code and the SPAR models, which already model plant response to many types of trips. For this purpose, either a change set or the Graphical Evaluation Module (GEM) software can be used to model the components failed due to fire. The scenario may cause multiple SSCs to fail, even redundant trains of a mitigating system.

New event and fault trees may need to be created, if the scenario does not lead to (i.e., transfer to) an already existing event tree (typically one for the existing internal events model). Figure 2-1 shows a new event tree model that is made for the example calculations.

After the CCDPs have been determined, the sequence CDFs can be calculated. Table 2-2 shows an example set of sequence CDF calculations. The overall CDF estimate is the sum of all sequence CDF estimates.

Once the CDF is known, it can be used to estimate event/condition importance.

2.3 Examples

This section discusses examples for illustrative purposes; the values used in the examples are for illustration only.

2.3.1 Example 1 - Event Analysis

A fire initiating event occurs in plant X. A 4160 VAC bus is damaged (any suppression attempt prior to damage would have to be assumed to have been unsuccessful). This is assumed to be the only equipment damaged by the fire. The reactor is manually tripped.

Use the existing Loss of a 4160 VAC bus initiating event model in SPAR, with an initiating event frequency of 1.0 (GEM or SAPHIRE can be used). Calculate event CCDP as

CCDP = 4.3E-04.

This is the fire initiating event importance, conditional to fire severity factor and non-suppression.

2.3.2 Example 2 - Plant Condition Analysis

In 480 volt switchgear room E7 (Fire Area DG-8), Division II (Train B) circuits in two conduits were routed closer than 20 feet from the redundant Division I (Train A) circuits in the designated separation zone without being protected by a one-hour fire rated barrier, as required. A fire in this area could damage the unprotected cables to components required to achieve and maintain safe shutdown.

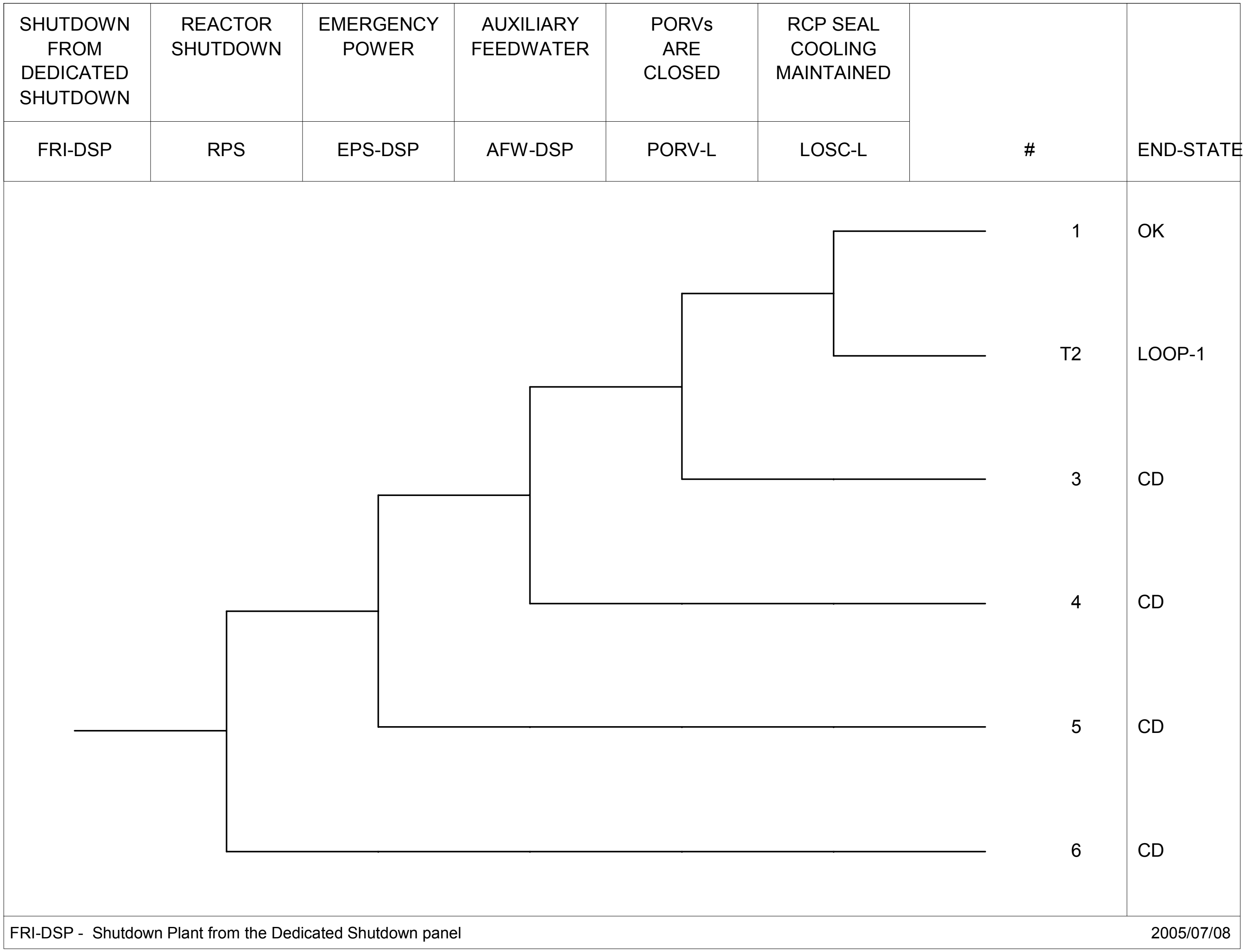


Figure 2-1 An Example New Event Tree Model

Table 2-2 An Example Calculation of Sequence CDFs

	Event	Description	Initiating Event Frequency	Type of Trip	CCDP	CDF
1	FRI-FI1	Auxiliary Building MCC 1-62J Room	2.63E-04	TRANS	5.96E-07	1.57E-10
2	FRI-FI2	MCC 62A Scenario	1.34E-03	TRANS	2.84E-03	3.81E-06
3	FRI-FI3	4.16KV SWGR Room 16 buses 1 and 2, beneath cable tray 1AT9N	9.50E-05	LOMFW	1.21E-05	1.15E-09
4	FRI-FI4	Diesel B Oil Fire	8.90E-03	FRI-MCR-E-0-07	2.28E-03	2.03E-05
5	FRI-FI5	Fire in Relay room	6.78E-07	FRI-DSP	1.86E-01	1.26E-07
6	FRI-FI6	Turbine Building AFW Pump A oil fire	6.45E-04	FRI-MCR-E-0-07	8.68E-02	5.60E-05
7	FRI-FI7	AFW Pump B Oil Fire	6.20E-05	FRI-DSP	1.86E-01	1.15E-05
8	FRI-FI8	Fire Near buses 51 and 52	4.65E-05	FRI-MCR-E-0-07	8.67E-02	4.03E-06
9	FRI-FI1	Fire in MCR Bus 5 Switches Occurs	2.02E-04	FRI-MCR-E-0-07	8.44E-02	1.71E-05
10	FRI-FI11	Fire in MCR Bus 6 Switches Occurs	2.20E-04	FRI-MCR-E-0-07	4.55E-02	1.00E-05
11	FRI-FI12	Fire in SG PORV Switches	1.76E-05	LOMFW	1.21E-05	2.13E-10
12	FRI-FI13	Fire in Pressurizer PORV Switches	1.39E-04	SLOCA	3.03E-04	4.21E-08
		SUM =	1.19E-02			**1.23E-04**

Define base and condition case fire scenarios as in Figures 2-2 and 2-3 (note that Ffi = 3.25E-3/yr and SF =1).

Use SAPHIRE and SPAR to calculate the CCDPs for plant trips with loss of either one or two 4160 buses as 4E-04 and 0.05, respectively.

The following probabilities are introduced in Figures 2-2 and 2-3 to calculate fire scenario frequencies:

DET: Detection. 0.05 failure probability (NUREG/CR-6850, Ref. 2-2)

SUP: Suppression. 0.05 failure probability (NUREG/CR-6850, Ref. 2-2)

If the event tree nodes DET and SUP are successful, only the affected component is assumed damaged (and the CCDP = 4.0E-4 applies). However, if either DET or SUP is unsuccessful, fire propagation from one division to the other is credible and is so modeled (i.e., the CCDP = 0.05 potentially applies), depending upon the conditional probability as calculated.

2ndTR: Fire engulfs second train (conditional upon unsuccessful detection or suppression). Effectively zero for the base case (0.00001 is assigned for this assumed configuration. Note that the potential for fire damage with no fixed suppression system may be high for rooms where the fire can produce a hot gas layer. For this particular example, assuming that separation alone protects this redundant train is conservative, since a low CDF for the base case increases the delta CDF due to the presence of transients in the exclusion zone).

For the condition, 0.01 is assigned, corresponding to 87 hours/year of presence of transient combustibles in the fire area and probability of 1.0 of fire propagating to the opposing division if the fire occurs while the transient combustibles are present. Without the presence of transient combustibles, the fire in one train affecting the second train is assumed to be not credible.

CCDP: Conditional core damage probability, given a fire scenario occurs.

For the base case fire scenario, with one 480V bus assumed unavailable, the CCDP is 4E-04 (GEM output).

For the condition with both 480V buses unavailable, CCDP is calculated to be 4.8E-2 (GEM output). 0.05 is used for calculations.

Base case CDF is calculated as shown in Figure 2-2 as 1.3E-06/yr, as a sum of five fire sequences defined by the same figure.

The condition CDF is calculated as shown in Figure 2-3 as 1.46E-06/yr.

The condition importance, defined as the difference between CDFs for the plant condition case and the base case, is calculated for a one-year exposure time as

Condition Importance = (1.46E-06 - 1.3E-06) * 1yr = 1.6E-07.

2.3.3 Example 3 - Plant Condition Analysis (Shortcut)

The example in Section 2.3.2 can also be treated in a shortcut manner as follows:

The scenario of concern is the failure of both trains due to fire engulfing the second train. The probability of fire propagating to second train is 0.01 (P2ndtr). The scenario frequency is:

IEfreq = Ffi * SF * Pns * P2ndTR

With SF = 1 and Pns = 0.1 (approximate Boolean sum for failure of detection or suppression), the scenario frequency is

IEfreq = 3.25E-03 * 1 * 0.1 * 0.01 = 3.25E-06 /yr.

CCDP with loss of two trains is 0.05. Thus the scenario CDF is

CDF = IEfreq * CCDP = 3.25E-06 * 0.05

CDF = 1.6E-07/yr

This result matches that of the example in Section 2.3.2.

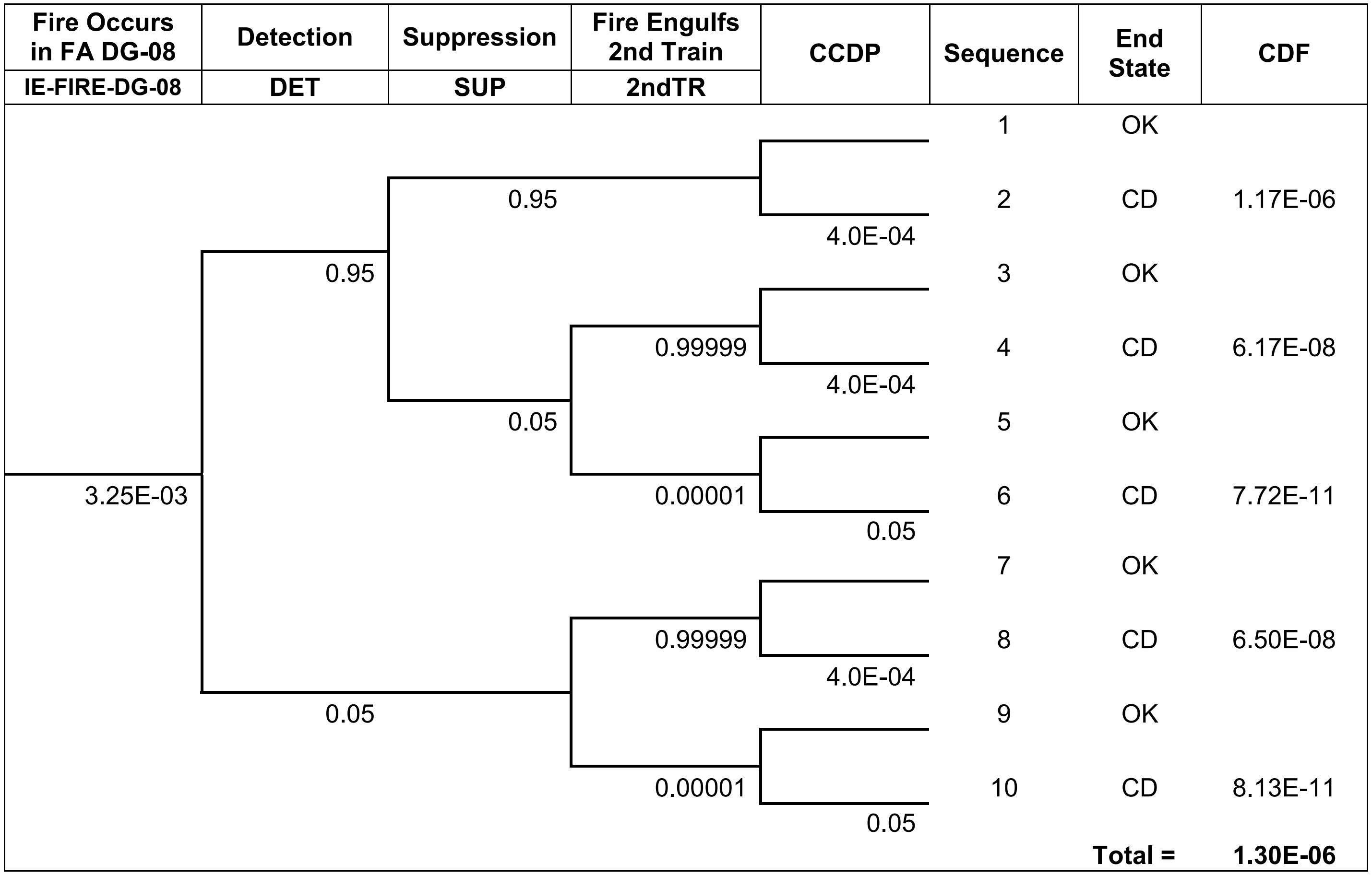


Figure 2-2 Fire in DG-08 Base Case (Example 2)

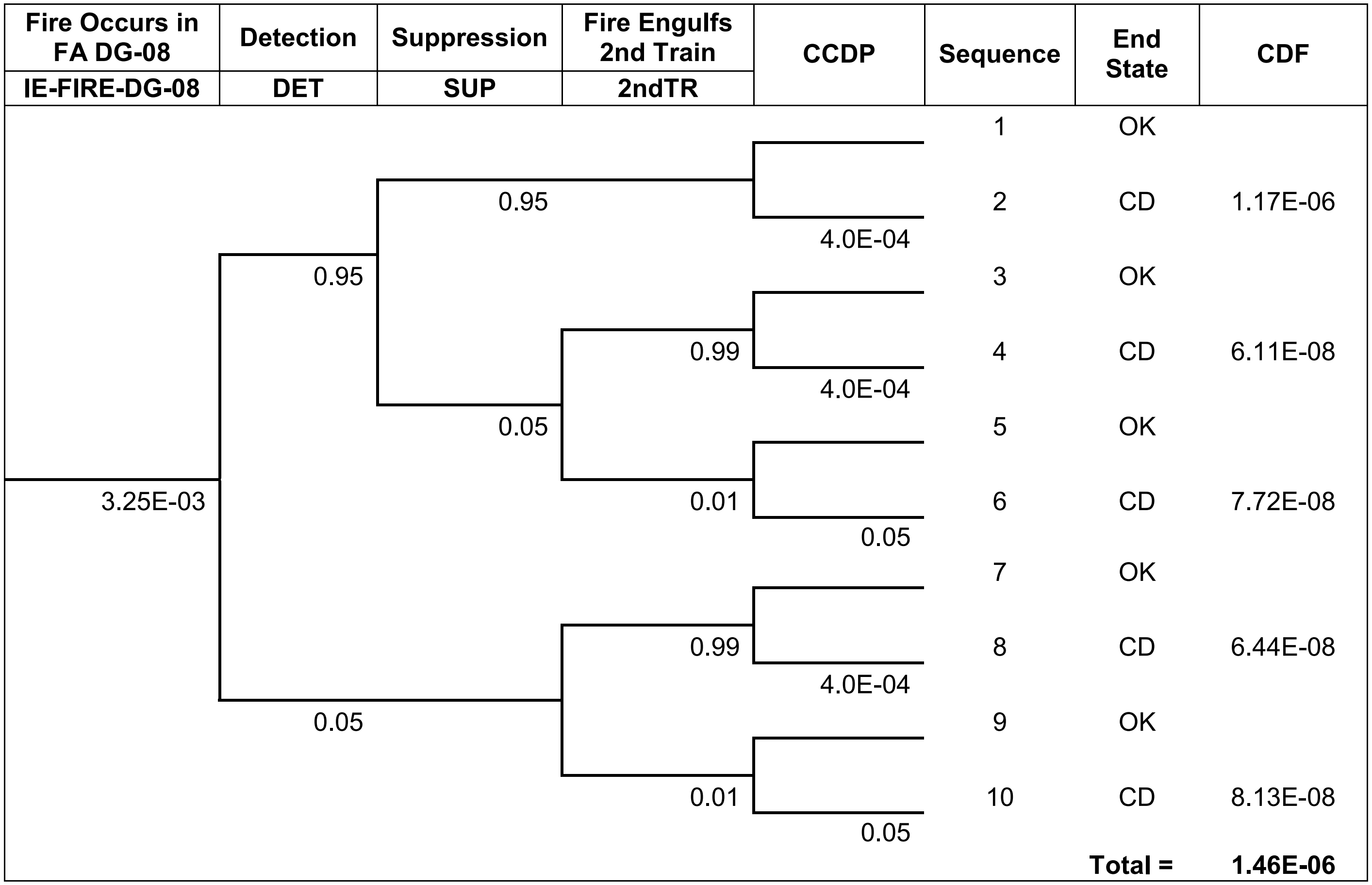


Figure 2-3 Fire in DG-08 with Plant Condition in Effect (Example 2)

2.3.4 Example 4 – Main Control Room (MCR) Fire

In the absence of more detailed MCR fire modeling, the following model with three scenarios can be used for MCR fire CDF estimation, with adjustment of the number of electrical cabinets for a specific plant.

The three MCR scenarios are:

FRI-MCR-NS = Fire in non-safety cabinets in MCR. Loss of all non-safety systems and a transient event is assumed.

FRI-MCR-S = Fire in safety cabinets in MCR. Loss of all trains of one of two safety-related equipment and transient is assumed.

FRI-MCR-EVAC = MCR evacuation with shutdown from remote shutdown panel.

For a MCR fire, with 103 electrical cabinets (each with a fire ignition frequency of 9.45E-5/yr/cabinet) in the MCR, the following limiting fire scenarios are modeled for a plant:

Scenario	Ignition Frequency	Ignition Frequency	Reactor Trip
Fire in non-safety electrical cabinets	73* 9.45E-05 [1]	6.9E-03	Transient without non-safety systems
Fire in safety-related electrical cabinets	30* 9.45E-05 [1]	2.83E-03	Transient without one safety train
MCR evacuation	[2]	[2]	Shutdown from remote shutdown panel

Notes:
[1] 73 non-safety- related and 30 safety-related cabinets.
[2] MCR evacuation analysis is complex. A specialist should be consulted for guidance.

An illustrative set of ignition frequencies, CCDPs, and CDFs for these scenarios is given below:

Scenario	Ignition Frequency	CCDP	CDF
FRI-MCR-NS	6.90E-03	1.77E-07	1.22E-09
FRI-MCR-S	2.83E-03	2.06E-03	5.82E-06
FRI-MCR-EVAC	[See Note]	[See Note]	[See Note]

Note: FRI-MCR-EVAC sequence may have a significant contribution to the total CDF. A specialist should be consulted for guidance to perform an MCR evacuation analysis.

2.3.5 Other Examples and References

Other examples can be found in NUREG/CR-6850 (Ref. 2-2) and the Handbook for Phase 3 Fire Detection SDP (Ref. 2-3). Ref. 2-3 also contains information about methods that can be used to perform Phase 3 SDP analysis of a sample of fire protection issues. Ref. 2-3 is a specific application of those methods detailed in NUREG/CR-6850.

2.4 References

2-1. U.S. Nuclear Regulatory Commission, “Risk Assessment of Operational Events Handbook: Volume 1 - Internal Events,” Revision 1, September 2007.

2-2. U.S. Nuclear Regulatory Commission, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology,” NUREG/CR-6850, September 2005. http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6850/

2-3. U.S. Nuclear Regulatory Commission, “Handbook for Phase 3 Fire Protection (FP) Significance Determination Process (SDP) Analysis,” December 2005. (ADAMS Accession Number ML053620267)

2-4. U.S. Nuclear Regulatory Commission, “Fire Events – Update of U.S. Operating Experience, 1986-1999,” RES/OERAB/S01-01, December 2001. http://nrcoe.inel.gov/results/index.cfm?fuseaction=Fire.showMenu

2-5 U.S. Nuclear Regulatory Commission, “Fire Protection Significance Determination Process,” Inspection Manual Chapter 0609, Appendix F, February 28, 2005 or current revision. http://www.nrc.gov/reading-rm/doc-collections/insp-manual/manual-chapter/index.html

Appendix 2A. Fire Scenarios/Accident Sequences

Fire scenarios may be defined either with respect to a location in the plant, or with respect to specific ignition sources in an area. Location-based scenario definition is easier to model and requires less detailed layout information, but would be more conservative. Ignition-source-based scenario definition would allow more realistic modeling but would require more information, resources, and expertise. The first method is favored in this handbook for first-cut modeling for an event analysis. The second method may require the assistance of a fire PRA analyst.

2A-1 Fire Sequences for a Single Fire Area – No propagation to another area (boundary intact)

When a fire ignition in a given fire area (or compartment) is postulated, at least the following need to be considered to define fire scenarios and calculate scenario frequencies:

- Fire ignition frequency
- Fire severity level
- Fire detection
- Fire suppression

Other special considerations, such as spurious actuations due to hot shorts and operator actions introduced by the scenario, can be added, as needed. These considerations are discussed in the next Appendices.

The above considerations can be quantitatively factored into the scenario logic to define one or more potential core damage sequences. An event tree model can be used to formally define sequences based on various developments following a fire. Figure 2A-1-1 depicts such an event tree, where potential core damage sequences SC-1 and SC-2 are defined. Such an event tree can be simply made by hand, using MS EXCEL, or using SAPHIRE software. Also see the example in Section 2.3.3 where a shortcut is used in lieu of developing an event tree.

- ***Summary of fire scenarios.*** To each core damage sequence, attributes can be assigned such as

    - Fire ignition frequency
    - Damaged equipment
    - Type of plant trip (initiating event) caused by the scenario
    - Effect of scenario on existing operator actions, success criteria, etc.
    - New operator actions introduced by the scenario, etc.

    Each sequence frequency should be calculated and a summary should be generated, as shown in Table 2-1. This information is then used to calculate the CCDP by using the SPAR model and the SAPHIRE software.

    Appendices 2B through 2E provide some data for the various event tree nodes that can be considered in fire sequence definition. These appendices are

    - 2B – Generic Fire Ignition Frequencies
    - 2C – Severity Factors Data
    - 2D – Detection Failure Data

- 2E – Suppression Failure Data
- 2F – Spurious Actuations (due to hot shorts) Probabilities
- 2G – Operator Actions
- 2H – Smoke Damage

- ***Fire ignition frequency.*** For fire ignition frequency, two methods are available and may be used: component-based ignition frequencies or plant area-based ignition frequencies. Details are provided in Appendix 2B.

- ***Damaged equipment.*** In defining the effect of the fire on the equipment in the area, as a first approximation, all PRA relevant equipment in the area may be assumed damaged by the fire. If the sequence CDF becomes unduly conservative, further fire growth/development modeling, PRA analysis, and walkdowns to credit the actual layout and combustible materials may be needed.

2A-2 Multiple Fire Areas – Propagation to adjacent area possible (boundary compromised)

In some fire scenarios, the fire area X boundary may be compromised and the possibility of a fire initiating in X propagating into an adjacent area Y (also a fire originating in Y propagating into X) may arise. Such fire scenarios can be modeled in various ways; one way based on expanding the formal logic of Figure 2A-1-1 is shown in Figure 2A-2-1. The reverse propagation from Y into X must also be modeled in a similar manner.

- ***Example 1.*** From the example depicted in Figure 2A-2-1, the top events in the event tree and branch point probabilities may be defined as follows:

  A. *Fire Occurs in Fire Area X.* This is the fire ignition frequency from Appendix 2B.

  B. *Severity Level.* From Appendix 2C, a value of 1 was chosen for this example.

  C. *Fire Detected in Fire Area X.* The automatic fire detection system in this example meets all applicable codes, and are designed and installed for the hazard – thus they are effective. From Appendix 2D, the unavailability of an automatic detection system in this example is 0.05.

  D. *Fire Suppressed in Fire Area X.* The sprinkler suppression system in this example meets all applicable codes, and are designed and installed for the hazard – thus they are effective. From Appendix 2E, the unavailability of the sprinkler system in this example is 0.05.

  E. *Fire Propagates into Fire Area Y.* For this example, the combustible loading is high for fire area X, and is capable of failing the fire barrier between fire areas X and Y. This particular 3-hour fire rated barrier is degraded.

     For the sequences where automatic detection system fails randomly (0.05) in the area of fire origin, the fire brigade response in the example is delayed and the barrier to the adjoining area fails, despite the fact that the brigade performs remedial efforts to prevent barrier failure after arrival. For this particular barrier and set of combustibles, failure of the fire brigade to suppress the fire prior to the barrier failure is assumed to be 0.5 (for illustrative purposes).

Success of manual suppression is likely to be greater for the case where detection is not delayed in Fire Area X ; however, for illustrative purposes the same failure probability (0.5) is assumed.

Guidance for estimating the failure probability of manual suppression based on available time is provided in Appendix 2E.

F. *Fire Detected in Fire Area Y.* For sequences where the fire propagates into Fire Area Y due to failure to manually suppress the fire before fire barrier breach, two branch point paths are provided for this top event. The first branch point (F1) assumes higher success of detection from the combination of automatic and manual detection. Due to plant practice to check neighboring areas upon such a fire, it is very likely that the spread of this fire into the adjoining area will be detected. In this example, manual detection is assumed to be likely due to early detection of the fire in Fire Area X (i.e., successful detection). The failure probability (0.05 x 0.1 = 0.005 ~ 0.01) is the product of random unavailability of the automatic fire detection system for Fire Area Y (0.05) and failure to manually detect the fire (assumed to be a probability of 0.1 in this example).

The second branch point (F2) conservatively assumes no credit for manual fire detection given failure of the automatic fire detection in Fire Area X.

Guidance for estimating the failure probability of manual detection based on available time is provided in Appendix 2D.

G. *Fire Suppressed in Fire Area Y.* For the sequences where detection in Fire Area Y succeeds, two branch point paths are provided for this top event. The first branch point (G1) assumes higher success of suppression due to the combination of the fixed suppression system and manual suppression.

Further, the overall top event assumes that the successful actuation of the detection and fixed sprinkler systems, with or without manual suppression, is likely to be adequate to control the fire after it has breached the barrier and prior to damage of the redundant train in Fire Area Y. This assumption implies that the equipment in Fire Area Y is adequately separated from the failed barrier. In this example, the fixed sprinkler system and/or manual suppression in Fire Area Y was determined to be effective in preventing fire to the redundant train since some separation exists between the failed barrier and raceway containing that train. This assumption may not be applicable for cases where cables related to trains contained in raceways border the fire barrier, even in the presence of a sprinkler system up to code. A fire protection specialist should be consulted to determine the effectiveness of a fire suppression system with degraded fire barriers.

The second branch point (G2) assumes no credit for manual suppression given failure of the automatic fire detection in Fire Area X.

Guidance for estimating the failure probability of manual suppression based on available time is provided in Appendix 2E. As indicated earlier, the analyst is responsible for determining that probabilities associated with the scenarios are appropriate for the analysis.

- ***Example 2.*** Figure 2-2 can be interpreted as another instance of fire propagation from one fire compartment to another if the two redundant trains are in separate fire compartments (not necessarily reflected in the original development of this scenario-sequence). In such a case, the distance between the two fire compartments (where a physical boundary other than distance is not present between the compartments) is observed to be shorter than the design condition of 20 ft with no intervening combustibles, allowing potential fire propagation from train A to train B electrical buses of a redundant safety system.

  Generic fire barrier failure probabilities by barrier type from NUREG/CR-6850 (Ref. 2-2, Table 11-3) are given in Table 2A-2-1. Note that these probabilities do not represent the failure of a barrier given a challenge by a particular fire hazard.

Table 2A-2-1 Random Barrier Failure Probabilities from NUREG/CR-6850

Barrier Type	Barrier Failure Probability/Demand
Fire, security, and water tight doors	7.4E-03
Fire and ventilation dampers	2.7E-03
Penetration seals, fire walls	1.2E-03

2A-3 Spurious Actuation (hot-shorts)

Spurious actuation of components due to hot shorts due to fire in or between cable trays and conduits is a concern that is given attention in fire PRAs. An accurate treatment of such concerns in a given scenario requires intimate knowledge of the cable types, specific cable tray/conduit layouts, their relative locations to ignition sources, and the relative locations of multiple trays. Appendix 2F provides spurious actuation probabilities for various characteristics of cables.

In general, estimation of spurious actuation probabilities must be left to fire PRA experts and should include detailed fire modeling and walkdowns. In some cases, bounding or simple estimates may be useful to assess the risk. An actual example of scenario definitions which included potential spurious actuation concerns for three types of failures is shown in Figure 2A-3-1. The specific concerns were:

- Spurious opening of pressurizer power-operated relief valves (PORVs) causes small LOCA,
- Spurious opening of one or more valves transfers inventory from the refueling water storage tank (RWST) to sump,
- Spurious closure of intake valves can fail charging/safety injection pumps, and component cooling water leading to potential reactor coolant pump seal failure (small LOCA).

These concerns were modeled by a bounding analysis; by assigning 0.30 probability of failure to a specific set of hot-short failures (see Appendix 2F). If two such set of failures occurred, then the third set is assigned a probability of 1.0 for hot-shorts. A detailed modeling would reduce these probabilities and also take into account the various fire ignition sources that would only challenge certain cable trays.

2A-4 Main Control Room Fires

Fires that require evacuation of the MCR need to be modeled using a custom made event tree capturing the plant-specific procedures and equipment available for this case. Figure 2-1 shows such a custom-made event tree for plant shutdown from the dedicated shutdown panel (remote shutdown panel). The equipment that are available on this panel for shutdown are usually more limited than those available in the MCR. This needs to be reflected in the fault tree models supporting this event tree. Crediting of local recovery actions by operators (such as local valve manipulation) must be done judiciously to avoid non-conservative modeling.

In these scenarios, the CCDP, given fire occurs, tends to be dominated by human error, rather than equipment failure.

See Example 2.3.4 for a limiting set of MCR fire scenarios that capture the essence of MCR fire scenario concerns.

2A-5 Containment Fires

Containment fire scenarios have been generally considered as low contributors to plant risk due to their low frequencies. However, if such a scenario needs to be modeled to study a specific plant condition or event, modeling may pose at least two difficulties:

- Assigning the proper ignition frequency to the model;
- Since containment generally does not have formally defined fire areas and can be loosely viewed as a single fire area, it may be difficult to limit the fire scenario to a compartment of the containment. Establishing a basis for limiting the fire targets to a compartment of the containment may require detailed fire analysis and knowledge of layout details.

If the event/condition involves one of the following two issues, containment fire modeling may be further pursued with a qualitative or a quantitative assessment; otherwise it may be screened out:

- There are more combustible materials allowed by the design in a part of the containment;
- Ignition sources are present in a close proximity of a cable or equipment configuration that can render inoperable multiple redundant safety-related trains of equipment.

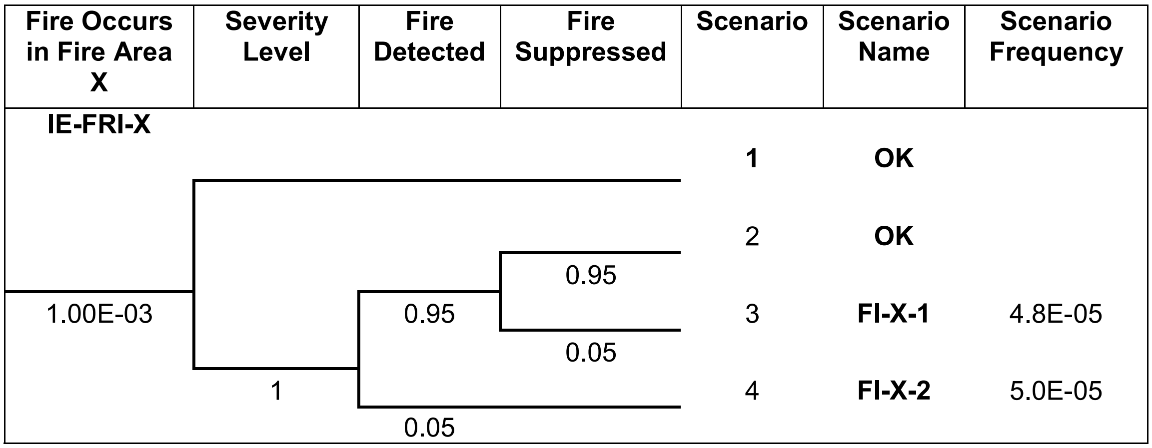


Figure 2A-1-1 Example Event Tree Model Showing Fire Scenario Definitions

Note:
In scenario #2, the fire, although detected and suppressed, may still manage to have damaged some equipment which contributes to core damage prior to being suppressed. In that case, this scenario can also be added to the list of fire scenarios for which CDF is to be quantified.

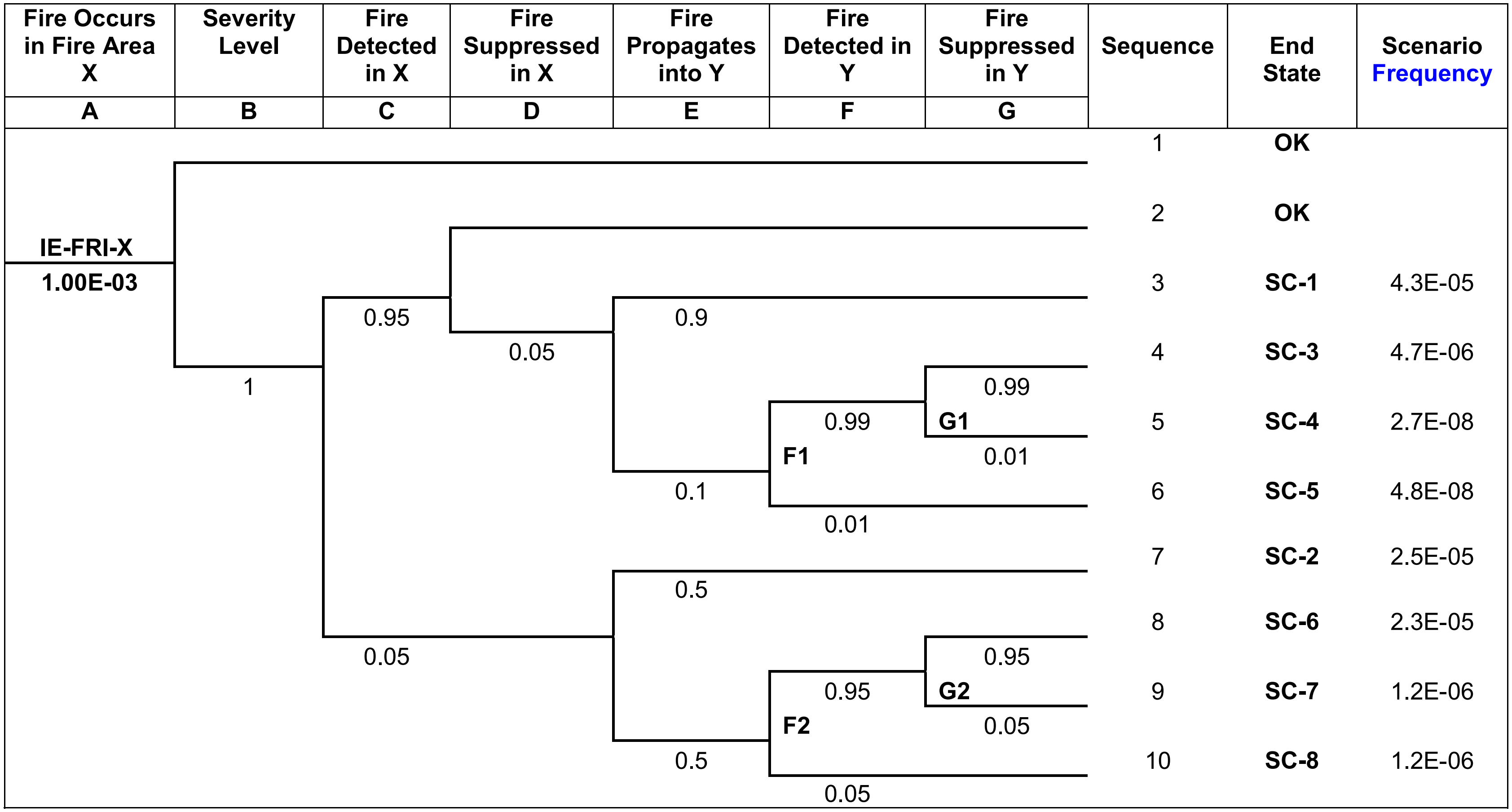


Figure 2A-2-1 An Example Event Tree Model with Possible Propagation

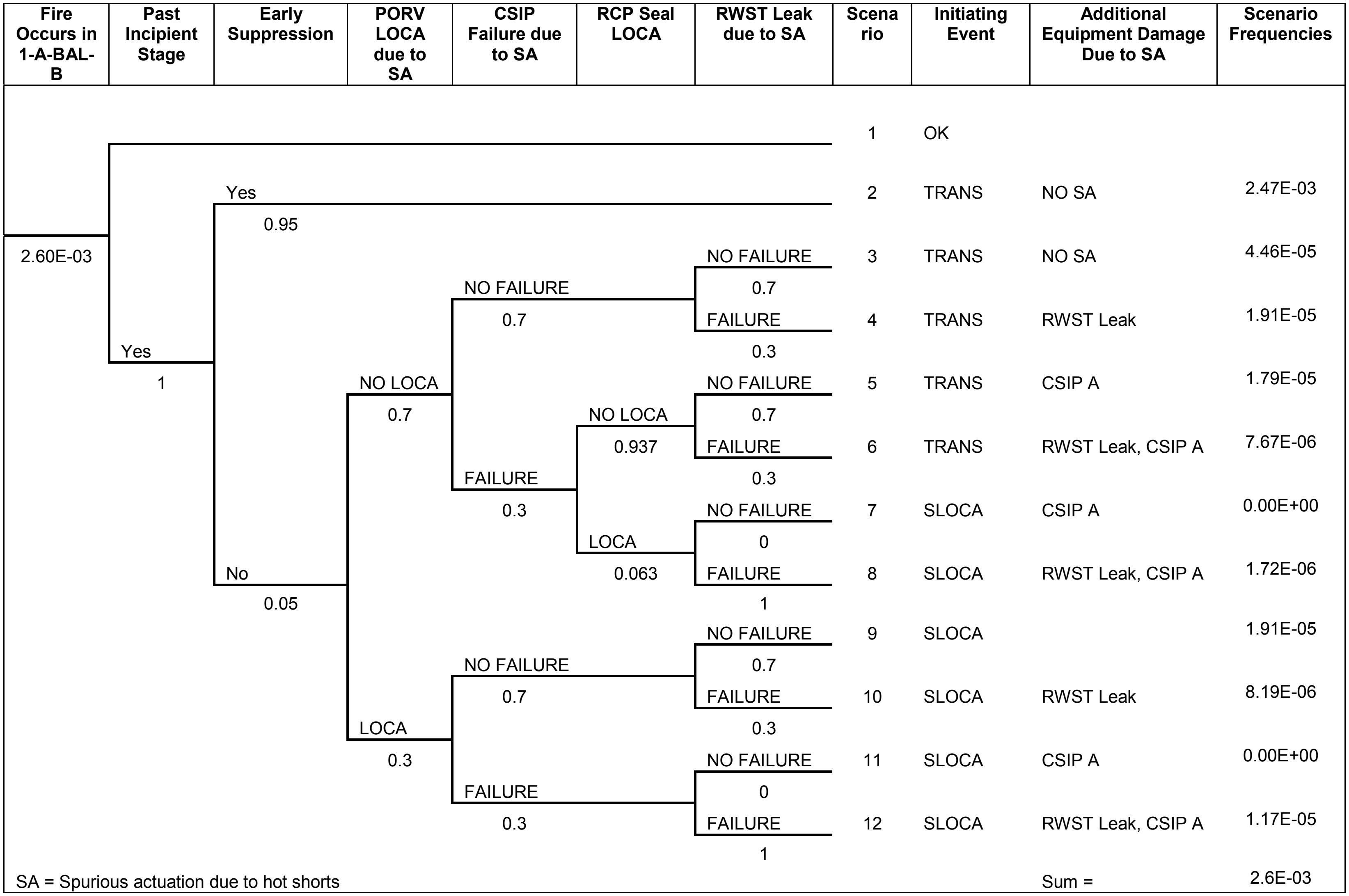


Figure 2A-3-1 An Example Event Tree Model with Possible Spurious Actuations Due to Hot Shorts

Appendix 2B. Generic Fire Ignition Frequencies

For fire ignition frequency, two methods are available: component-based ignition frequencies or plant area-based ignition frequencies.

- ***Component-based ignition frequencies.*** Assemble a fire ignition frequency from plant-wide components in the fire area, based on the information presented in NUREG/CR-6850 (Ref. 2-2), Table 6-1 or Table C-3. However, this can be done only if the number of components in the plant for the plant-wide components are already known or can be reliably estimated; otherwise, a determination of this data may be resource intensive.

  A reduced version of this table is given as Table 2B-1.

- ***Plant area-based ignition frequencies.*** Use generic fire area frequencies as provided in Table 2B-2. This method is useful for screening purposes, and if area fire ignition source details are not readily available. The fire area frequencies in Table 2B-2 are based on the information presented in the NRC study: “Fire Events – Update of U.S. Operating Experience, 1986-1999,” (Ref. 2-4).

Component-based frequencies should be used in the evaluation of fire protection structures, systems, and components (e.g., fire suppression system, fire-related barrier, smoke detection system). For these issues, which affect the risk from fire primarily, key insights from fire scenarios based upon components are important to understand and communicate the risk significance.

Table 2B-1 Fire Frequency Bins and Generic Frequencies from NUREG/CR-6850, Table 6-1 (Ref. 2-2)

ID	Location	Ignition Source (Equipment Type)	Mode	Generic Frequency (per reactor-yr)
I	Battery Room	Batteries	All	7.50E-04
2	Containment (PWR)	Reactor Coolant Pump	Power	6.10E-03
3	Containment (PWR)	Transient Combustibles and Hotwork	Power	2.00E-03
4	Control Room	Main Control Board	All	2.50E-03
5	Control/Aux/Reactor Building	Cable fires caused by welding and cutting	Power	1.60E-03
6	Control/Aux/Reactor Building	Transient fires caused by welding and cutting	Power	9.70E-03
7	Control/Aux/Reactor Building	Transient Combustibles	Power	3.90E-03
8	Diesel Generator Room	Diesel Generators	All	2.10E-02
9	Plant-Wide Components	Air Compressors	All	2.40E-03
10	Plant-Wide Components	Battery Chargers	All	1.80E-03
11	Plant-Wide Components	Cable fires caused by welding and cutting	Power	2.00E-03
12	Plant-Wide Components	Cable Run (Self-ignited cable fires)	All	4.40E-03
13	Plant-Wide Components	Dryers	All	2.60E-03
14	Plant-Wide Components	Electric Motors	All	4.60E-03
15	Plant-Wide Components	Electrical Cabinets	All	4.50E-02
16	Plant-Wide Components	High Energy Arcing Faults	All	1.50E-03
17	Plant-Wide Components	Hydrogen Tanks	All	1.70E-03
18	Plant-Wide Components	Junction Boxes	All	1.90E-03
19	Plant-Wide Components	Misc. Hydrogen Fires	All	2.50E-03
20	Plant-Wide Components	Off-gas/H2 Recombiner (BWR)	Power	4.40E-02
21	Plant-Wide Components	Pumps	All	2.10E-02
22	Plant-Wide Components	RPS MG Sets	Power	1.60E-03
23a	Plant-Wide Components	Transformers (Oil filled)	All	9.90E-03
23b	Plant-Wide Components	Transformers (Dry)	All	9.90E-03
24	Plant-Wide Components	Transient fires caused by welding and cutting	Power	4.90E-03
25	Plant-Wide Components	Transient Combustibles	Power	9.90E-03
26	Plant-Wide Components	Ventilation Subsystems	All	7.40E-03
27	Transformer Yard	Transformer -Catastrophic 2	Power	6.00E-03
28	Transformer Yard	Transformer -Non Catastrophic	Power	1.20E-02
29	Transformer Yard	Yard transformers (Others)	Power	2.20E-03
30	Turbine Building	Boiler	All	1.10E-03
31	Turbine Building	Cable fires caused by welding and cutting	Power	1.60E-03
32	Turbine Building	Main Feedwater Pumps	Power	1.30E-02
33	Turbine Building	Turbine Generator Excitor	Power	3.90E-03
34	Turbine Building	Turbine Generator Hydrogen	Power	6.50E-03
35	Turbine Building	Turbine Generator Oil	Power	9.50E-03
36	Turbine Building	Transient fires caused by welding and cutting	Power	8.20E-03
37	Turbine Building	Transient Combustibles	Power	8.50E-03

Notes (Refer to NUREG/CR-6850):
1. See Appendix M for a description of high-energy arcing fault (HEAF) fires.
2. See Section 6.5.6 .
3. The event should be considered either as an electrical or oil fire, whichever yields the worst consequences.

Table 2B-2 Fire Ignition Frequencies for Power Operation by Plant Location from NRC Fire Study 1986-1999 (Ref. 2-4)

Plant Location	No. of Fires (Note 3)	No. of Reactor Critical Years	Ignition Frequency (Mean)
Auxiliary Building (PWR)	10.07	398.0	2.7E-02
Battery Room	0	596.5	8.4E-04
Cable Spreading Room	0	596.5	8.4E-04
Containment	1.26	596.5	3.0E-03
Control Room	3.78	596.5	7.2E-03
Diesel Generator Building	7.56	596.5	1.4E-02
Reactor Building (BWR)	5.04	198.5	2.8E-02
Service Water Pump-house	3.78	596.5	7.2E-03
Switchgear Room	2.52	596.5	5.1E-03
Switch Yard	10.07	596.5	1.8E-02
Turbine Building	23.93	596.5	4.1E-02

Notes: The following explanations apply only if Table 2B-2 is used:

1. Only, "severe" fires are considered with duration greater than five minutes and not self extinguished. These fire area frequencies should only be used in analyses of temporary conditions when fire contributes to the risk from other hazard groups, e.g. internal events. As such, these fire area frequencies should not be used to evaluate findings from degraded fire protection structures, systems, and components (e.g., fire suppression system, fire-related barrier, smoke detection system). An all encompassing fire, in the location of interest, should accompany the use of these fire area frequencies
2. For a severe fire in switchgear, switch yard electrical transformers, diesel generators, and cables/cable trays, the initiating fire frequency is developed from the number of power operation fires in the plant location (i.e., Switchgear Room, Switch Yard, Cable Spreading Room, Diesel Generator Building, etc.) based on the NRC proprietary fire event database with updated fire event data through 1999. Table 2B-2 provides "severe" fire frequencies for most plant location areas, from the updated fire event database.
3. The distribution of the NEIL fire events in the 68 plants were extrapolated to include the 41 plants that did not report to NEIL. Refer to Section A-1.2 in the fire study (Ref. 2-4) for details of the extrapolation.
4. A Jeffrey's prior (0.5 failures) is added to the number of severe fire events occurring during the 1986-1999 period and then divided by the number of power operation reactor years for the 1986–1999 period. For multiple rooms/fire zones within a plant location, the denominator is increased proportionately. For durations less than one year the frequency will be multiplied by the fractional year.

Example : Potential fire in Switchgear Room B (two Switchgear Rooms, A and B):

Switchgear Room Fire Frequency (F_i) = (2.51 + 0.5 power operation fires) ÷ (596.5 power operation reactor-years) x 1 year (duration) = 5.0 x 10^{-3}

Switchgear Room B Fire Frequency (F_{iB}) = F_i ÷ 2 = 2.5 x 10^{-3}.

Note: A "severity factor" has been directly included in the fire frequency by limiting fires to those greater than five minutes and were not self-extinguished. (This, too, must be consistent with Ref. 2-2.)

Appendix 2C. Severity Factors Data

Be cautious in assigning severity factors other than 1, unless one is already calculated for a scenario. Otherwise, inadvertent double-counting with ignition frequency assumptions is possible (non-conservative).

See Table 11-1 of NUREG/CR-6850 (Ref. 2-2) for recommended types of severity factors for ignition sources and locations. Then see Appendix E Tables E-2 through E-9 of NUREG/CR-6850 (Ref. 2-2) for severity factor values for different ignition sources. If severity credit is needed, seek expert help.

Note: one cannot mix the fire severity factors developed in of NUREG/CR-6850 (Ref. 2-2) with a fire ignition database that is developed from a different reference unless the same assumptions were consistently employed. This needs to be checked.

Appendix 2D. Detection Failure Data

Generic probability of failure of auto detection = 0.05

Source: NUREG/CR-6850, Appendix P (Ref. 2-2)

See also Figure P-4 of NUREG/CR-6850 (Ref. 2-2) for a complicated calculation of detection-suppression by using an event tree model and crediting prompt /automatic /manual detection and suppression means.

Appendix 2E. Suppression Failure Data

- ***Fixed suppression systems***. Unreliability values for fixed suppression systems from NUREG/CR-6850 are given in Table 2E-1, below.

Table 2E-1 Generic Failure Probabilities of Suppression Systems from NUREG/CR-6850, Appendix P (Ref. 2-2)

Fixed Suppression System	Unavailability
Carbon dioxide	0.04
Halon system	0.05
Wet pipe sprinkler systems	0.02
Deluge or preaction sprinkler systems	0.05

- ***Manual suppression (fire brigade).*** The manual suppression failure probability, Pms, can be calculated using the following equation:

  Pms = EXP(-LAMDA * delta T), where

Delta T (minutes) = (Time to target damage) - (Response time of the brigade) – (Time to detection)

Appendix P of NUREG/CR-6850 (Ref. 2-2) contains suppression probability curves as a function of time for various types of fires. Table P-2 contains a summary of all available curves. This table is given in Table 2E-2 in reduced form.

Should an all consuming fire be postulated to fail all equipment in a fire area, a choice must be made for which suppression curve to use in the analysis. For fire areas which contain two fixed ignition sources, the more conservative suppression curve (Lambda) should be utilized. For fire areas which contain many ignition sources, the "all fires" suppression curve (LAMDA) should be utilized. The control room type of fire should be applied to evaluate the control room. Exceptions should be justified.

Table 2E-2 Manual Suppression Probability per Unit Time (Lambda) and Failure Probability at Delta Time (Minutes) from NUREG/CR-6850, Table P-2 (Ref. 2-2)

Type of Fire	Lambda (/minute)	ΔT 1 min	ΔT 5 min	ΔT 10 min	ΔT 15 min	ΔT 20 min	ΔT 25 min	ΔT 30 min	ΔT 45 min	ΔT 60 min
		Manual Suppression Failure Probability (Pms)								
T/G fires	0.03	0.970	0.861	0.741	0.638	0.549	0.472	0.407	0.259	0.165
Control room	0.33	0.719	0.192	0.037	0.007	0.001	0.000	0.000	0.000	0.000
PWR containment	0.13	0.878	0.522	0.273	0.142	0.074	0.039	0.020	0.003	0.000
Outdoor transformers	0.04	0.961	0.819	0.670	0.549	0.449	0.368	0.301	0.165	0.091
Flammable gas	0.03	0.970	0.861	0.741	0.638	0.549	0.472	0.407	0.259	0.165
Oil fires	0.09	0.914	0.638	0.407	0.259	0.165	0.105	0.067	0.017	0.005
Cable fires	0.36	0.698	0.165	0.027	0.005	0.001	0.000	0.000	0.000	0.000
Electrical fires	0.12	0.887	0.549	0.301	0.165	0.091	0.050	0.027	0.005	0.001
Welding fires	0.19	0.827	0.387	0.150	0.058	0.022	0.009	0.003	0.000	0.000
Transient fires	0.12	0.887	0.549	0.301	0.165	0.091	0.050	0.027	0.005	0.001
High energy arcing faults	0.04	0.961	0.819	0.670	0.549	0.449	0.368	0.301	0.165	0.091
All fires	0.08	0.923	0.670	0.449	0.301	0.202	0.135	0.091	0.027	0.008

Notes:
1. Minimum Pms = 0.001.
2. Pms = EXP (-Lambda * ΔT), where ΔT in minutes = (Time to target damage) - (Response time of the brigade) - (Time to detection)

Example: If 30 minutes is available from start of fire to target damage, the detection occurs in 3 minutes, and the fire brigade response time is 7 minutes based on fire drills, then

Delta T = 30 – 7 – 3 = 20 minutes.

Then, the probability of manual suppression failure before the target is damaged is 0.09.

Appendix 2F. Spurious Actuation (due to hot shorts) Probabilities

For probabilities of spurious actuations due to hot shorts, refer to Section 10 of NUREG/CR-6850 (Ref. 2-2). Tables 10-1 through 10-5 from of NUREG/CR-6850 are given below for convenience. See table notes following the last table.

Caution: If detailed circuit analysis calculations need to be done, seek expert help.

NUREG/CR-6850, Table 10-1 Failure Mode Probability Estimates Given Cable Damage Thermoset Cable with Control Power Transformer (CPT)

Raceway Type	Description of Hot Short	Best Estimate	High Confidence Range
Tray	M/C Intra-cable	0.30	0.10 – 0.50
	1/C Inter-cable	0.20	0.05 – 0.30
	M/C → 1/C Inter-cable	0.10	0.05 – 0.20
	M/C → M/C Inter-cable	0.01 – 0.05	
Conduit	M/C Intra-cable	0.075	0.025 – 0.125
	1/C Inter-cable	0.05	0.0125 – 0.075
	M/C → 1/C Inter-cable	0.025	0.0125 – 0.05
	M/C → M/C Inter-cable	0.005 – 0.01	

M/C: Multi-conductor cable
1/C: Single conductor cable
Intra-cable: An internally generated hot short. The source conductor is part of the cable of interest
Inter-cable: An externally generated hot short. The source conductor is from a separate cable.

NUREG/CR-6850, Table 10-2 Failure Mode Probability Estimates Given Cable Damage Thermoset Cable without CPT

Raceway Type	Description of Hot Short	Best Estimate	High Confidence Range
Tray	M/C Intra-cable	0.60	0.20 – 1.0
	1/C Inter-cable	0.40	0.1 – 0.60
	M/C → 1/C Inter-cable	0.20	0.1 – 0.40
	M/C → M/C Inter-cable	0.02 – 0.1	
Conduit	M/C Intra-cable	0.15	0.05 – 0.25
	1/C Inter-cable	0.1	0.025 – 0.15
	M/C → 1/C Inter-cable	0.05	0.025 – 0.1
	M/C → M/C Inter-cable	0.01 – 0.02	

M/C: Multi-conductor cable
1/C: Single conductor cable
Intra-cable: An internally generated hot short. The source conductor is part of the cable of interest
Inter-cable: An externally generated hot short. The source conductor is from a separate cable.

NUREG/CR-6850, Table 10-3 Failure Mode Probability Estimates Given Cable Damage Thermoplastic Cable with CPT

Raceway Type	Description of Hot Short	Best Estimate	High Confidence Range
Tray	M/C Intra-cable	0.30	0.10 – 0.50
	1/C Inter-cable	0.20	0.05 – 0.30
	M/C → 1/C Inter-cable	0.10	0.05 – 0.20
	M/C → M/C Inter-cable	0.01 – 0.05	
Conduit	M/C Intra-cable	0.075	0.025 – 0.125
	1/C Inter-cable	0.05	0.0125 – 0.075
	M/C → 1/C Inter-cable	0.025	0.0125 – 0.05
	M/C → M/C Inter-cable	0.005 – 0.01	

M/C: Multi-conductor cable
1/C: Single conductor cable
Intra-cable: An internally generated hot short. The source conductor is part of the cable of interest
Inter-cable: An externally generated hot short. The source conductor is from a separate cable.

NUREG/CR-6850, Table 10-4 Failure Mode Probability Estimates Given Cable Damage Thermoplastic Cable without CPT

Raceway Type	Description of Hot Short	Best Estimate	High Confidence Range
Tray	M/C Intra-cable	0.60	0.20 – 1.0
	1/C Inter-cable	0.40	0.1 – 0.60
	M/C → 1/C Inter-cable	0.20	0.1 – 0.40
	M/C → M/C Inter-cable	0.02 – 0.1	
Conduit	M/C Intra-cable	0.15	0.05 – 0.25
	1/C Inter-cable	0.1	0.025 – 0.15
	M/C → 1/C Inter-cable	0.05	0.025 – 0.1
	M/C → M/C Inter-cable	0.01 – 0.02	

M/C: Multi-conductor cable
1/C: Single conductor cable
Intra-cable: An internally generated hot short. The source conductor is part of the cable of interest
Inter-cable: An externally generated hot short. The source conductor is from a separate cable.

NUREG/CR-6850, Table 10-5 Failure Mode Probability Estimates Given Cable Damage Armored or Shielded Cable

Raceway Type	Description of Hot Short	Best Estimate	High Confidence Range
With CPT	M/C Intra-cable	0.075	0.02 – 0.15
Without CPT	M/C Intra-cable	0.15	0.04 – 0.30

M/C: Multi-conductor cable
1/C: Single conductor cable
Intra-cable: An internally generated hot short. The source conductor is part of the cable of interest
Inter-cable: An externally generated hot short. The source conductor is from a separate cable.

NUREG/CR-6850, Table 8-2 Screening Criteria to Assess the Ignition and Damage Potential of Electrical Cables

Cable Type	Radiant Heating Criteria	Temperature Criteria
Thermoplastic	6 kW/m2 (0.5 BTU/ft2s)	205°C (400°F)
Thermoset	11 kW/m2 (1.0 BTU/ft2s)	330°C (625°F)

Notes for Failure Mode Probability Estimate Tables.

1. Categorize the circuit of interest based on the configuration attributes collected in Step 1.

2. From the appropriate table (Table 10-1 to 10-5), select the probability estimates for the failure modes of concern.

3. If the cable failure mode can occur due to different cable interactions, the probability estimate is taken as the simple sum of both estimates. For example, if a particular thermoset cable failure mode can be induced either by an intra-cable shorting event (P = 0.30) or by an inter-cable shorting event (P = 0.03; mid-range of 0.01–0.05), the overall probability of that failure mode is estimated to be 0.33.

4. When more than one cable can cause the component failure mode of concern, and those cables are within the boundary of influence for the scenario under investigation, the probability estimates associated with all affected cables should be considered when deriving a failure estimate for the component. In general, the probabilities should be combined as an "Exclusive Or" function, as shown:

   PComponent failure = (PFailure Cable A) + (PFailure Cable B) - (PFailure Cable A) (PFailure Cable B)

Appendix 2G. Operator Actions

In calculating scenario frequency and sequence CCDP, the following considerations about operator actions must be taken into account:

1. The scenario may affect some mitigative or recovery operator actions that are defined in the base internal events PRA. An operator action may either become impossible to perform, or its human error probability may increase. Especially, local operator actions

(outside the main control room) already credited in the PRA need to be considered: such actions may require the operator to go to the fire area in question or go through the same area to perform the action in another area. The fire may prohibit the operator action in both cases. This would affect the CCDP calculation.

2. New recovery actions may be introduced in defining the sequence, for suppression, component recovery, etc. Some new operator actions may also be introduced in the system models, which would affect the CCDP calculations. Such new human error probabilities must be introduced only when there is supporting basis to do so.

   Manual suppression (fire brigade) is discussed in Appendix 2E.

Appendix 2H. Smoke Damage

Appendix T of NUREG/CR-6850 (Ref. 2-2) discusses the smoke damage due to a fire event. It concludes that the current state of knowledge can not support detailed quantitative assessment.

3.0 Internal Flood Modeling and Risk Quantification

3.1 Objectives and Scope

This document is intended to provide a concise and practical handbook to NRC risk analysts who routinely use the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software and the Standardized Plant Analysis Risk (SPAR) probabilistic risk assessment (PRA) models to quantify event and plant condition importances, and other ad-hoc risk analyses. It is a complementary document to the handbook cited in Ref. 3-1.

NRC risk analysts encounter many plant conditions and events reported by such means as inspection reports, licensee event reports (LERs), generic risk issues that lend themselves to PRA quantification and evaluation, every year. The need for quantification of the event / condition importance in terms of the two common risk measures of core damage frequency (CDF) and large early release frequency (LERF) arise in many of these cases.

This handbook provides NRC risk analysts practical guidance for modeling internal flooding scenarios and quantifying their CDF using SPAR models and SAPHIRE software.

The handbook assumes that:

1. The user has hands-on experience with the SAPHIRE code;

2. The user has performed and documented event/condition importance analysis or plant risk assessment cases for a period of at least three months (this is a suggested period, not a firm limit) under the supervision of an experienced (qualified) senior PRA analyst. The user is the primary author of documentation packages for such analyses which are reviewed and accepted by an NRC program.

The current scope is limited to internal flooding events during power operation and calculation of CDF only.

Mainstream PRA terms and abbreviations that are used in this document are not defined; the intended reader is assumed to be familiar with them.

Both internal flooding and internal fire events are also known as “area events”. They both share modeling characteristics such as:

i). they can fail multiple components in the same area;

ii). they can propagate from their immediate area to adjacent areas and can potentially cause additional failures, despite the existence of “formal barriers” (due to barrier failure or design deficiency).

3.2 Internal Flooding Scenario Definition and Quantification

A two-step process is discussed to model internal flooding (FLI) scenarios and quantify their CDFs:

1. Define flooding scenarios that lead to core damage. For this purpose, an event tree logic structure such as the one given in Figure 3A-1-1 may be used. Using such a modeling structure, calculate scenario frequencies. Definition of a flooding scenario is discussed in Section 3.2.1.

2. Quantify the CDF of these scenarios using a SPAR model and the SAPHIRE software. For this purpose, first the scenario conditional core damage probability (CCDP) is calculated. Then this CCDP is multiplied by the scenario frequency calculated in Step 1. From a single flooding source, multiple scenarios may be derived, leading to multiple flooding sequences whose CDFs need to be summed. Quantification of sequence CDF is discussed in Section 3.2.2.

3.2.1 Define Internal Flooding Scenarios

For the event (or plant condition) in question, one or more flooding scenarios must be defined. Depending upon the issue at hand, the following cases are envisioned and are included in the scope:

1. FLIs that can be terminated by operator action before critical flood height for equipment damage is reached.

2. FLIs that are not terminated early, but are limited to a single flood area.

3. FLIs that are not terminated early and can propagate to additional flood areas.

A systematic method to define FLI scenarios that fit into one of these cases, using simple event tree logic is given in Appendix 3A. After the plant response is incorporated to define a flooding sequence, those FLI sequence scenarios that can lead to core damage are selected, and their CDFs are quantified.

The flooding sequences defined can be summarized in terms of a matrix containing the minimum amount of information to be able to quantify the scenario frequency, the scenario CCDP, and thus the scenario CDF:

CDF = Scenario Frequency * CCDP.

Potential sources of flooding events may include failures in hydraulic components, such as piping, expansion joints, heat exchangers, valves, tanks, vessels, and flanges, as well as inadvertent fire water actuation by steam or fire, in the following systems:

- Fire water system
- Emergency service water (ESW)/component cooling water (CCW) system
- Circulating water/nuclear service water (NSW) system.

Steamline break events, which by themselves may not pose a flooding threat, can actuate fire protection sprinklers and cause consequential flooding.

Potential damage to electrical equipment, such as in emergency diesel generator (EDG) rooms, alternating current (AC) switchgear rooms, electrical cabinets in other locations, must be considered, since they may have high consequences.

Damage modes to be considered include:

- equipment submergence
- equipment spray

Potential loss of a system or a train due to the equipment break causing the flood must also be considered, in addition to the equipment damage caused by the consequences of the flood. An example may be a non-recoverable loss of service water (SW) due to pipe break.

Initiating event frequencies of pipe breaks and other equipment that can cause flooding can be calculated by using failure frequencies available in the literature. Example sets of such data are given in Tables 3A-2-1 and 3A-2-3. An example calculation is shown in Section 3.3.

Operator actions to diagnose and isolate/ terminate the flood can be introduced into a scenario as shown in Figure 3A-1-1. This requires determination of the time window available to the operators to implement such actions, before the critical flood height is reached and the subject equipment is failed.

Examine the event/condition characteristics and refer to Section 3A-1 to define scenarios that lead to core damage. Summarize those scenarios in terms of a table, such as Table 3A-1-1. The columns of this table are discussed below. Note that, each of these scenarios is treated as an initiating event and will be transferred to an event tree already modeled in the internal events SPAR model. In very special cases, a new event tree representing the plant response to the flooding may be constructed, if needed.

1. Scenario name (initiating event ID). This always starts with FLI and is used both for the event tree and the initiating event names.

2. Scenario description

3. Scenario frequency IEfreq (initiating event frequency). This is calculated using models such as the one discussed in Figure 3A-1-1.

4. Equipment lost. Equipment credited in the PRA that is lost due to flood is listed in this column. Include trains/system that caused the flood and is also lost.

5. Initiating event caused. This is the initiating event caused by the flood. In most cases, it is one of the internal initiating event categories already defined (such as loss of main feedwater (LOMFW), TRANS, loss of service water system (LOSWS), etc.).

6. Human error probabilities (HEPs) and other basic events affected. List the basic events and operator actions that are affected by the flood (failed, degraded). This is in addition to equipment listed in item 5 above.

7. New basic events (failures) introduced. List any new basic events introduced (such as scenario initiating event frequencies, operator actions to isolate flood, etc.) to model the scenarios.

Other columns may be introduced as needed.

3.2.2 Quantify Sequence CDFs

When plant response is modeled (e.g., by transferring to the appropriate event tree), a scenario sequence is defined. The CDF of each sequence can be calculated as a product of the scenario frequency and the CCDP given the scenario has occurred:

CDF = IEfreq * CCDP.

The scenario frequency IEfreq is already calculated in the earlier step. The CCDP can be calculated by using the SAPHIRE code and the SPAR models. For this purpose, either a change set or the GEM software can be used.

The scenario may cause multiple structures, systems, and components (SSCs) to fail, even redundant trains of a mitigating system.

New event and fault trees may need to be made, if the scenario does not lead to (transfer to) an already existing event tree (typically one for the existing internal events model).

Table 3-1 shows an example set of scenario CDF calculations.

Table 3-1 Example Internal Flooding Results by Scenario

	Event	Description	Initiating Event Frequency	Type of Trip	CCDP	CDF
1	FLI-FL1	Turbine Building Basement Flood - Winter Conditions	8.90E-05	IE-LOMFW	1.21E-05	1.08E-09
2	FLI--FL2	Turbine Building Basement Flood - Summer Conditions	1.10E-04	IE-LOMFW	1.21E-05	1.33E-09
3	FLI-FL3	Diesel Generator Room A SW Connection Failure Flood	5.00E-04	IE-TRANS	1.68E-05	8.42E-09
4	FLI-FL4	Diesel Generator Room B SW Connection Failure Flood	5.00E-04	IE-TRANS	6.57E-06	3.29E-09
5	FLI-FL5	Relay Room Potable Water Flood	1.50E-04	IE-TRANS	5.97E-07	8.95E-11
6	FLI-FL6	Control Rod Drive Equipment Room Service Water Flood	1.50E-04	IE-TRANS	5.97E-07	8.95E-11
		Sum =	**1.50E-03**			**1.43E-08**

Once the sequence CDF is known, it can be used to estimate event/condition importance.

3.3 Examples

This section discusses examples for illustrative purposes; the values used in the examples are for illustration only.

See Ref. 3-3 for additional discussion and examples.

3.3.1 Example Event Analysis

An internal flooding initiating event occurs in plant X due to a rupture in one SWS train. Main feedwater (MFW) is lost due to flooding. The ruptured SW train had to be isolated to terminate the flooding, leaving only one train of SWS support to frontline systems. The plant is automatically tripped due to loss of MFW. Propagation of flood into other areas is not a concern.

The failure of isolation of the flooding source is calculated to be 1.0E-02. If this failure occurs, the AFW pump supported by ruptured SWS train will fail.

The event importance can be calculated as:

EVENT-IMP = (1-0.01)*CCDP1 + 0.01*CCDP2,

where CCDP1 and CCDP2 are the conditional core damage probabilities with or without success of isolation, respectively.

If the isolation is successful, use the existing transient event tree model in SPAR, with an initiating event frequency of 1.0 (GEM or SAPHIRE can be used). Also fail the MFW system and the one train of SWS. Calculate event CCDP1 as

CCDP1 = 1.0E-04.

If the isolation fails, the same CCDP value is calculated, since AFW pump supported by the faulted SW train is not credited anyway in the first case. The faulted SW train is still ineffective and MFW is inoperable. Thus,

CCDP2 = 1.0E-04.

The event importance is 1.0E-04. (Even with the modification above, this will still be approximately correct since the non-isolated case will dominate.)

Also consider the following variation: if the isolation fails, the flooding will propagate into a switchgear area, rendering a 4160 VAC train inoperable (the bus supports the failed SW train), in addition to the already existing failures of the MFW and one SW train. In that case, the SPAR model gives a CCDP2 value of

CCDP2 = 1.0E-03.

Thus, with the variation, the event importance is calculated as:

EVENT-IMP = 0.99*1.0E-04 + 0.01*1.0E-03

EVENT-IMP = 1.1E-04. (Based on the above discussion, with CCDP1 reduced by at least a factor of 100, this value will be no greater than 1.1E-5.)

3.3.2 Example Condition Analysis

A plant inspection revealed that the flood barrier between flood areas X and Y was compromised for a period of 3 months, so that a large flood in area X can propagate to area Y and render both 4160 VAC emergency buses inoperable (Ppr). There are no flood sources in area Y. Large flood in area X will also render the MFW system in operable. Time window to critical height is so short that no credible operator action to isolate large flood sources exists (HEPiso). The total initiating event frequency from different potential large flood sources in area X is calculated to be 5E-03/year.

Both the plant base and condition cases must be evaluated to calculate condition importance.

For the base case, a transient with loss of MFW is modeled and the CCDP-base is calculated as 1.0E-06 using SPAR.

For the condition case, the CCDP-cond calculated by using the SPAR model with TRANS event tree without MFW, both emergency 4160 AC buses failed, and potential RCP seal LOCA is 0.2. The exposure time to this plant condition is 0.25 years. Thus the plant condition importance is calculated as:

COND-IMP = exposure time * initiating event frequency * (CCDP-cond - CCDP-base)

COND-IMP = 0.25 * 5.0E-03 * (0.2 – 1.0E-06)

COND-IMP = 2.5E-04.

Note that

HEPiso = 1.0
Ppr = 1.0

in this example. Thus, the scenario frequency is equal to flood initiating event frequency and there is only one scenario generated from the flood initiating event.

3.3.3 Example Initiating Event Frequency Calculation

This example calculation is for the initiating event frequency of large flooding (IE-FLI-X) from the circulating water system inlet lines in a pressurized water reactor (PWR).

Three failure modes are considered:

1. failure of the expansion joints (F1)
2. rupture of the piping and components in the system (F2)
3. maintenance errors (F3);

IE-FLI-X = F1 + F2 + F3.

The expansion joints would not be subject to water hammer because they are located downstream of the isolation valves and the joints are not connected to a common header after the isolation valves until the lines combine in the circulating water discharge tunnel, well past the expansion joints. Expansion joint failures are typically caused by either misapplication of the expansion joint for the intended service or poor installation. The physical condition of the expansion joints has been evaluated by the vendor and the condition of the expansion joints found acceptable for the life of the plant with no expected deterioration in performance. With four inlet expansion joints, the total frequency of expansion joint failures is calculated to be:

F1 = 4.5E-5 * 4 = 1.80E-04 per year,

Where the expansion joint failure is taken from Table 3A-2-1.

Circulating water inlet piping contains ten pipe segments and four valves. Therefore, the frequency of large (D >= 6”) circulating water inlet-initiated pipe rupture events was calculated to be:

F2 = F(piping) + F (valves)

F2 = 8760 hours/year *((10 pipe segments) * (1.39E-10 / pipe segment-hour) + (4 valves) * (4.0E-10 / valve-hour)) * 0.5

F(piping) = 1.31E-05 per year,

where data is taken from Tables 3A-2-3 (Generic PWR Pipe Rupture in “Other Safety-Related Systems” for D >= 6’), 3A-2-1 (Valve non-PCS Rupture) and 3A-2-2 (0.5 for large failure given a break in large piping (D >= 6’)).

Flooding events initiated by maintenance on the circulating water system are considered negligible contributors to the overall initiating event frequency (assume an upper bound F3 for completeness):

F3 = 1.0E-06/yr.

Thus, the total frequency of large breaks in the circulating system inlet piping is

IE-FLI-X = 1.8E-04 + 1.31E-05 + 1.0E-06
IE-FLI-X = 1.9E-04 /year.

3.4 References

3-1. U.S. Nuclear Regulatory Commission, "Risk Assessment of Operational Events Handbook: Volume 1 - Internal Events," Revision 1.01. December 2007.

3-2. Electric Power Research Institute, "Pipe Failure Study Update," EPRI TR-102266, April 1993.

3-3. Idaho National Laboratory, "A Feasibility and Demonstration Study – Incorporating External Events into SPAR Models," February 2005.

3-4. Reserved.

3-5. Idaho National Engineering Laboratory, Component External Leakage and Rupture Frequency Estimates," EGG-SSRE-9639, November 1991.

3-6. U.S. Nuclear Regulatory Commission, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995," NUREG/CR-5750, February 1999.

Appendix 3A. Model and Data for Internal Flooding

3A-1 Scenario Definition

An event tree model that defines a set of generic internal flooding scenario sequences is illustrated in Figure 3A-1-1. The end states are transferred to existing event trees (already made for internal events), with additional equipment damage due to the scenario. The event tree model considers at least the following aspects of an FLI scenario:

1. Definition of the FLI source in flood area X, its flow rate, critical flood height for equipment damage, and time window for reaching the critical height. The frequency of the initiating event is also calculated.

2. Credible detection/isolation by operators to terminate IF to either prevent equipment damage or limit the extent of equipment damage.

3. Potential for propagation from flood area X to another flood area Y due to barrier failure or design deficiency.

Additional event tree nodes to better define scenario-specific issues can also be introduced into the event tree to better define FLI scenarios.

The frequency IEfreq of a limiting FLI scenario can be defined as

IEfreq = Fif * HEPiso * Ppr, where

Fif	=	FLI frequency
HEPiso	=	Failure to terminate the flood source
Ppr =		Probability of propagation to another flood area.

Other scenario-specific factors can be introduced to the above equation, as warranted.

An example of such a matrix for multiple FLI scenarios is given in Table 3A-1-1. This matrix must contain enough information for a PRA analyst to calculate the scenario CCDPs, using existing event trees in the internal events PRA. Very special scenarios may require construction of new custom-made event/fault trees to address a specific issue.

Table 3A-1-2 shows another table where the scenario information is tabulated for CCDP calculation.

3A-2 Initiating Event Frequency Data

Table 3A-2-1 provides pipe and other equipment rupture frequencies assembled from different sources.

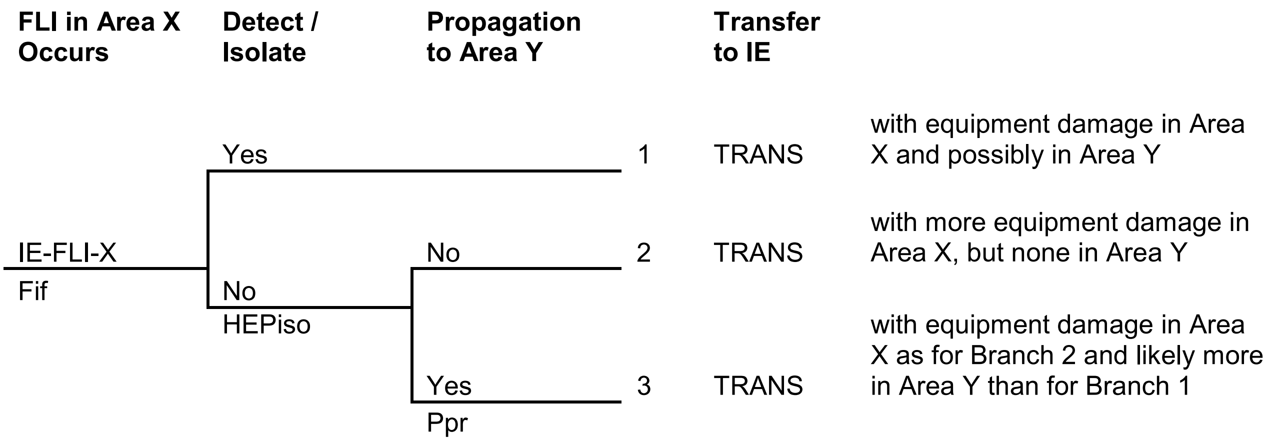
In medium and large diameter pipes, the breaks of smaller equivalent sizes can occur. The fraction of smaller sizes of breaks, given a failure in a larger pipe, can be calculated by using data in Table 3A-2-2.

A more recent data set for pipe failures by system and reactor type is also given in Table 3A-2-3 in units of per hr-per segment. Use of this data requires knowing the number of segments in question.

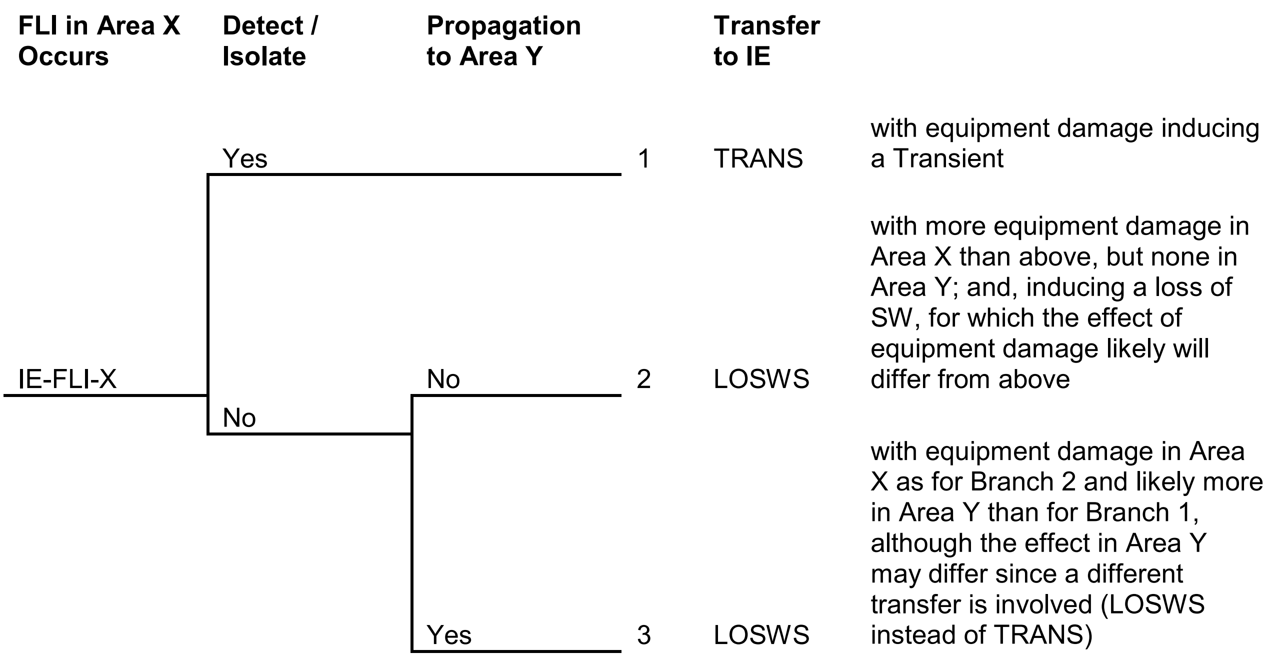
Finally, the initiating event frequencies of steam and feedline breaks are given in Table 3A-2-4.

3A-3 Quantification of Internal Flooding Initiating Event Frequencies

To calculate flooding initiating event frequencies, data from Tables 3A-2-1 through 3A-2-4 may be used. This requires knowing the number of segments or ft of piping involved. An example calculation is given in Section 3.3.



Additional event tree nodes may be added to introduce scenario-specific issues.
Transfers to other event trees are for illustration purposes only; others may be substituted, as needed.
For example:



Frequency of Scenario 1 = Fif * (1-HEPiso)
Frequency of Scenario 2 = Fif * HEPiso* (1-Ppr)
Frequency of Scenario 3 = Fif * HEPiso* Ppr

Figure 3A-1-1 Event Tree Model for Internal Flooding Scenario

Table 3A-1-1 Example Matrix Defining Internal Flooding Scenarios

	Name	Description	IE Frequency	Equipment Lost	IE Caused	HEPs / Basic Events Affected	New Basic Events (failures) Introduced
1	FLI-FL1	Turbine Building Basement Flood - Winter Conditions	8.90E-05	Non-vital air compressors; MCCs for non-vital air compressors and other components	IE-LOMFW	None/ None	IE-FLI-FL1
2	FLI-FL2	Turbine Building Basement Flood - Summer Conditions	1.10E-04	Non-vital air compressors; MCCs for non-vital air compressors and other components	IE-LOMFW	None/ None	IE-FLI-FL2
3	FLI-FL3	Diesel Generator Room A SW Connection Failure Flood	5.00E-04	4.16KV Bus 5; EDG A	IE-TRANS	None/ None	IE-FLI-FL3
4	FLI-FL4	Diesel Generator Room B SW Connection Failure Flood	5.00E-04	4.16KV Bus 6; EDG B	IE-TRANS	None/ None	IE-FLI-FL4
5	FLI-FL5	Relay Room Potable Water Flood	1.50E-04	None	IE-TRANS	None / None	IE-FLI-FL5
6	FLI-FL6	Control Rod Drive Equipment Room Service Water Flood	1.50E-04	None	IE-TRANS	None/ None	IE-FLI-FL6
		Sum =	**1.50E-03**				

Table 3A-1-2 Example Summary of A Plant X Turbine Building Flood Scenario

IE Name	Description	Flood Damage	Detection/ Isolation Means	Failed Gate or Component BEs[1]
CI06B	Rupture of an Inlet Condenser Expansion Joint in TU-22-1	Propagate: TU-94 TU-95B-1 Damage: Air Compressor 1F Air Compressor 1G Condensate Pump 1A Condensate Pump 1B Feedwater Pump 1A Feedwater Pump 1B Rx Makeup Pump 1A Rx Makeup Pump 1B Plt Equip Wtr Pump 1A Plt Equip Wtr Pump 1B MCC-32D MCC-42B MCC-42D AOV PW-52	Detect: Reactor Trip due to Loss of Condenser Vacuum Isolate: Trip both Circulating Water Pumps	Initiating Event: IE-CI06B Failed BEs: 01-CM-SIAC1F-PR 01-CM-SIAC1G-PR 03-PM--CDP1A-PR 03-PM--CDP1B-PR 05APM--FWP1A-PR 05APM--FWP1B-PR 27APM--RMP1A-PR 27APM--RMP1B-PR 27BPM-PEWPA—PR 27BPM-PEWPB—PR 40-BS-MCC32D-SG 40-BS-MCC42B-SG 40-BS-MCC42D-SG 26-AV-PW52---OC

Table 3A-2-1 Data for Calculating Internal Flooding Initiating Event Frequencies

Component Type	Rupture/Leakage (Note 4)	Rate (/hr)	Error Factor (Note 2)
Generic Piping (including elbows)	Leakage	3.0E-09 /hr-ft	10
	Non-PCS Rupture	1.2.0E-10 /hr-ft	30
	PCS Rupture	3.0E-11 /hr-ft	30
Valve	Leakage	1.0E-08	10
	Non-PCS Rupture	4.0E-10	30
	PCS Rupture	1.0E-10	30
Pump	Leakage	3.0E-08	10
	Non-PCS Rupture	1.2E-09	30
	PCS Rupture	3.0E-10	30
Flange	Leakage	1.0E-08	10
	Rupture (all)	1.0E-10	10
Heat Exchanger Tube Side	Leakage	1.0E-07	10
	Non-PCS Rupture	4.0E-09	30
	PCS Rupture	1.0E-09	30
Heat Exchanger Shell Side	Leakage	1.0E-08	10
	Non-PCS Rupture	4.0E-10	30
	PCS Rupture	1.0E-10	30
Tank	Leakage	1.0E-08	10
	Non-PCS Rupture	4.0E-10	30
	PCS Rupture	1.0E-10	30
Circulating Water Expansion Joint (Note 1)	Rupture	4.5E-05 /yr	

Notes:
1. Taken from Internal Flooding Analysis Supplemental Report for the Surry Nuclear Power Plant Individual Plant Examination, VEPCO/NUS, November 1991 (ADAMS microfiche no. 9112060076). All other data in the table are taken from Ref. 3-5.
2. Lognormal distribution is postulated.
3. It was assumed that the rupture of valves, pump casings, and other components have the same conditional probability of small, medium, large ruptures as for piping, as given in Table 3A-2-2.
4. Leakage <50 gpm; rupture >= 50gpm.

Table 3A-2-2 Conditional Probability of Small, Medium, Large Ruptures for Piping

Given Break in Medium Size Pipe (2”<=D<6”)	
Probability of Small Failure (D<2”)	0.5
Probability of Medium Failure	0.5
Given Break in Large Size Pipe (D>=6”)	
Probability of Small or Medium Failure (D<6”)	0.25
Probability of Large Failure	0.5

Data from Ref. 3-2. Breaks include all ruptures.

Table 3A-2-3 Rupture Failure Rates for Generic System Groups for Piping (1)

System	**Failure rate (per Section-hour) for Pipe Size Groups (2)**		
	.5” <= ID < 2”	**2” <= ID < 6”**	**6” <= ID**
BWR – Reactor Coolant System	7.54E-11	1.05E-10	1.06E-10
BWR – Safety Injection and Recirculation	1.47E-9	2.02E-9	2.06E-9
BWR - Other Safety-related Systems	8.65E-10	2.12E-10	6.62E-10
BWR – Main and Auxiliary Emergency Feedwater	2.30E-9	1.17E-9	3.4E-10
BWR - Main and Auxiliary and Extraction Steam and Turbine Systems	7.62E-11	2.72E-10	9.63E-10
Generic BWR	8.54E-10	4.66E-10	8.26E-10
PWR – Reactor Coolant System	2.13E-10	1.70E-11	2.87E-11
PWR – Safety Injection and Recirculation	1.42E-9	1.13E-10	1.92E-10
PWR - Other Safety-related Systems	7.09E-10	7.03E-11	1.39E-10
PWR – Main and Auxiliary Emergency Feedwater and Condensate Systems	7.39E-10	1.17E-9	6.4E-10
PWR - Main and Auxiliary and Extraction Steam and Turbine Systems	3.5E-10	9.77E-10	8.9E-10
Generic PWR	6.01E-10	3.98E-10	5.64E-10
Generic Plant	7.05E-10	4.16E-10	6.53E-10

Notes:

1. Rupture >50gpm. Use together with Table 3A-2-2 to calculate small, medium and large failures.
2. A pipe section is a segment of piping between major discontinuities such as valves, pumps, reducers, trees, etc. A pipe section is typically 10 to 100 feet long, and contains four to eight welds. Each pipe section can also contain several elbows and flanges. Instrumentation connections are not considered as major discontinuities.
3. Data from Ref. 3-2.

Table 3A-2-4 Generic Frequencies of Steam and Feedline Break Initiating Events

Event	Category	Mean Frequency	95th percentile
High Energy Line Steam Breaks/Leaks (combined)	K	1.3e-02	2.1e-02
Steam Line Break/leak Outside Containment	K1	1.0e-02	1.7e-02
Steam Line Break/leak Inside Containment – PWR only	K3	1.0e-03	3.9e-03
Feedwater Line Break/leak	K2	3.4e-03	7.6e-03

Notes:

K: High energy line break

K1: Steam line break outside containment: is a break of one inch equivalent diameter or more in a steam line located outside the primary containment that contains main turbine working fluid at or above atmospheric saturation conditions.

K2: Feedwater line break is a break of one inch equivalent diameter or more in a feedwater or condensate line that contains main turbine working fluid at or above atmospheric saturation conditions.

K3: Steam line break inside containment: is a break of one inch equivalent diameter or more in a steam line located inside the primary containment that contains main turbine working fluid at or above atmospheric saturation conditions.

See Ref. 3-6 for the Categories.

4.0 Seismic Event Modeling and Seismic Risk Quantification

4.1 Objectives and Scope

This document is intended to provide a concise and practical handbook to NRC risk analysts who routinely use the Systems Analysis Programs for Hands-on Integrated Reliability (SAPHIRE) software and the Standardized Plant Analysis Risk (SPAR) probabilistic risk assessment (PRA) models to quantify event and plant condition importances, and other ad-hoc risk analyses. It is a complementary document to the handbook cited in Ref. 4-1.

NRC risk analysts encounter many plant conditions and events reported by such means as inspection reports, licensee event reports (LERs), generic risk issues that lend themselves to PRA quantification and evaluation, every year. The need for quantification of the event / condition importance in terms of the two common risk measures of core damage frequency (CDF) and large early release frequency (LERF) arise in many of these cases.

This handbook provides NRC risk analysts practical guidance for modeling seismic event scenarios and quantifying their CDF using SPAR models and SAPHIRE software.

The handbook assumes that:

1. The user has hands-on experience with the SAPHIRE code;

2. The user has performed and documented event/condition importance analysis or plant risk assessment cases for a period of at least three months (this is a suggested period, not a firm limit) under the supervision of an experienced (qualified) senior PRA analyst. The user is the primary author of documentation packages for such analyses which are reviewed and accepted by an NRC program.

The current scope is limited to seismic events during power operation and calculation of CDF only.

Mainstream PRA terms and abbreviations that are used in this document are not defined; the intended reader is assumed to be familiar with them.

The seismic PRA (SPRA) model described in this handbook can be used for plants with SMA. See Section 4.2.8.

4.2 Seismic Event Scenario Definition

4.2.1 Minimum Input Requirements

The minimum input requirements for the seismic SPAR PRA model are as follows:

1. Seismic Hazard Vector (frequencies of seismic events)

2. Seismic fragilities of major structures, systems, and components (SSCs).

   Both of these inputs can be found in plants with SPRAs, and some of this information may be available for plants with seismic margins analyses. If not, generic hazard curves given in Appendix 4A, and SSC fragilities given in Appendix 4B may be used.

3. An event tree model representing the seismic sequences.

   Such an event tree model is provided as a default in a later section.

Those plants with existing SPRAs would also have dominant seismic sequences that can be used to validate the SPAR-EE model.

4.2.2 Example Seismic Hazard Vector

The seismic hazard vectors for 69 sites to the East of Rocky Mountains are given in Ref. 4-2. The default seismic hazard vectors for 72 SPAR model plants are given in Appendix 4A.

The seismic hazard vector for the example is taken from Ref. 4-2:

Table 4-1 Example Seismic Hazard Vector
(cumulative frequency of exceedance of a g value)

g value	mean f per year
0.05	3.040E-04
0.08	1.777E-04
0.15	6.422E-05
0.25	2.748E-05
0.30	1.979E-05
0.40	1.141E-05
0.50	7.212E-06
0.65	4.043E-06
0.80	2.474E-06
1.00	1.409E-06

This vector provides the seismic initiating event frequencies (seismic hazard distribution) as a function of seismic g level. The frequency of a seismic event of magnitude 0.05g or higher is given as 3.04E-04/year.

The plant is designed to withstand a design basis earthquake (DBE) (also known as safe shutdown earthquake (SSE)) of 0.12g peak ground acceleration (PGA). The operating-basis earthquake (OBE) is 0.06g.

4.2.3 Seismic Event Categories

The seismic acceleration range can be partitioned into N categories (bins) to define N discrete seismic event scenarios with increasing intensity. This handbook recommends using three seismic bins as defined below, unless plant-specific considerations require more bins. A larger number bins can be readily introduced into the SPAR models without taxing their running times.

A larger number of bins may be appropriate for the four sites to the West of the Rocky Mountains.

For the example case above, three seismic event categories are defined as follows:

		IE Frequency
IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.84E-04
IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.26E-05
IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.21E-06

The frequencies are calculated as shown in Table 4-2. Any reasonable number of seismic bins may be defined, as needed. The need may be based on two factors:

1. Seismicity of the site (seismically more active sites may require more bins);

2. Fragility grouping of major SSCs (one or more key SSCs with a fragility in a seismic range may warrant a bin in that range to make the model more realistic).

The three seismic bins chosen here follow the Limerick external events feasibility study (Ref. 4-3). The first bin is driven by seismically induced loss-of-offsite power (LOOP) events; the third bin is driven by the seismic failure of major structures, leading to direct core damage. The second bin captures other modeled events (small loss-of-coolant accident (SLOCA), large loss-of-coolant accident (LLOCA), LOOP, structural failures).

Table 4-2 Calculation of Bin Accelerations and Frequencies

Ground Acceleration (g)	Exceedance Frequency	Seismic Bin	Bin Acceleration	Bin Frequency
0.05	3.040E-04	1 (0.05-0.3g)	0.122474	2.842E-04
0.08	1.777E-04			
0.15	6.422E-05			
0.25	2.748E-05			
0.30	1.979E-05	2 (0.3-0.5g)	0.387298	1.258E-05
0.40	1.141E-05			
0.50	7.212E-06	3 (>0.5g)	0.707107	7.212E-06
0.65	4.043E-06			
0.80	2.474E-06			
1.00	1.409E-06			
			Sum =	**3.040E-04**

Bin acceleration is calculated as a geometric average of two bin range limits.
Bin frequency is calculated as the difference of the frequencies of two bin range limits.

To each bin, a mean acceleration is assigned in terms of the geometric average of the bin end points. For the three bins in question, the bin accelerations are:

Seismic Bin	Bin Acceleration
BIN-1 (0.05-0.3g)	0.122
BIN-2 (0.3-0.5g)	0.387
BIN-3 (>0.5g)	0.707

The seismic failure probabilities of SSCs are calculated at these bin acceleration levels, in the next task.

After the next step (4.2.4) is completed, redefinition of the seismic event categories (number of bins, or the bin ranges) may be required if plant-specific low fragility SSCs are identified.

4.2.4 SSC Seismic Fragilities

The fragilities of the major SSCs must be obtained (or assigned from generic sources) to calculate mean seismic failure probabilities, which then are added to the existing random failure probabilities. The example SPRA provides various SSC probabilities; it also defines a surrogate SSC, whose fragility is used in a conservative, generic manner for some key SSCs. Table 4-3 shows an example of the fragilities considered and how they are treated for SPAR-EE purposes. The list of key SSCs are taken from the Example SPRA.

The fragility information needed for a SSC is either,

Median capacity a_m and β_c OR

Median capacity a_m, β_r and β_u.

$\beta_c = (\beta_r^2 + \beta_u^2)^{1/2}$

The mean seismic failure probability $P_{fail}(a)$ at a bin acceleration level can be calculated by using the following equation:

$P_{fail}(a) = \Phi\ [\ \ln(a/a_m) / (\beta_r^2 + \beta_u^2)^{1/2}]$

Where Φ is the standard normal cumulative distribution function and

a = median acceleration level of the seismic event;
a_m = median of the component fragility (or median capacity);
β_r = logarithmic standard deviation representing random uncertainty;
β_u = logarithmic standard deviation representing systematic or modeling uncertainty.

The SSC high confidence of low probability of failure (HCLPF) value is calculated by the equation:

HCLPF = $a_m \exp(1.645(\beta_r + \beta_u)$

Table 4-3 SSC Fragilities and Their Treatment in SPAR-EE

SSC Description	Median Capacity (g)	β_c OR β_r	β_u	SSC Failure probability	Comment	HCLPF
Offsite Power	0.35	0.55		2.77E-02	LOOP-EQ-1	
	0.35	0.55		5.72E-01	LOOP-EQ-1	
	0.35	0.55		8.99E-01	LOOP-EQ-3	
RHR Heat Exchanger	0.63	0.46		1.79E-04	RHR-HX-EQ1	
	0.63	0.46		1.45E-01	RHR-HX-EQ2	
	0.63	0.46		5.99E-01	RHR-HX-EQ3	
Surrogate Element	0.64	0.3		1.65E-08		
	0.64	0.3		4.68E-02		0.68
	0.64	0.3		6.30E-01		
Reactor Pressure Vessel	2	0.3	0.35	6.53E-10	CD	
Reactor Pressure Vessel Supports	2	0.3	0.35	1.83E-04	CD	0.75
	2	0.3	0.35	1.20E-02	CD	
Steam Generators	2.5	0.3	0.4	7.73E-10	CD	
Steam Generator Supports	2.5	0.3	0.4	9.53E-05	CD	0.75
	2.5	0.3	0.4	5.77E-03	CD	
Pressurizer	2.5	0.3	0.4	7.73E-10	LLOCA	
Pressurizer Supports	2.5	0.3	0.4	9.53E-05	LLOCA	0.75
	2.5	0.3	0.4	5.77E-03	LLOCA	
Reactor Coolant Pumps	2.5	0.3	0.4	7.73E-10	LLOCA	
Reactor Coolant Pump Supports	2.5	0.3	0.4	9.53E-05	LLOCA	0.75
	2.5	0.3	0.4	5.77E-03	LLOCA	
Control Rod Drive Mechanism	2.5	0.3	0.4	7.73E-10	ATWS	
Reactor Core Upper Internals	2.5	0.3	0.4	9.53E-05	ATWS	0.93
	2.5	0.3	0.4	5.77E-03	ATWS	
Reactor Coolant System Piping	3.8	0.35	0.5	8.82E-09	CD	
	3.8	0.35	0.5	9.10E-05	CD	
	3.8	0.35	0.5	2.93E-03	CD	0.37
Containment Building	1.1	0.3	3.50E-01	9.20E-07	CD	
Auxiliary Building	1.1	0.3	3.50E-01	1.17E-02	CD	
Turbine Building	1.1	0.3	3.50E-01	1.69E-01	CD	

SSC Description	Median Capacity (g)	β_c OR β_r	β_u	SSC Failure probability	Comment	HCLPF
Reactor Coolant Pump Seals	not modeled				SLOCA	
Secondary Side Piping and Supports	not modeled				SLB	
Switchyard Ceramic Insulators	modeled above				LOOP	
Screenhouse	surrogate element is used in SWS FT				SW	
Instrument Air	Assumed failed in SPRA due to low fragility. Feed and bleed not credited in SPR. Feed and bleed is not credited in SPAR-EE either.					
CST	Assumed failed due to low fragility in SPRA. SWS is credited as alternate. However, could not be modeled in SPAR since SPAR success criteria requires both.					
RPS	Failure to scram is modeled in the RPS fault tree; surrogate element is used.					

g	SLOCA	MLOCA	LLOCA	ATWS	LOOP	CD-EQ
0.122	1.50E-05	1.00E-07	1.23E-08	7.73E-10	2.77E-02	2.77E-06
0.387	4.50E-02	4.00E-03	5.91E-04	9.53E-05	5.72E-01	3.55E-02
0.707	2.50E-01	4.00E-02	1.55E-02	5.77E-03	8.99E-01	5.27E-01

SLOCA and MLOCA — IE frequencies are taken from NURE/CR-4840, Figure 3-6, as in SPRA.
LLOCA — sum of SG, RCP, PRESURIZER, and .1 times MLOCA.
ATWS — from RPS
LOOP — From Offsite Power
CD-EQ — Sum of RVF,SG,RCS piping, and 3 buildings (Containment, Aux., Turbine)
Plant-specific SPRA assignments are used when available

Table 4-4 SSC Fragilities and Their Treatment in Plant C SPAR-EE

	SSC Description	Median Capacity (g)	β_r	β_u	SSC Failure probability	Comment	HCLPF
1	Reactor Pressure Vessel	2	0.3	0.35	6.53E-10	CD	0.69
	Reactor Pressure Vessel Supports	2	0.3	0.35	1.83E-04	CD	
		2	0.3	0.35	1.20E-02	CD	
2	Steam Generators	2.5	0.3	0.40	7.73E-10	CD	0.79
	Steam Generator Supports	2.5	0.3	0.40	9.53E-05	CD	
		2.5	0.3	0.40	5.77E-03	CD	
3	Reactor Coolant System Piping	3.8	0.35	0.50	8.82E-09	CD	0.94
		3.8	0.3	0.35	3.61E-07	CD	
		3.8	0.3	0.35	1.32E-04	CD	
4	Buildings (including containment, turbine and auxiliary buildings)	1.1	0.2	0.35	2.45E-08	CD	0.45
		1.1	0.2	0.35	4.78E-03	CD	
		1.1	0.2	0.35	1.36E-01	CD	
5	CD-EQ1	sum of 1,2,3,4			3.48E-08	CD	
	CD-EQ2				5.06E-03	CD	
	CD-EQ3				1.54E-01	CD	
6	Reactor Coolant Pumps	2.5	0.3	0.40	7.73E-10	LLOCA	0.79
	Reactor Coolant Pump Supports	2.5	0.3	0.40	9.53E-05	LLOCA	
		2.5	0.3	0.40	5.77E-03	LLOCA	
7	Pressurizer	2.5	0.3	0.40	7.73E-10	LLOCA	0.79
	Pressurizer Supports	2.5	0.3	0.40	9.53E-05	LLOCA	
		2.5	0.3	0.40	5.77E-03	LLOCA	
8	10% of MLOCA	**			1.00E-08	LLOCA	
		**			4.00E-04	LLOCA	
		**			4.00E-03	LLOCA	
9	LLOCA-EQ1	sum of 6,7,8			1.15E-08	LLOCA	
	LLOCA-EQ2				5.91E-04	LLOCA	
	LLOCA-EQ3				1.55E-02	LLOCA	
10	SLOCA-EQ1	**			1.50E-05	SLOCA	
	SLOCA-EQ2	**			4.50E-02	SLOCA	
	SLOCA-EQ3	**			2.50E-01	SLOCA	
11	Offsite Power	0.3	0.3	0.35	2.55E-02	LOOP-EQ-1	0.10
		0.3	0.3	0.35	7.10E-01	LOOP-EQ-1	

	SSC Description	Median Capacity (g)	β_r	β_u	SSC Failure probability	Comment	HCLPF
		0.3	0.3	0.35	9.69E-01	LOOP-EQ-3	
12	Control Rod Drive Mechanism	1.8	0.3	0.40	3.67E-08	RPS-EQ-1	0.57
	Reactor Core Upper Internals	1.8	0.3	0.40	1.06E-03	RPS-EQ-2	
		1.8	0.3	0.40	3.08E-02	RPS-EQ-3	
13	EDGs	1.45	0.3	0.35	3.95E-08	EDG-EQ-1	0.50
		1.45	0.3	0.35	2.08E-03	EDG-EQ-2	
		1.45	0.3	0.35	5.96E-02	EDG-EQ-3	
14	CST	1.1	0.3	0.35	9.20E-07	AFW-EQ-1	0.38
		1.1	0.3	0.35	1.17E-02	AFW-EQ-2	
		1.1	0.3	0.35	1.69E-01	AFW-EQ-3	
15	CCW	1.45	0.3	0.35	3.95E-08	CCW-EQ-1	0.50
		1.45	0.3	0.35	2.08E-03	CCW-EQ-2	
		1.45	0.3	0.35	5.96E-02	CCW-EQ-3	
16	RWST	1.1	0.3	0.35	9.20E-07	HPI-EQ-1 *	0.38
		1.1	0.3	0.35	1.17E-02	HPI-EQ-2 *	
		1.1	0.3	0.35	1.69E-01	HPI-EQ-3 *	
17	Screenhouse	1.1	0.3	0.35	9.20E-07	SWS-EQ-1	0.38
		1.1	0.3	0.35	1.17E-02	SWS-EQ-2	
		1.1	0.3	0.35	1.69E-01	SWS-EQ-2	
18	Battery Chargers	1.6	0.3	0.35	1.18E-08	DC-EQ-1	0.55
		1.6	0.3	0.35	1.04E-03	DC-EQ-2	
		1.6	0.3	0.35	3.82E-02	DC-EQ-3	

Notes:

* also use in LPI-EQ1
LPI-EQ2
LPI-EQ3

** SLOCA and MLOCA IE frequencies are taken from NURE/CR-4840, Figure 3-6.

g level	SLOCA	MLOCA
0.122	1.50E-05	1.00E-07
0.387	4.50E-02	4.00E-03
0.707	2.50E-01	4.00E-02

Generally, the ceramic insulators with the lowest fragilities among the SSCs modeled in the PRAs govern the occurrence of LOOP following a seismic event in many plants. The generic fragility data for ceramic insulators may be taken from Ref. 4-4, if not already available in the plant-specific documentation. Appendix 4B provides a convenient table for generic seismic fragilities of commonly considered SSCs.

The fragilities of the key SSCs can be ordered from lowest to highest in a table; the lower fragilities will determine the number of bins and their ranges; the lowest of the critical SSC fragilities would help determine the highest bin. A critical SSC is one if failed would lead to core damage: examples include containment, fuel, reactor pressure vessel, Steam generators including their supports, etc.

Bin definitions may be revisited/revised after SSC fragilities are modeled.

Table 4B-1 gives the generic SSC seismic fragilities. The table also provides the SSC failure probabilities in each bin.

Tables 4-2 and 4-3 show some examples of how SSC fragilities are used in two plant SPAR-EE models.

The following list illustrates the candidate SSCs for a SPRA (the list is taken from a specific SPAR and is not intended to be an exhaustive list).

Important Structures
Containment building Concrete internal structure Auxiliary building Turbine building Intake structure Refueling water and condensate storage tanks Diesel Generator fuel oil storage tank (buried) Auxiliary saltwater system piping (buried)
Major Plant System
Nuclear steam supply system Residual heat removal system Safety Injection system Component cooling water system Chemical and volume control system Auxiliary saltwater system Containment spray system Main steam system Auxiliary feedwater system Diesel generator and auxiliaries Containment building ventilation system Control room ventilation system Vital electrical room ventilation system 4160 V (vital) electrical system 480 V (vital) electrical system 125 V DC electrical system Operator instrumentation and control system NSSS instrumentation and control system Off-site power system

Typical Generic Component Categories
Electrical penetrations Balance-of-plant piping and supports Air and motor operated valves Cable tray, conduits, and supports HVAC ducting and supports

4.2.5 Event Tree Models

The three seismic event tree models developed for the three seismic bins are shown in Figures 4-1 through 4-3.

The example SPRA also modeled medium loss-of-coolant accident (MLOCA), but its CDF was not dominant. It is left out of the current SPAR-EE model. If necessary, it can be added as a transfer into the seismic event trees with minimal additional work. Other events may also be considered on a plant-specific basis and may be added to the model as needed.

4.2.6 Fault Tree Models

The following new fault trees are introduced to represent the seismic event tree nodes. Each of these fault trees contain a single probability and allow transfer into a target event tree, or directly go to a CD end state:

CD-EQ1
CD-EQ2
CD-EQ3
LLOCA-EQ1
LLOCA-EQ2
LLOCA-EQ3
LOOP-EQ1
LOOP-EQ2
LOOP-EQ3
SLOCA-EQ1
SLOCA-EQ2
SLOCA-EQ3

The existing front line and support system fault trees need to be modified to include seismic faults. Figure 4-4 shows an example for a front line system. The RPS fault tree top logic is revised to include seismic failure basic events. The seismic subtree introduced into the RPS fault tree is shown in Figure 4-5.

Figures 4-6, 4-7, and 4-8 show how seismic subtrees are introduced into a support system.

Seismic fault trees can be added to as many system models as needed, determined by the number of low fragility SSCs.

The seismic sub trees are only activated when the seismic event bin in question is quantified and its flag is set to TRUE.

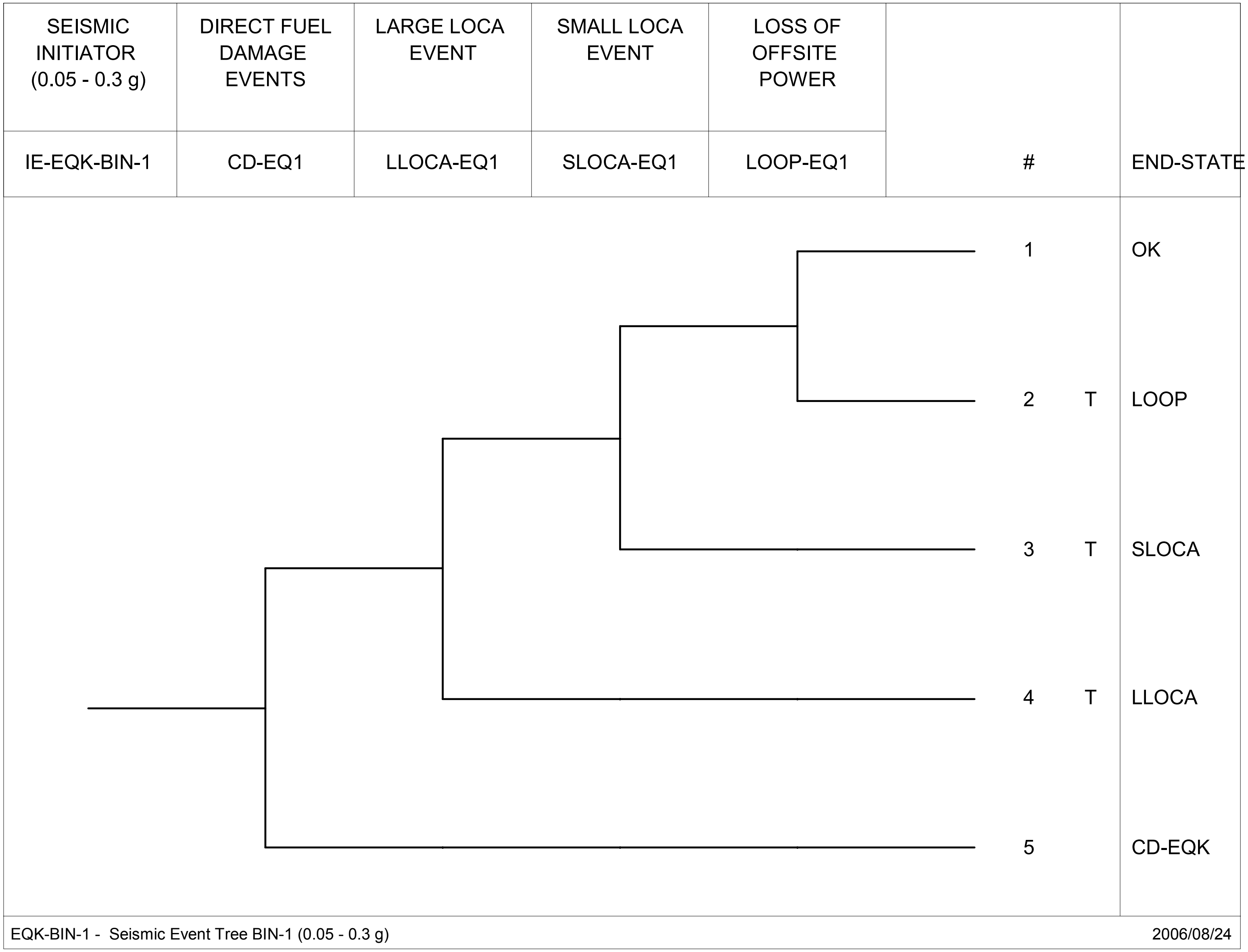


Figure 4-1 Seismic Event BIN-1 Event Tree

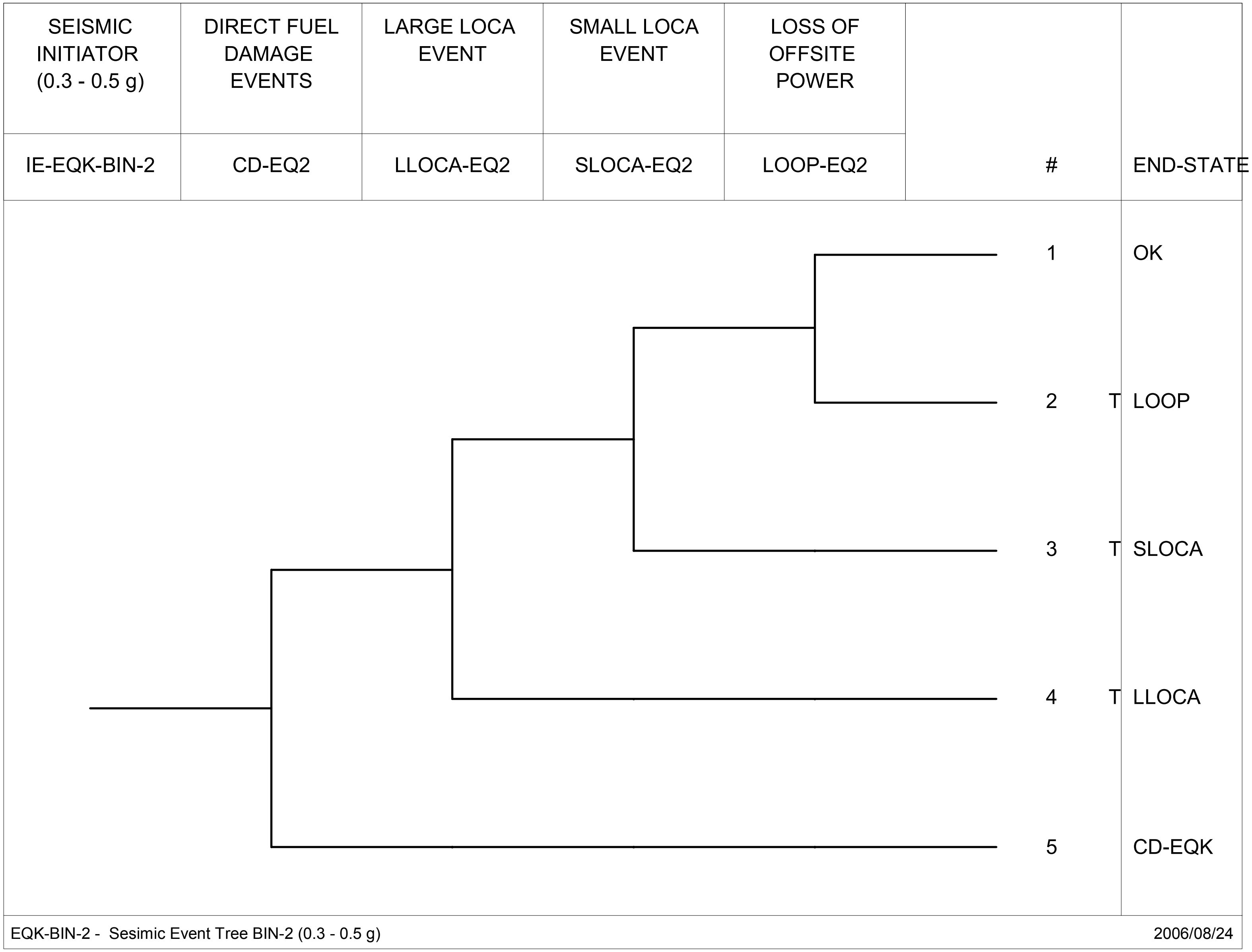


Figure 4-2 Seismic Event BIN-2 Event Tree

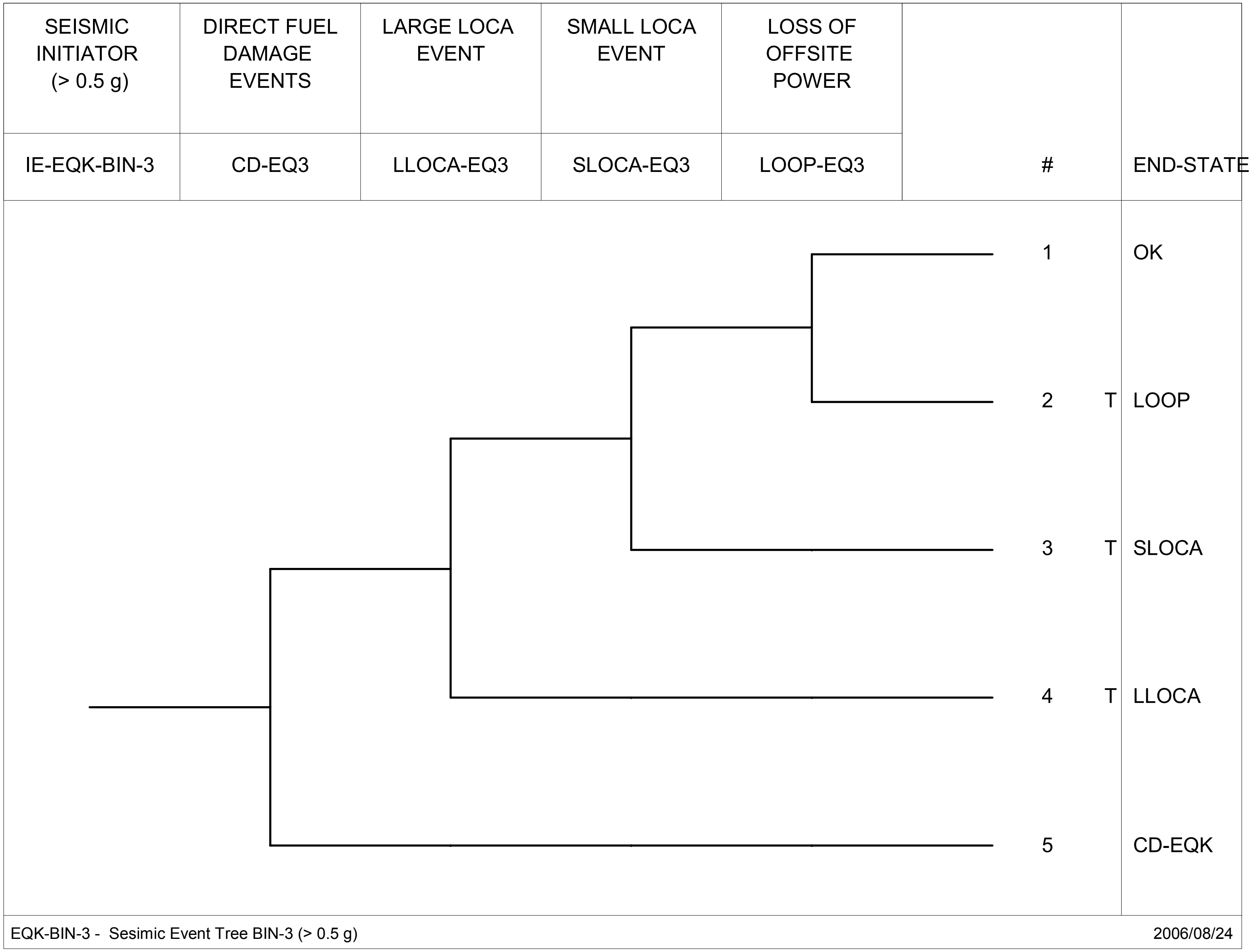


Figure 4-3 Seismic Event BIN-3 Event Tree

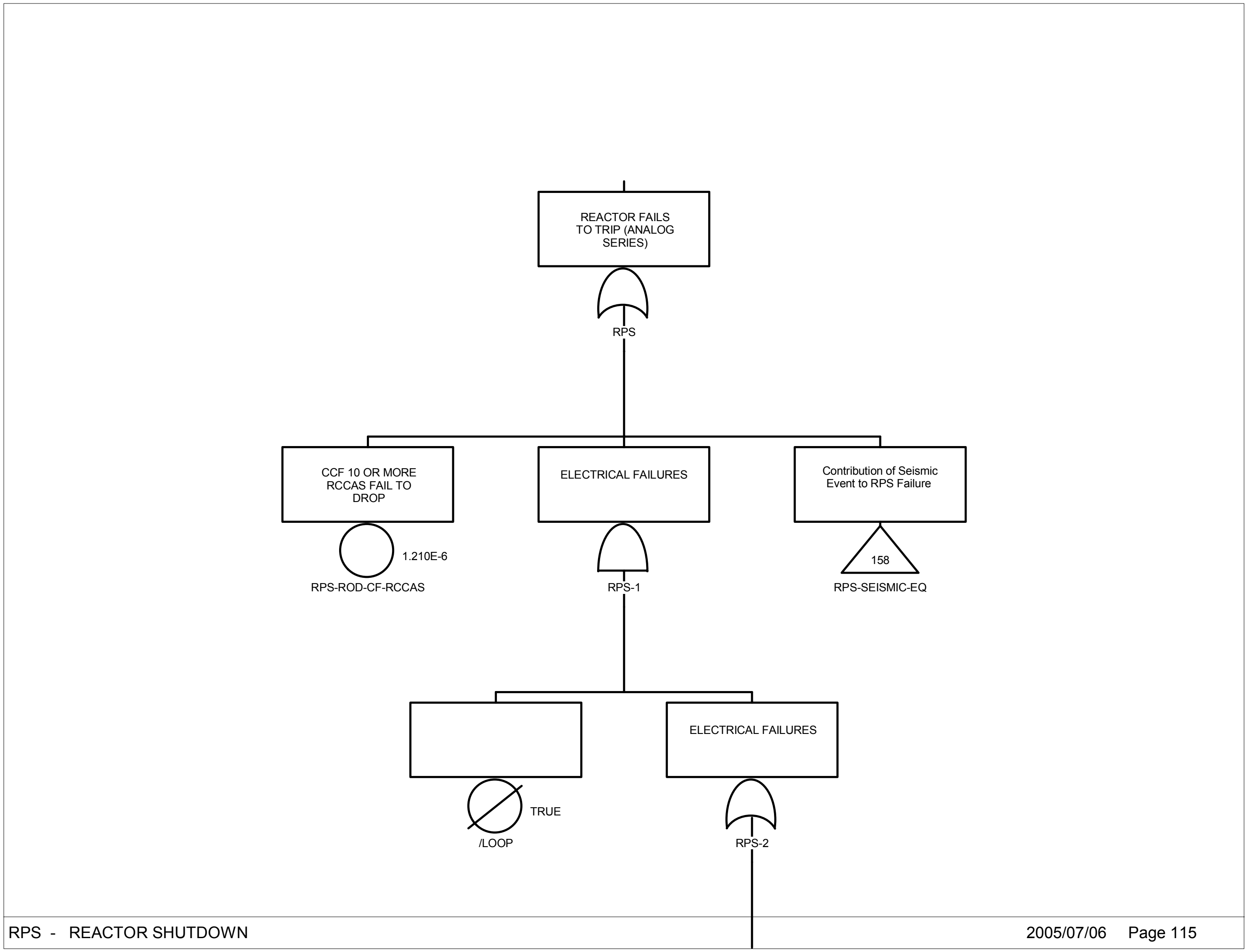


Figure 4-4 RPS Fault Tree (partial top showing introduction of seismic faults)

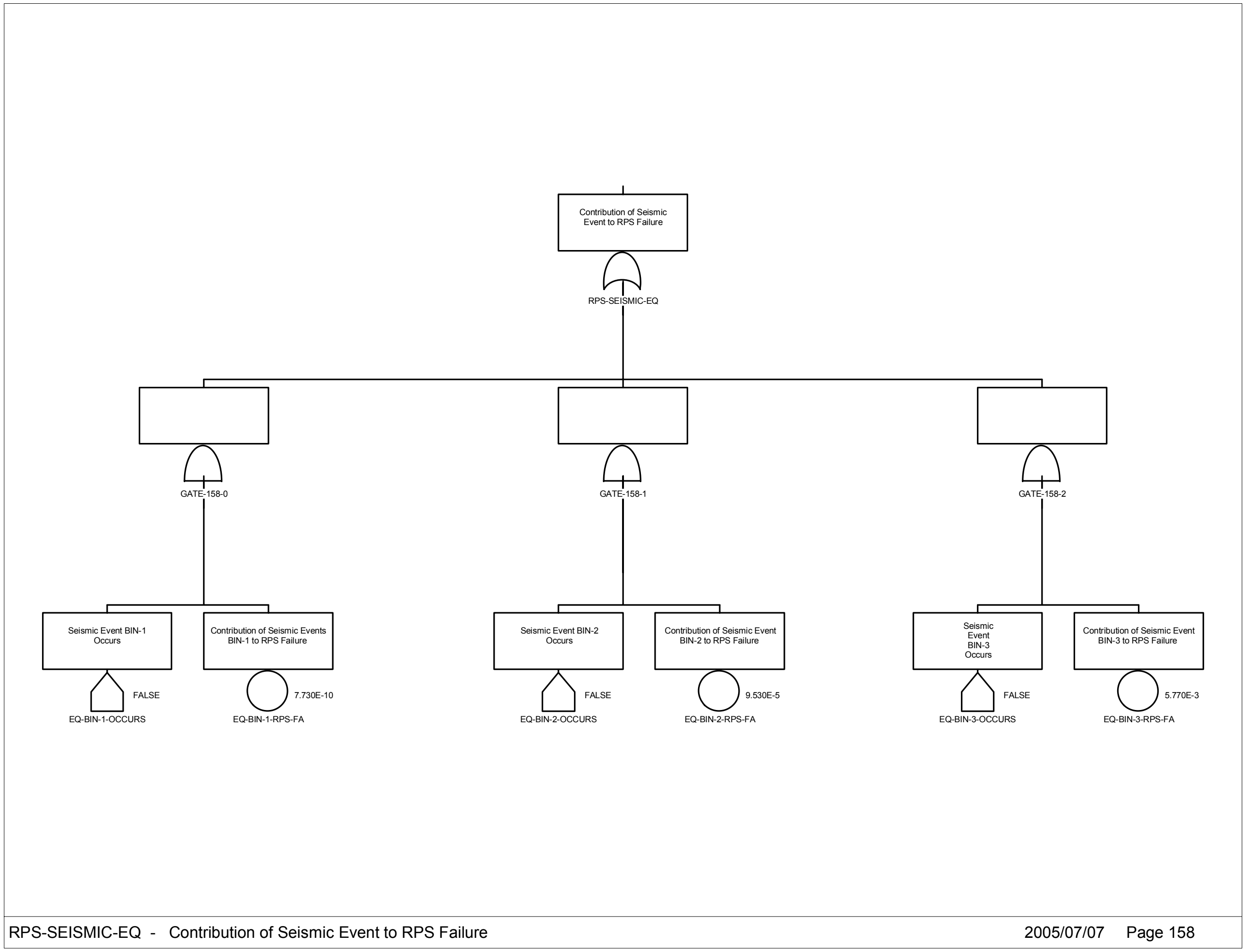


Figure 4-5 RPS-SEISMIC-EQ Fault Tree

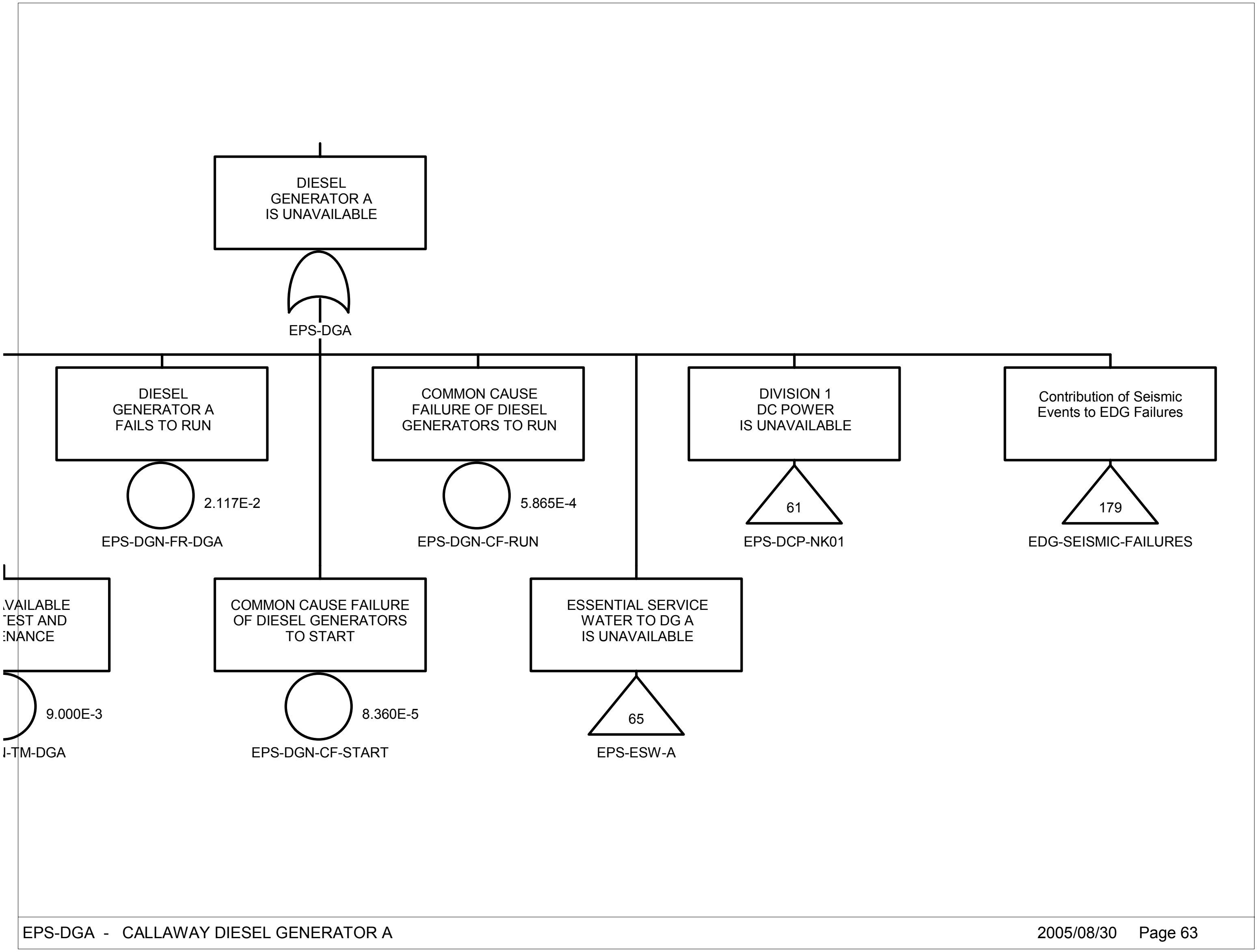


Figure 4-6 Adding Seismic Failures to a Support System - Figure 1 of 3

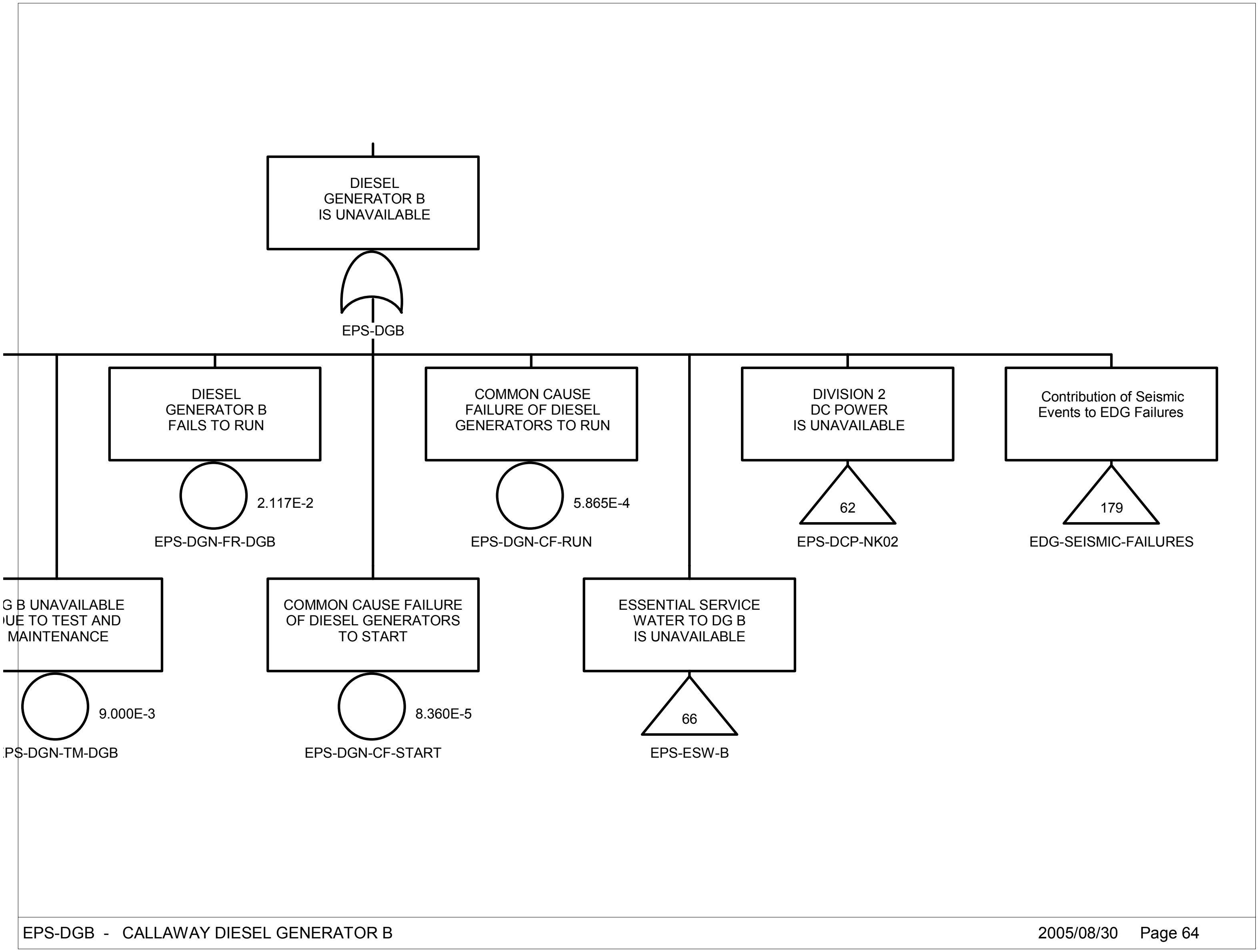


Figure 4-7 Adding Seismic Failures to a Support System - Figure 2 of 3

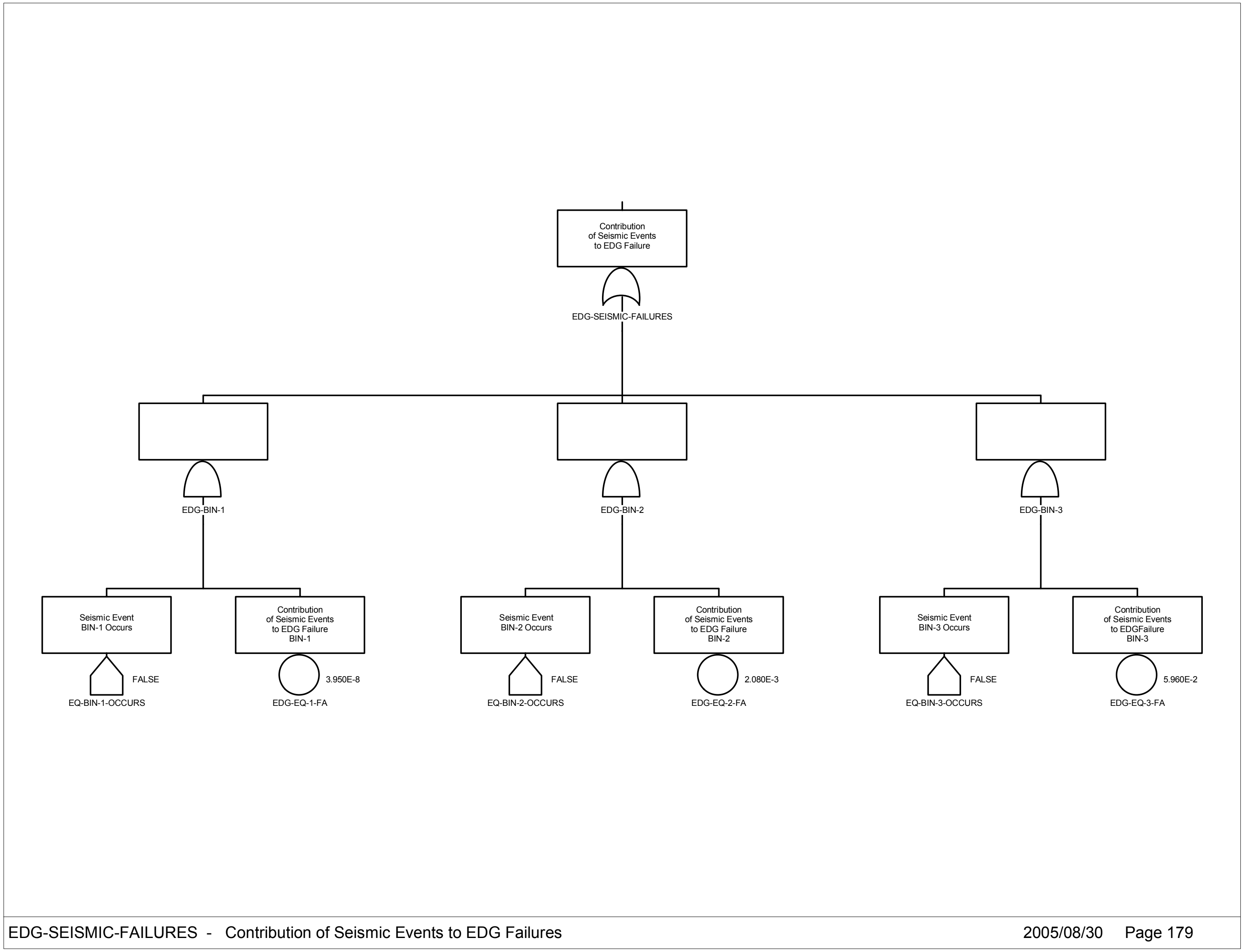


Figure 4-8 Adding Seismic Failures to a Support System - Figure 3 of 3

4.2.7 New Basic Events

The phrase basic event is used for any record in the SPAR-EE data base, which can be accessed with the SAPHIRE menu items MODIFY / BASIC EVENT. Four types of new basic events are introduced:

1. Initiating event frequencies;
2. basic events;
3. flags – house events;
4. Fault tree (FT) names; some FT names can be used as basic events (FT not further developed; FT name is used as the basic event).

Example of basic events introduced in this SPAR-EE are given in Table 4-5.

For some basic events represented by the FT value, the process flags are set to type W to make sure that the success path includes the success probability of the FT. This is done for basic events like CD-EQ3 where the seismic failure probability is very high.

4.2.8 Application to SMA Plants

The model described above is applicable to plants which have SMA. For an SMA plant, the following process applies:

i). Obtain the seismic hazard vector from Appendix 4A. Calculate BIN frequencies and assign bin acceleration levels.

ii). Examine the SMA documentation to locate any SSC fragilities and/or HCLPFs. Supplement that information with generic fragilities from Appendix 4B.

If a plant-specific HCLPF value is given in SMA, use that value and the corresponding β_r and β_u from Table 4B-1 to calculate median acceleration. Then use the median acceleration and the betas to calculate SSC failure probabilities for each BIN.

iii). Once the above data is assembled, proceed with modeling as in SPRA.

4.3 Special Modeling Considerations

This section discusses some special issues worth noting for seismic scenario modeling.

4.3.1 Non-safety Systems

The non-safety systems credited in the PRA have high likelihood of failure in BINs 2 and 3. As a precaution, they should not be credited at least in BINs 2 and 3. Such systems include main feedwater, normal service water, and instrument and service air.

4.3.2 Seismically-induced LOOP

The frequencies of seismically-induced LOOP events, based on the lowest fragility SSCs (such as ceramic insulators) can be calculated with the information available in Appendices 4A and 4B. Such a calculation is done for all 72 SPAR model plants and is given in Appendix 1 of this volume.

Table 4-5 New Basic Events

Name	Description	Calc. Prob.	
CD-EQ1	DIRECT FUEL DAMAGE EVENTS	2.77E-06	FT name; also used as BE
CD-EQ2	DIRECT FUEL DAMAGE EVENTS	3.55E-02	FT name; also used as BE
CD-EQ3	DIRECT FUEL DAMAGE EVENTS	5.27E-01	FT name; also used as BE
EQ-BIN-1-OCCURS	Seismic Event BIN-1 Occurs	0.00E+00	Flag (house event)
EQ-BIN-1-RHR-FA	Contribution of Seismic Event BIN-1 to RHR Failure	1.79E-04	BE
EQ-BIN-1-RPS-FA	Contribution of Seismic Events BIN-1 to RPS Failure	7.73E-10	BE
EQ-BIN-1-SWS-FA	Contribution of Seismic BIN-1 to SWS Failure	1.65E-08	BE
EQ-BIN-2-OCCURS	Seismic Event BIN-2 Occurs	0.00E+00	Flag (house event)
EQ-BIN-2-RHR-FA	Contribution of Seismic BIN-2 to RHR Failure	1.45E-01	BE
EQ-BIN-2-RPS-FA	Contribution of Seismic Event BIN-2 to RPS Failure	9.53E-05	BE
EQ-BIN-2-SWS-FA	Contribution of Seismic BIN-2 to SWS Failure	4.68E-02	BE
EQ-BIN-3-OCCURS	Seismic Event BIN-3 Occurs	0.00E+00	Flag (house event)
EQ-BIN-3-RHR-FA	Contribution of Seismic Event BIN-3 to RHR Failure	5.99E-01	BE
EQ-BIN-3-RPS-FA	Contribution of Seismic Event BIN-3 to RPS Failure	5.77E-03	BE
EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.30E-01	BE
IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.84E-04	IE
IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.26E-05	IE
IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.21E-06	IE
LLOCA-EQ1	LARGE LOCA EVENT	1.23E-08	FT name; also used as BE
LLOCA-EQ2	LARGE LOCA EVENT	5.91E-04	FT name; also used as BE
LLOCA-EQ3	LARGE LOCA EVENT	1.55E-02	FT name; also used as BE
LOOP-EQ1	LOSS OF OFFSITE POWER	2.77E-02	FT name; also used as BE
LOOP-EQ2	LOSS OF OFFSITE POWER	5.72E-01	FT name; also used as BE
LOOP-EQ3	LOSS OF OFFSITE POWER	8.99E-01	FT name; also used as BE
RHR-SEISMIC-EQ	Contribution of Seismic Event to RHR Failure	1.00E+00	FT name
RPS-SEISMIC-EQ	Contribution of Seismic Event to RPS Failure	1.00E+00	FT name
SLOCA-EQ1	SMALL LOCA EVENT	1.50E-05	FT name; also used as BE
SLOCA-EQ2	SMALL LOCA EVENT	4.50E-02	FT name; also used as BE
SLOCA-EQ3	SMALL LOCA EVENT	2.50E-01	FT name; also used as BE
SWS-SEISMIC-EQ	Contribution of Seismic Events to SWS Failure (Screenhouse)	1.00E+00	FT name

It is recommended that LOOP conditions are postulated without offsite power recovery for SLOCA and LLOCA paths (e.g., emergency buses are supported only by the onsite safety-related power sources).

If credit is taken for other AC power sources (other than normal offsite power and onsite emergency power) for SBO analysis, these power sources may need to be discredited.

4.3.3 Operator Actions

The failure probabilities of some operator actions may increase under high-g seismic event conditions. To be prudent the analyst should examine the set of operator actions modeled in the PRA and revise their HEPs if needed, for seismic scenarios. Especially, operator actions implied in recovery (such as power recovery) must be critically examined and adjusted if necessary.

In the absence of accepted methods, sensitivity analyses may be performed to understand and assess risk factors.

See also relay chatter section below.

4.3.4 Relay Chatter

The relay chatter evaluation addresses the questions of

a. whether the overall plant safety system could be adversely affected by relay malfunction in a seismic event and

b. whether the relays for which malfunction is unacceptable have an adequate seismic capacity.

Relay chatter may introduce system actuation failure or spurious actuation. Operator actions may be needed for starting otherwise auto-start safety systems. This handbook does not address modeling of relay chatter explicitly. However, it should be noted that generic relay seismic fragilities may be on the lower side, as shown in Table 4B-1.

See NUREG/CR-4840, page 3-32 for a discussion.

Unless the Individual Plant Examination of External Events (IPEEE) or similar reports identified relay chatter vulnerabilities, this issue need not be pursued for evaluation purposes.

4.3.5 Seismically-induced Internal Flooding

In seismic BINs 2 and 3, non-safety system piping failures in the Turbine building could create internal flooding concerns that can potentially fail other components either directly or through propagation of the flood into other areas. These issues are not further pursued in this handbook.

4.3.6 Seismically-induced Fires

The following four seismic-fire interaction issues are identified in the literature:

1. Seismically induced fires,
2. Degradation of fire suppression systems and features,
3. Spurious actuation of suppression and/or detection systems, and
4. Degradation of manual firefighting effectiveness.

It is recommended that a Fire PRA include a qualitative assessment of these issues.

4.3.7 Seismically-induced SLOCA and MLOCA

Generic frequencies of seismically induced SLOCA and MLOCA can be calculated from Figure 3-6 of NUREG/CR-4840. Figure 4-9 of this handbook shows the calculations for the pga values for the three seismic bins discussed in Section 4.2. A curve-fit has been implemented to the graph in Figure 3-6 to provide analysts SLOCA and MLOCA probabilities for different pga values. An MS EXCEL file containing these values is placed in ADAMS with accession number ML071220066.

4.4 CDF Quantification for Seismic Events

This section summarizes the CDF quantification for seismic events only.

Seismic sequences are automatically generated from the three seismic event trees and their CDF frequencies are quantified and CDF cutsets are identified using the SAPHIRE software. Tables 4-6 through 4-8 provide an illustration of the results and output for a plant-specific SPAR-EE seismic PRA model.

4.5 LERF Quantification for Seismic Events

LERF modeling and quantification is not currently addressed.

4.6 References

4-1. U.S. Nuclear Regulatory Commission, “Risk Assessment of Operational Events Handbook: Volume 1 - Internal Events,” Revision 1.01, December 2007.

4-2. U.S. Nuclear Regulatory Commission, “Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains,” NUREG-1488, April 1994.

4-3. Idaho National Laboratory, “A Feasibility and Demonstration Study – Incorporating External Events into SPAR Models,” February 2005.

4-4. U.S. Nuclear Regulatory Commission, “Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences," NUREG/CR-6544, April 1998.

4-5. U.S. Nuclear Regulatory Commission, "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150," NUREG/CR-4840, November 1990.

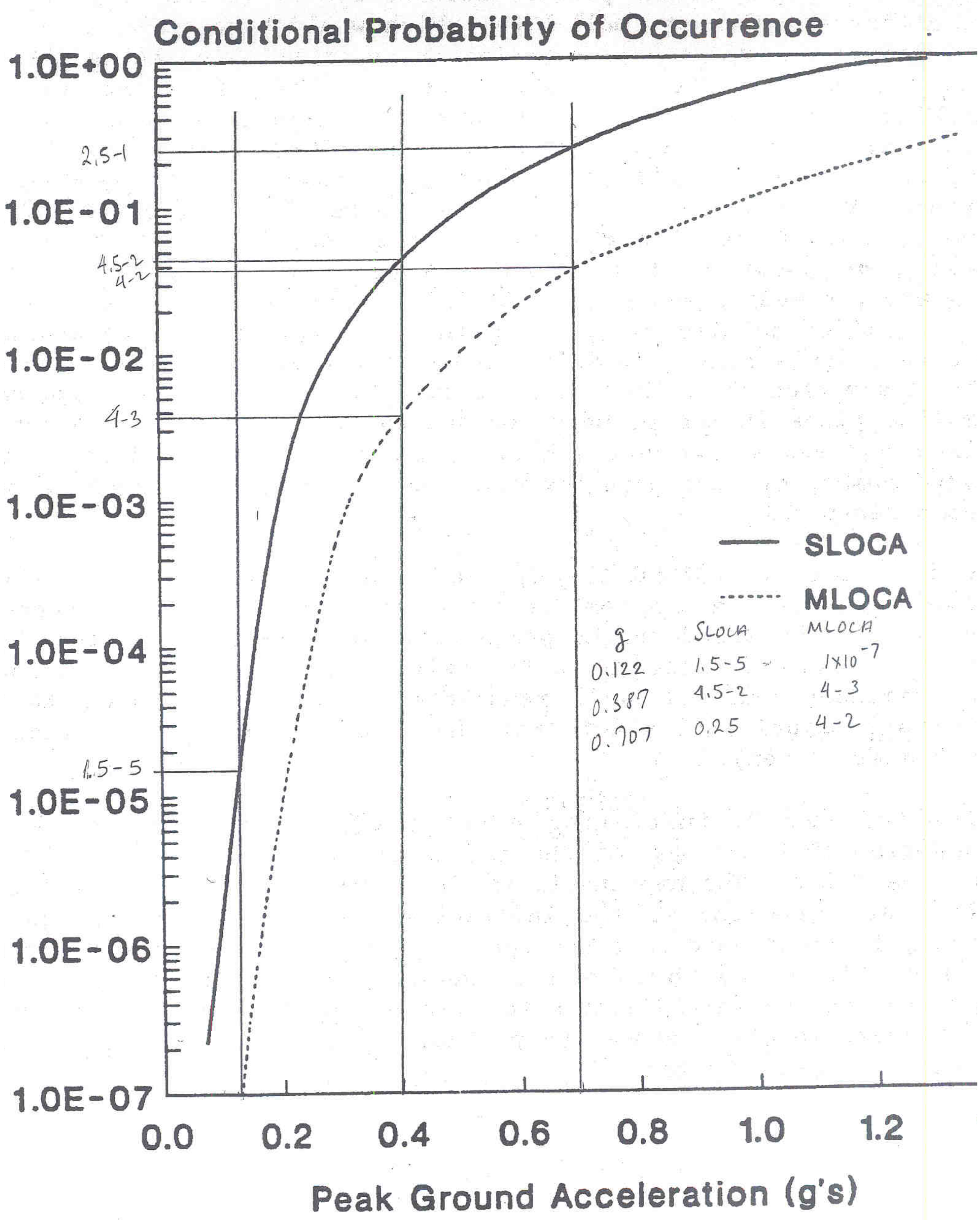


Figure 4-9 Estimation of Seismically-induced SLOCA and MLOCA Probabilities (NUREG/CR-4840. Figure 3-6)

Table 4-6 Seismic Event BIN Frequencies

	IE Freq.	CCDP	CDF
EQK-BIN-1	2.84E-04	2.55E-05	7.26E-09
EQK-BIN-2	1.26E-05	3.86E-02	4.86E-07
EQK-BIN-3	7.21E-06	6.13E-01	4.42E-06
Sum =	**3.04E-04**		**4.91E-06**

Table 4-7 Seismic Event Sequence Frequencies

Event tree	Sequence	CDF	Cutsets	End State	
EQK-BIN-3	5	3.80E-06	1	CD-EQK	Direct CD
EQK-BIN-3	3-11	5.37E-07	3	CD-EQK	SLOCA
EQK-BIN-2	5	4.47E-07	1	CD-EQK	Direct CD
EQK-BIN-3	4-3	3.33E-08	2	CD-EQK	LLOCA
EQK-BIN-2	3-11	2.65E-08	2	CD-EQK	SLOCA
EQK-BIN-3	2-17	2.48E-08	32	CD-EQK	LOOP
EQK-BIN-2	2-17	7.17E-09	29	CD-EQK	LOOP
EQK-BIN-3	3-13	5.37E-09	1	CD-EQK	SLOCA
EQK-BIN-3	3-24	4.92E-09	4	CD-EQK	SLOCA
EQK-BIN-3	3-03	3.37E-09	56	CD-EQK	SLOCA
EQK-BIN-1	2-18-03	3.04E-09	32	CD-EQK	LOOP
EQK-BIN-2	2-18-03	2.77E-09	32	CD-EQK	LOOP
EQK-BIN-3	2-19-13	2.01E-09	6	CD-EQK	LOOP
EQK-BIN-3	2-19-04	1.68E-09	6	CD-EQK	LOOP
EQK-BIN-1	2-17	1.67E-09	18	CD-EQK	LOOP
EQK-BIN-1	2-18-06	1.51E-09	26	CD-EQK	LOOP
EQK-BIN-2	2-18-06	1.38E-09	26	CD-EQK	LOOP
EQK-BIN-3	2-18-03	8.83E-10	24	CD-EQK	LOOP
EQK-BIN-1	5	7.87E-10	1	CD-EQK	Direct CD
EQK-BIN-2	3-03	6.15E-10	34	CD-EQK	SLOCA
EQK-BIN-3	3-12	5.37E-10	1	CD-EQK	SLOCA
EQK-BIN-3	2-19-20	4.51E-10	5	CD-EQK	LOOP
EQK-BIN-3	2-18-06	4.38E-10	20	CD-EQK	LOOP
EQK-BIN-2	4-3	3.48E-10	1	CD-EQK	LLOCA
EQK-BIN-3	4-2	3.28E-10	7	CD-EQK	LLOCA
EQK-BIN-3	2-19-09	2.65E-10	1	CD-EQK	LOOP
EQK-BIN-2	3-13	2.65E-10	1	CD-EQK	SLOCA
EQK-BIN-3	2-19-19	2.24E-10	23	CD-EQK	LOOP
EQK-BIN-3	2-19-18	2.20E-10	3	CD-EQK	LOOP
EQK-BIN-1	2-18-45	1.80E-10	32	CD-EQK	LOOP
EQK-BIN-2	2-18-45	1.64E-10	31	CD-EQK	LOOP
EQK-BIN-3	3-23	1.58E-10	14	CD-EQK	SLOCA
EQK-BIN-2	3-24	5.40E-11	1	CD-EQK	SLOCA
EQK-BIN-3	2-12	5.03E-11	8	CD-EQK	LOOP

Event tree	Sequence	CDF	Cutsets	End State	
EQK-BIN-3	2-02-05	4.70E-11	10	CD-EQK	LOOP
EQK-BIN-2	4-2	4.46E-11	1	CD-EQK	LLOCA
EQK-BIN-3	3-07	4.44E-11	1	CD-EQK	SLOCA
EQK-BIN-3	2-18-45	4.01E-11	12	CD-EQK	LOOP
EQK-BIN-1	2-18-09	3.28E-11	7	CD-EQK	LOOP
EQK-BIN-2	2-18-09	3.00E-11	7	CD-EQK	LOOP
EQK-BIN-2	3-07	2.94E-11	1	CD-EQK	SLOCA
EQK-BIN-2	3-12	2.65E-11	1	CD-EQK	SLOCA
EQK-BIN-2	2-19-20	2.33E-11	5	CD-EQK	LOOP
EQK-BIN-1	2-18-12	2.32E-11	7	CD-EQK	LOOP
EQK-BIN-2	2-18-12	2.12E-11	7	CD-EQK	LOOP
EQK-BIN-1	2-18-42	2.08E-11	8	CD-EQK	LOOP
EQK-BIN-2	2-18-42	1.90E-11	8	CD-EQK	LOOP
EQK-BIN-2	2-19-09	1.37E-11	1	CD-EQK	LOOP
EQK-BIN-2	3-23	9.83E-12	6	CD-EQK	SLOCA
EQK-BIN-2	2-12	9.76E-12	4	CD-EQK	LOOP
EQK-BIN-3	2-18-09	8.14E-12	4	CD-EQK	LOOP
EQK-BIN-2	2-02-05	7.90E-12	4	CD-EQK	LOOP
EQK-BIN-2	2-19-04	6.42E-12	2	CD-EQK	LOOP
EQK-BIN-2	2-19-13	6.42E-12	4	CD-EQK	LOOP
EQK-BIN-3	2-18-12	4.21E-12	2	CD-EQK	LOOP
EQK-BIN-2	2-19-18	2.74E-12	1	CD-EQK	LOOP
EQK-BIN-3	2-18-42	2.52E-12	2	CD-EQK	LOOP
EQK-BIN-3	3-05	1.71E-12	1	CD-EQK	SLOCA
EQK-BIN-2	3-05	1.13E-12	1	CD-EQK	SLOCA
	TOTALS	**4.91E-06**	**591**		

Table 4-8 Seismic Event CDF Cutsets

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
1	82.97	3.80E-6	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			CD-EQ3	DIRECT FUEL DAMAGE EVENTS	5.270E-01
2	11.73	5.37E-7	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			SLOCA-EQ3	SMALL LOCA EVENT	2.500E-01
3	9.75	4.47E-7	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			CD-EQ2	DIRECT FUEL DAMAGE EVENTS	3.550E-02
4	0.73	3.33E-8	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LLOCA-EQ3	LARGE LOCA EVENT	1.550E-02
5	0.58	2.65E-8	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			EQ-BIN-2-SWS-FA	Contribution of Seismic BIN-2 to SWS Failure	4.680E-02
			SLOCA-EQ2	SMALL LOCA EVENT	4.500E-02
6	0.20	8.69E-9	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			AFW-TDP-FS-1C	AFW TDP 1C FAILS TO START	6.000E-03
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
7	0.16	7.25E-9	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			AFW-TDP-TM-1C	AFW TDP 1C UNAVAILABLE DUE TO TEST AND MAINTENANCE	5.000E-03

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
8	0.14	6.00E-9	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			AFW-TDP-FR-1C	AFW TDP 1C FAILS TO RUN	4.141E-03
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
9	0.12	5.37E-9	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			RCS-XHE-XM-CDOWN1	OPERATOR FAILS TO INITIATE RAPID COOLDOWN	1.000E-02
			SLOCA-EQ3	SMALL LOCA EVENT	2.500E-01
10	0.11	4.92E-9	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-RPS-FA	Contribution of Seismic Event BIN-3 to RPS Failure	5.770E-03
			SLOCA-EQ3	SMALL LOCA EVENT	2.500E-01
11	0.07	3.07E-9	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-RHR-FA	Contribution of Seismic Event BIN-3 to RHR Failure	5.990E-01
			LPR-XHE-XM	OPERATOR FAILS TO INITIATE LPR SYSTEM	6.000E-03
			SLOCA-EQ3	SMALL LOCA EVENT	2.500E-01

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
12	0.05	2.02E-9	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			AFW-TDP-FS-1C	AFW TDP 1C FAILS TO START	6.000E-03
			EQ-BIN-2-SWS-FA	Contribution of Seismic BIN-2 to SWS Failure	4.680E-02
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
13	0.04	1.68E-9	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			AFW-TDP-TM-1C	AFW TDP 1C UNAVAILABLE DUE TO TEST AND MAINTENANCE	5.000E-03
			EQ-BIN-2-SWS-FA	Contribution of Seismic BIN-2 to SWS Failure	4.680E-02
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
14	0.04	1.45E-9	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			AFW-MOV-CC-102	AFW TDP 1C MAIN STEAM VALVE 102 FAILS TO OPEN	1.000E-03
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
15	0.04	1.40E-9	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			AFW-TDP-FR-1C	AFW TDP 1C FAILS TO RUN	4.141E-03
			EQ-BIN-2-SWS-FA	Contribution of Seismic BIN-2 to SWS Failure	4.680E-02
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
16	0.03	9.23E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.842E-04
			EPS-DGN-CF-RUN	COMMON CAUSE FAILURE OF DIESEL GENERATORS TO RUN	5.865E-04
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01
			LOOP-EQ1	LOSS OF OFFSITE POWER	2.770E-02

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01
17	0.02	8.44E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			EPS-DGN-CF-RUN	COMMON CAUSE FAILURE OF DIESEL GENERATORS TO RUN	5.865E-04
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01
18	0.02	8.36E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-RPS-FA	Contribution of Seismic Event BIN-3 to RPS Failure	5.770E-03
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			PPR-SRV-OO-SRV3BLIQ	SAFETY RELIEF VALVE 3B FAILS TO RECLOSE AFTER PASSING WATER	1.000E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
19	0.02	8.36E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-RPS-FA	Contribution of Seismic Event BIN-3 to RPS Failure	5.770E-03
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			PPR-SRV-OO-SRV3ALIQ	SAFETY RELIEF VALVE 3A FAILS TO RECLOSE AFTER PASSING WATER	1.000E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
20	0.02	7.87E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.842E-04
			CD-EQ1	DIRECT FUEL DAMAGE EVENTS	2.770E-06
21	0.02	7.87E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.842E-04
			AFW-CKV-CC-301	CONDENSATE STORAGE TANK DISCHARGE CHECK VALVE FAILS	1.000E-04
			LOOP-EQ1	LOSS OF OFFSITE POWER	2.770E-02
22	0.02	7.87E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.842E-04
			AFW-XHE-XA-SUCT	OPERATOR FAILS TO ALIGN SWS/XTIE RMST TO AFW SYSTEM	1.000E-04
			LOOP-EQ1	LOSS OF OFFSITE POWER	2.770E-02
23	0.02	7.20E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			AFW-CKV-CC-301	CONDENSATE STORAGE TANK DISCHARGE CHECK VALVE FAILS	1.000E-04
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
24	0.02	7.20E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			AFW-XHE-XA-SUCT	OPERATOR FAILS TO ALIGN SWS/XTIE RMST TO AFW SYSTEM	1.000E-04
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
25	0.02	7.06E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.842E-04
			EPS-DGN-FR-1A	DIESEL GENERATOR 1A FAILS TO RUN	2.117E-02
			EPS-DGN-FR-1B	DIESEL GENERATOR 1B FAILS TO RUN	2.117E-02
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01
			LOOP-EQ1	LOSS OF OFFSITE POWER	2.770E-02
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
26	0.02	6.45E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			EPS-DGN-FR-1A	DIESEL GENERATOR 1A FAILS TO RUN	2.117E-02
			EPS-DGN-FR-1B	DIESEL GENERATOR 1B FAILS TO RUN	2.117E-02
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01
27	0.02	5.80E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			AFW-PMP-FR-TD1C	AFW TURBINE-DRIVEN 1C PUMP UNIT ONLY FAILS TO RUN	4.000E-04
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
28	0.02	5.37E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			RCS-XHE-XM-RCSDEP	OPERATOR FAILS TO DEPRESSURIZE THE RCS	1.000E-03
			SLOCA-EQ3	SMALL LOCA EVENT	2.500E-01
29	0.02	4.93E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			ED-BIN-2-RHR-FA	Contribution of Seismic Event BIN-2 to RHR Failure	1.450E-01
			LPR-XHE-XM	OPERATOR FAILS TO INITIATE LPR SYSTEM	6.000E-03
			SLOCA-EQ2	SMALL LOCA EVENT	4.500E-02
30	0.02	4.62E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.842E-04

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
			EPS-DGN-CF-RUN	COMMON CAUSE FAILURE OF DIESEL GENERATORS TO RUN	5.865E-04
			EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS	5.000E-01
			LOOP-EQ1	LOSS OF OFFSITE POWER	2.770E-02
			RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	2.000E-01
31	0.01	4.22E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			EPS-DGN-CF-RUN	COMMON CAUSE FAILURE OF DIESEL GENERATORS TO RUN	5.865E-04
			EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS	5.000E-01
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
			RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	2.000E-01
32	0.01	4.18E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-RPS-FA	Contribution of Seismic Event BIN-3 to RPS Failure	5.770E-03
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			PPR-SRV-OO-SRV3ALIQ	SAFETY RELIEF VALVE 3A FAILS TO RECLOSE AFTER PASSING WATER	1.000E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
			SWS-TRAINA-ALIGNED	SW TRAIN A ALIGNED TO TURBINE BLDG	5.000E-01
33	0.01	4.18E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
			EQ-BIN-3-RPS-FA	Contribution of Seismic Event BIN-3 to RPS Failure	5.770E-03
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			PPR-SRV-OO-SRV3ALIQ	SAFETY RELIEF VALVE 3A FAILS TO RECLOSE AFTER PASSING WATER	1.000E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
			SWS-TRAINB-ALIGNED	SW TRAIN B ALIGNED TO TURBINE BLDG	5.000E-01
34	0.01	4.18E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-RPS-FA	Contribution of Seismic Event BIN-3 to RPS Failure	5.770E-03
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			PPR-SRV-OO-SRV3BLIQ	SAFETY RELIEF VALVE 3B FAILS TO RECLOSE AFTER PASSING WATER	1.000E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
			SWS-TRAINA-ALIGNED	SW TRAIN A ALIGNED TO TURBINE BLDG	5.000E-01
35	0.01	4.18E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EQ-BIN-3-RPS-FA	Contribution of Seismic Event BIN-3 to RPS Failure	5.770E-03
			EQ-BIN-3-SWS-FA	Contribution of Seismic BIN-3 to SWS Failure	6.300E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			PPR-SRV-OO-SRV3BLIQ	SAFETY RELIEF VALVE 3B FAILS TO RECLOSE AFTER PASSING WATER	1.000E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
			SWS-TRAINB-ALIGNED	SW TRAIN B ALIGNED TO TURBINE BLDG	5.000E-01
36	0.01	3.53E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.842E-04
			EPS-DGN-FR-1A	DIESEL GENERATOR 1A FAILS TO RUN	2.117E-02
			EPS-DGN-FR-1B	DIESEL GENERATOR 1B FAILS TO RUN	2.117E-02
			EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS	5.000E-01
			LOOP-EQ1	LOSS OF OFFSITE POWER	2.770E-02
			RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	2.000E-01
37	0.01	3.48E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			EQ-BIN-2-SWS-FA	Contribution of Seismic BIN-2 to SWS Failure	4.680E-02
			LLOCA-EQ2	LARGE LOCA EVENT	5.910E-04
38	0.01	3.37E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			AFW-MOV-CC-102	AFW TDP 1C MAIN STEAM VALVE 102 FAILS TO OPEN	1.000E-03
			EQ-BIN-2-SWS-FA	Contribution of Seismic BIN-2 to SWS Failure	4.680E-02
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
39	0.01	3.23E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			EPS-DGN-FR-1A	DIESEL GENERATOR 1A FAILS TO RUN	2.117E-02
			EPS-DGN-FR-1B	DIESEL GENERATOR 1B FAILS TO RUN	2.117E-02
			EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS	5.000E-01
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
			RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	2.000E-01
40	0.01	3.17E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			LLOCA-EQ3	LARGE LOCA EVENT	1.550E-02
			LPR-XHE-XM	OPERATOR FAILS TO INITIATE LPR SYSTEM	6.000E-03
41	0.01	3.00E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.842E-04
			EPS-DGN-FR-1A	DIESEL GENERATOR 1A FAILS TO RUN	2.117E-02
			EPS-DGN-TM-1B	DIESEL GENERATOR 1B UNAVAILABLE DUE TO TEST AND MAINTENANCE	9.000E-03
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01
			LOOP-EQ1	LOSS OF OFFSITE POWER	2.770E-02
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01
42	0.01	3.00E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05 - 0.3 g)	2.842E-04
			EPS-DGN-FR-1B	DIESEL GENERATOR 1B FAILS TO RUN	2.117E-02
			EPS-DGN-TM-1A	DIESEL GENERATOR 1A UNAVAILABLE DUE TO TEST AND MAINTENANCE	9.000E-03
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01
			LOOP-EQ1	LOSS OF OFFSITE POWER	2.770E-02
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01
43	0.01	2.74E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			EPS-DGN-FR-1B	DIESEL GENERATOR 1B FAILS TO RUN	2.117E-02
			EPS-DGN-TM-1A	DIESEL GENERATOR 1A UNAVAILABLE DUE TO TEST AND MAINTENANCE	9.000E-03
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01
44	0.01	2.74E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			EPS-DGN-FR-1A	DIESEL GENERATOR 1A FAILS TO RUN	2.117E-02
			EPS-DGN-TM-1B	DIESEL GENERATOR 1B UNAVAILABLE DUE TO TEST AND MAINTENANCE	9.000E-03
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01
45	0.01	2.70E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EPS-DGN-CF-RUN	COMMON CAUSE FAILURE OF DIESEL GENERATORS TO RUN	5.865E-04
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
46	0.01	2.65E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			CVC-XHE-XM-BOR	OPERATOR FAILS TO INITIATE EMERGENCY BORATION	2.000E-02
			EQ-BIN-3-RPS-FA	Contribution of Seismic Event BIN-3 to RPS Failure	5.770E-03
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
47	0.01	2.65E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3 - 0.5 g)	1.258E-05
			EQ-BIN-2-SWS-FA	Contribution of Seismic BIN-2 to SWS Failure	4.680E-02
			RCS-XHE-XM-CDOWN1	OPERATOR FAILS TO INITIATE RAPID COOLDOWN	1.000E-02
			SLOCA-EQ2	SMALL LOCA EVENT	4.500E-02
48	0.01	2.30E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			AFW-XHE-XA-SUCT	OPERATOR FAILS TO ALIGN SWS/XTIE RMST TO AFW SYSTEM	1.000E-04
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
49	0.01	2.30E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			AFW-CKV-CC-301	CONDENSATE STORAGE TANK DISCHARGE CHECK VALVE FAILS	1.000E-04
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
50	0.01	2.06E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5 g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01
			EPS-DGN-FR-1A	DIESEL GENERATOR 1A FAILS TO RUN	2.117E-02
			EPS-DGN-FR-1B	DIESEL GENERATOR 1B FAILS TO RUN	2.117E-02
			EPS-XHE-XL-NR08H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 8 HOURS	2.500E-01
			LOOP-EQ3	LOSS OF OFFSITE POWER	8.990E-01

Cut No.	% Cut Set	Frequency	Basic Event	Description	Event Prob.
			/RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	8.000E-01
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01

Appendix 4A. Generic Seismic Hazard Vectors

The generic hazard vectors for 69 sites east of the Rocky Mountains are taken from Ref. 4-2.

The hazard vectors for the remaining 4 sites are taken from their IPEEE submittals to the NRC.

Table 4A-1 provides the seismic hazard vectors for the 72 SPAR plants.

G values are in term of peak ground acceleration (pga).

Table 4A-1 Seismic Hazard Vectors for the 72 SPAR Plants

mean frequency of exceedance (per year)									
	1/2	3/4	5	6/7	8	9	10	11	12
	ANO	Beaver Valley	Braidwood	Browns Ferry	Brunswick	Byron	Callaway	Calvert Cliffs	Catawba
g value	mean f per year	mean f per year	mean f per year	mean f per year	mean f per year	mean f per year	mean f per year	mean f per year	mean f per year
0.05	.1273E-02	.8778E-03	.4297E-03	.9121E-03	.1527E-02	.5091E-03	.1083E-02	.7674E-03	.1199E-02
0.08	.6698E-03	.4919E-03	.2313E-03	.4560E-03	.8013E-03	.2864E-03	.4763E-03	.4321E-03	.6295E-03
0.15	.2016E-03	.1686E-03	.7032E-04	.1247E-03	.2428E-03	.9093E-04	.9878E-04	.1459E-03	.1840E-03
0.25	.7274E-04	.7056E-04	.2448E-04	.4190E-04	.9380E-04	.3227E-04	.2739E-04	.5891E-04	.6334E-04
0.30	.4858E-04	.5056E-04	.1595E-04	.2724E-04	.6580E-04	.2116E-04	.1684E-04	.4141E-04	.4130E-04
0.40	.2442E-04	.2901E-04	.7622E-05	.1309E-04	.3692E-04	.1021E-04	.7532E-05	.2292E-04	.1987E-04
0.50	.1369E-04	.1832E-04	.4067E-05	.7065E-05	.2316E-04	.5489E-05	.3900E-05	.1402E-04	.1072E-04
0.65	.6568E-05	.1027E-04	.1825E-05	.3231E-05	.1306E-04	.2488E-05	.1723E-05	.7565E-05	.4910E-05
0.80	.3522E-05	.6292E-05	.9232E-06	.1663E-05	.8121E-05	.1269E-05	.8721E-06	.4498E-05	.2546E-05
1.00	.1729E-05	.3589E-05	.4239E-06	.7792E-06	.4751E-05	.5885E-06	.4048E-06	.2490E-05	.1215E-05
	13	14	15	16	17	18	19	20	21
	Clinton	Columbia	Comanche Peak	Cook	Cooper	Crystal River	Davis Besse	Diablo Canyon	Dresden
g value	mean f per year	mean f per year	mean f per year	mean f per year	mean f per year	mean f per year	mean f per year		mean f per year
0.05	.1547E-02		.1410E-03	.5010E-03	.1155E-02	.1482E-03	.1070E-02		.4576E-03
0.08	.8083E-03		.6790E-04	.2729E-03	.7283E-03	.8403E-04	.5745E-03		.2539E-03
0.15	.2457E-03		.1880E-04	.8900E-04	.2924E-03	.2765E-04	.1631E-03		.8120E-04
0.25	.9422E-04		.6420E-05	.3578E-04	.1335E-03	.1039E-04	.5326E-04		.2927E-04
0.30	.6543E-04		.4190E-05	.2528E-04	.9828E-04	.7035E-05	.3413E-04		.1929E-04
0.40	.3573E-04		.2020E-05	.1421E-04	.5867E-04	.3625E-05	.1604E-04		.9355E-05
0.50	.2171E-04		.1100E-05	.8843E-05	.3813E-04	.2083E-05	.8537E-05		.5034E-05
0.65	.1165E-04		.5080E-06	.4890E-05	.2211E-04	.1039E-05	.3868E-05		.2272E-05
0.80	.6894E-05		.2660E-06	.2969E-05	.1392E-04	.5796E-06	.1990E-05		.1150E-05
1.00	.3794E-05		.1280E-06	.1681E-05	.8187E-05	.2992E-06	.9390E-06		.5266E-06

	22	23	24	25	26	27	28	29	30
	Duane Arnold	**Farley**	**Fermi**	**Fitzpatrick**	**Fort Calhoun**	**Ginna**	**Grand Gulf**	**Hatch**	**Hope Creek**
g value	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**
0.05	.1548E-03	.1995E-03	.6010E-03	.7335E-03	.8778E-03	.8483E-03	.3306E-03	.6133E-03	.9721E-03
0.08	.8105E-04	.1092E-03	.2980E-03	.3537E-03	.5580E-03	.4657E-03	.1765E-03	.3186E-03	.5512E-03
0.15	.2378E-04	.3463E-04	.7740E-04	.8831E-04	.2306E-03	.1457E-03	.5513E-04	.9661E-04	.1836E-03
0.25	.8208E-05	.1268E-04	.2493E-04	.2764E-04	.1080E-03	.5283E-04	.2103E-04	.3713E-04	.7227E-04
0.30	.5359E-05	.8459E-05	.1603E-04	.1761E-04	.8024E-04	.3516E-04	.1448E-04	.2583E-04	.5028E-04
0.40	.2584E-05	.4230E-05	.7622E-05	.8284E-05	.4854E-04	.1742E-04	.7745E-05	.1416E-04	.2735E-04
0.50	.1397E-05	.2357E-05	.4108E-05	.4444E-05	.3187E-04	.9558E-05	.4613E-05	.8644E-05	.1651E-04
0.65	.6421E-06	.1124E-05	.1894E-05	.2048E-05	.1870E-04	.4403E-05	.2409E-05	.4680E-05	.8770E-05
0.80	.3338E-06	.6009E-06	.9902E-06	.1075E-05	.1187E-04	.2255E-05	.1395E-05	.2801E-05	.5156E-05
1.00	.1594E-06	.2947E-06	.4769E-06	.5206E-06	.7043E-05	.1041E-05	.7485E-06	.1567E-05	.2826E-05

	31/32	33	34	35	36	37/38	39	40/41	42
	Indian Point	**Kewaunee**	**LaSalle**	**Limerick**	**McGuire**	**Millstone 2 & 3**	**Monticello**	**Nine Mile Point 1 & 2**	**North Anna 1 & 2**
g value	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**
0.05	.1152E-02	.3040E-03	.8251E-03	.1220E-02	.1084E-02	.9965E-03	.3562E-03	.7302E-03	.1153E-02
0.08	.6552E-03	.1777E-03	.4633E-03	.6990E-03	.5582E-03	.5635E-03	.2131E-03	.3525E-03	.6606E-03
0.15	.2123E-03	.6422E-04	.1616E-03	.2290E-03	.1568E-03	.1823E-03	.7981E-04	.8831E-04	.2139E-03
0.25	.7736E-04	.2748E-04	.6797E-04	.8350E-04	.5192E-04	.6635E-04	.3511E-04	.2772E-04	.7505E-04
0.30	.5148E-04	.1979E-04	.4859E-04	.5550E-04	.3329E-04	.4410E-04	.2556E-04	.1769E-04	.4871E-04
0.40	.2562E-04	.1141E-04	.2765E-04	.2750E-04	.1553E-04	.2189E-04	.1500E-04	.8339E-05	.2301E-04
0.50	.1421E-04	.7212E-05	.1728E-04	.1520E-04	.8136E-05	.1211E-04	.9622E-05	.4483E-05	.1213E-04
0.65	.6738E-05	.4043E-05	.9548E-05	.7100E-05	.3580E-05	.5713E-05	.5493E-05	.2073E-05	.5362E-05
0.80	.3583E-05	.2474E-05	.5773E-05	.3730E-05	.1785E-05	.3025E-05	.3411E-05	.1090E-05	.2675E-05
1.00	.1749E-05	.1409E-05	.3244E-05	.1790E-05	.8089E-06	.1469E-05	.1976E-05	.5298E-06	.1209E-05

	43	44	45	46	47	48	49	50	51
	Oconee 1, 2 & 3	Oyster Creek	Palisades	Palo Verde 1, 2 & 3	Peach Bottom	Perry	Pilgrim	Point Beach	Prairie Island
g value	mean f per year	mean f per year	mean f per year		mean f per year	mean f per year	mean f per year	mean f per year	mean f per year
0.05	.1280E-02	.8528E-03	.3924E-03	[redacted]	.1058E-02	.4477E-03	.2814E-02	.3125E-03	.3154E-03
0.08	.6937E-03	.4839E-03	.2109E-03	[redacted]	.6043E-03	.2466E-03	.1777E-02	.1825E-03	.1907E-03
0.15	.2104E-03	.1626E-03	.6870E-04	[redacted]	.1982E-03	.7663E-04	.7154E-03	.6573E-04	.7272E-04
0.25	.7353E-04	.6463E-04	.2824E-04	[redacted]	.7229E-04	.2710E-04	.3272E-03	.2804E-04	.3233E-04
0.30	.4815E-04	.4510E-04	.2019E-04	[redacted]	.4793E-04	.1780E-04	.2410E-03	.2017E-04	.2361E-04
0.40	.2328E-04	.2461E-04	.1159E-04	[redacted]	.2357E-04	.8628E-05	.1441E-03	.1160E-04	.1394E-04
0.50	.1257E-04	.1486E-04	.7331E-05	[redacted]	.1288E-04	.4671E-05	.9379E-04	.7319E-05	.8998E-05
0.65	.5725E-05	.7875E-05	.4129E-05	[redacted]	.5953E-05	.2142E-05	.5446E-04	.4089E-05	.5185E-05
0.80	.2930E-05	.4606E-05	.2540E-05	[redacted]	.3085E-05	.1107E-05	.3430E-04	.2493E-05	.3251E-05
1.00	.1360E-05	.2499E-05	.1455E-05	[redacted]	.1454E-05	.5224E-06	.2016E-04	.1414E-05	.1905E-05
	52	53	54	55/56	57	58	59	60	61
	Quad Cities 1 & 2	River Bend	Robinson	Saint Lucie 1 & 2	Salem 1 & 2	San Onofre 2 & 3	Seabrook	Sequoyah	Shearon Harris
g value	mean f per year	mean f per year	mean f per year	mean f per year	mean f per year		mean f per year	mean f per year	mean f per year
0.05	.3658E-03	.1965E-03	.2717E-02	.1470E-03	.9589E-03	[redacted]	2.34E-03	1.33E-03	5.85E-04
0.08	.1948E-03	.1072E-03	.1565E-02	.8565E-04	.5429E-03	[redacted]	1.37E-03	7.54E-04	3.06E-04
0.15	.5727E-04	.3426E-04	.5469E-03	.3037E-04	.1805E-03	[redacted]	4.62E-04	2.44E-04	9.05E-05
0.25	.1965E-04	.1322E-04	.2256E-03	.1263E-04	.7102E-04	[redacted]	1.71E-04	8.80E-05	3.08E-05
0.30	.1280E-04	.9138E-05	.1600E-03	.8974E-05	.4940E-04	[redacted]	1.15E-04	5.82E-05	1.98E-05
0.40	.6139E-05	.4913E-05	.8990E-04	.5047E-05	.2686E-04	[redacted]	5.74E-05	2.85E-05	9.22E-06
0.50	.3295E-05	.2932E-05	.5574E-04	.3125E-05	.1620E-04	[redacted]	3.19E-05	1.55E-05	4.81E-06
0.65	.1492E-05	.1531E-05	.3062E-04	.1711E-05	.8599E-05	[redacted]	1.50E-05	7.14E-06	2.09E-06
0.80	.7603E-06	.8842E-06	.1849E-04	.1030E-05	.5051E-05	[redacted]	7.92E-06	3.68E-06	1.03E-06
1.00	.3519E-06	.4737E-06	.1043E-04	.5790E-06	.2764E-05	[redacted]	3.80E-06	1.72E-06	4.60E-07

	62	63	64	65	66	67	68	69	70
	South Texas 1 & 2	**Surry 1 & 2**	**Susquehanna 1 & 2**	**Three Mile Island**	**Turkey Point 3 & 4**	**V.C. Summer**	**Vermont Yankee**	**Vogtle**	**Waterford**
g value	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**	**mean f per year**
0.05	1.628E-04	6.033E-04	8.46E-04	.1108E-02	.1227E-03	.1833E-02	.1293E-02	.2500E-02	.2863E-03
0.08	9.408E-05	3.363E-04	4.68E-04	.6339E-03	.7052E-04	.9731E-03	.6738E-03	.1356E-02	.1655E-03
0.15	3.256E-05	1.104E-04	1.47E-04	.2076E-03	.2361E-04	.2842E-03	.1963E-03	.4152E-03	.5704E-04
0.25	1.312E-05	4.322E-05	5.29E-05	.7622E-04	.8904E-05	.9814E-04	.6913E-04	.1546E-03	.2291E-04
0.30	9.179E-06	2.998E-05	3.49E-05	.5082E-04	.6013E-05	.6448E-04	.4583E-04	.1060E-03	.1604E-04
0.40	5.024E-06	1.621E-05	1.71E-05	.2532E-04	.3074E-05	.3170E-04	.2286E-04	.5677E-04	.8804E-05
0.50	3.038E-06	9.713E-06	9.28E-06	.1404E-04	.1750E-05	.1755E-04	.1279E-05	.3415E-04	.5350E-05
0.65	1.613E-06	5.097E-06	4.26E-06	.6627E-05	.8616E-06	.8392E-05	.6148E-05	.1830E-04	.2865E-05
0.80	9.453E-07	2.955E-06	2.19E-06	.3504E-05	.4751E-06	.4535E-05	.3315E-05	.1094E-04	.1694E-05
1.00	5.144E-07	1.586E-06	1.02E-06	.1693E-05	.2425E-06	.2274E-05	.1644E-05	.6181E-05	.9316E-06

	71	72
	Watts Bar	**Wolf Creek**
g value	**mean f per year**	**mean f per year**
0.05	.1258E-02	.3290E-03
0.08	.7128E-03	.1664E-03
0.15	.2301E-03	.4581E-04
0.25	.8298E-04	.1526E-04
0.30	.5483E-04	.9857E-05
0.40	.2686E-04	.4666E-05
0.50	.1465E-04	.2480E-05
0.65	.6754E-05	.1110E-05
0.80	.3488E-05	.5607E-06
1.00	.1634E-05	.2569E-06

For 69 NPP sites east of Rocky Mountains, NUREG-1488 provides Seismic IEV frequencies.

For the four sites West of Rocky mountains, this information is obtained from IPEEE studies and is given below.

14 Columbia		20 Diablo Canyon		46 Palo Verde 1, 2 & 3		58 San Onofre 2 & 3	
g value	mean f per year	g value	mean f per year	g value	mean f per year	g value	mean f per year
0.05	1.30E-03	0.20	1.85E-02	1.00E-02	3.00E-02	0.20	5.20E-03
0.10	1.30E-03	0.50	7.44E-03	2.00E-02	5.70E-03	0.30	2.00E-03
0.20	3.00E-04	0.80	3.56E-03	5.00E-02	9.10E-04	0.40	8.00E-04
0.30	1.10E-04	1.00	2.19E-03	7.00E-02	5.30E-04	0.50	3.38E-04
0.40	5.00E-05	1.20	1.35E-03	1.00E-01	3.00E-04	0.60	2.10E-04
0.50	2.50E-05	1.50	6.26E-04	1.50E-01	1.50E-04	0.70	1.00E-04
0.60	1.30E-05	2.00	1.61E-04	2.00E-01	7.90E-05	0.80	4.80E-05
0.70	7.80E-06	2.50	3.73E-05	3.00E-01	2.20E-05	0.90	2.60E-05
0.80	4.60E-06	3.00	7.89E-06	5.00E-01	1.10E-06	1.00	1.30E-05
0.90	3.00E-06	4.00	2.42E-07	1.00E+00	9.30E-10	2.00	7.80E-08
1.00	1.80E-06						

Appendix 4B. Generic SSC Seismic Fragilities

Generic SSC seismic fragilities and the failure probabilities for SSCs in each seismic bin as derived from these fragilities are given in Table 4B-1. In the absence of plant-specific fragility information for a SSC, the values from this table can be used.

A seismic fragility library is being constructed from plant-specific SSC fragilities available in recent sources, such as IPEEEs. The currently available seismic fragility information in this library is placed in ADAMS as an EXCEL file with accession number ML71220070.

Table 4B-1 Generic SSC Seismic Fragilities

Component	Median Capacity, g	beta-r	beta-u	HCLPF Capacity, g	Failure Mode	Source	Failure probability Pf at X g		
							0.122	0.387	0.707
Offsite power	0.3	0.30	0.45	0.10	Failure of ceramics	1	4.81E-02	6.81E-01	9.44E-01
Electrical equipment - Function during seismic event	1.0	0.30	0.35	0.34	Chatter functional failure	1	2.52E-06	1.97E-02	2.26E-01
Large flat-bottom storage tanks	1.1	0.30	0.35	0.37	Buckling or wall failure	1	9.20E-07	1.17E-02	1.69E-01
Battery chargers	1.6	0.30	0.35	0.54	Functional failure	1	1.18E-08	1.04E-03	3.82E-02
Inverters	1.6	0.30	0.35	0.54	Functional failure	1	1.18E-08	1.04E-03	3.82E-02
Cable trays	2.5	0.35	0.50	0.61	Support failure	1	3.75E-07	1.12E-03	1.93E-02
HVAC ducts	2.5	0.35	0.50	0.61	Support failure	1	3.75E-07	1.12E-03	1.93E-02
Heat exchangers and small tanks	1.9	0.30	0.35	0.65	Rupture	1	1.30E-09	2.79E-04	1.60E-02
Recirculation pumps	1.9	0.30	0.35	0.65	Support failure	1	1.30E-09	2.79E-04	1.60E-02
Transformers	1.9	0.30	0.35	0.65	Loss of function /structural failure	1	1.30E-09	2.79E-04	1.60E-02
Motor-driven pumps	2.0	0.30	0.35	0.68	Support failure	1	6.53E-10	1.83E-04	1.20E-02
Air handling units	2.5	0.30	0.40	0.75	Structural failure	1	7.73E-10	9.53E-05	5.77E-03
Pressurizer	2.5	0.30	0.40	0.75	Structural failure of support	1	7.73E-10	9.53E-05	5.77E-03
Control rod drive and hydraulic drive units	2.5	0.30	0.40	0.76	Functional failure	1	7.73E-10	9.53E-05	5.77E-03

Component	Median Capacity, g	beta-r	beta-u	HCLPF Capacity, g	Failure Mode	Source	Failure probability Pf at X g		
							0.122	0.387	0.707
Electrical equipment - Function after seismic event	2.5	0.30	0.40	0.77	Chatter functional failure	1	7.73E-10	9.53E-05	5.77E-03
Buried welded steel piping	2.0	0.25	0.30	0.80	Buckling	1	4.00E-13	1.30E-05	3.87E-03
Accumulators	2.5	0.30	0.35	0.85	Structural failure	1	2.87E-11	2.59E-05	3.07E-03
Turbine-driven pumps	2.5	0.30	0.35	0.85	Support failure	1	2.87E-11	2.59E-05	3.07E-03
Air-operated valves	3.8	0.35	0.50	0.93	Loss of function	1	8.82E-09	9.10E-05	2.93E-03
Motor-operated valves	3.8	0.35	0.50	0.93	loss of function	1	8.82E-09	9.10E-05	2.93E-03
Piping	3.8	0.35	0.50	0.93	Loss of support	1	8.82E-09	9.10E-05	2.93E-03
Safety relief, manual and check valves	3.8	0.35	0.50	0.93	Loss of function	1	8.82E-09	9.10E-05	2.93E-03
Diesel generator and support systems	3.1	0.30	0.35	1.06	Functional failure	1	1.13E-12	3.19E-06	6.72E-04
Switchgear and motor control centers	3.1	0.30	0.35	1.06	Functional failure	1	1.13E-12	3.19E-06	6.72E-04
Batteries and battery racks	3.8	0.30	0.35	1.30	Structural failure of supports	1	4.37E-14	3.61E-07	1.32E-04
Panelboards and instrumentation panel	3.8	0.30	0.35	1.30	Functional failure	1	4.37E-14	3.61E-07	1.32E-04
Containment, buildings	1.1	0.30	0.35	0.45	Structural failure	Kew	9.20E-07	1.17E-02	1.69E-01

Component	Median Capacity, g	beta-r	beta-u	HCLPF Capacity, g	Failure Mode	Source	Failure probability Pf at X g 0.122	0.387	0.707
Reactor internals and core assembly	1.8	0.30	0.40	0.55	Structural failure	1	3.67E-08	1.06E-03	3.08E-02
Reactor pressure vessel	2.0	0.30	0.35	0.68	Support failure	1	6.53E-10	1.83E-04	1.20E-02
Steam generators	2.5	0.30	0.40	0.75	Structural failure of support	1	7.73E-10	9.53E-05	5.77E-03
Reactor coolant pump	2.5	0.30	0.40	0.75	Structural failure of support	1	7.73E-10	9.53E-05	5.77E-03
Seismically induced small LOCA probability						4840	1.50E-05	4.50E-02	2.50E-01
Seismically induced medium LOCA probability						4840	1.00E-07	4.00E-03	4.00E-02
Sources:	1	NUREG/CR-6544, Table 6-1							
	KEW	Kewaunee SPAR-EE model							
	4840	Seismically-induced SLOCA and MLOCA probabilities are taken from NURE/CR-4840, Figure 3-6.							

Appendix 4C. Seismic Fragility / pga / HCLPF

The complete fragility description of any particular SSC includes a representation of both the probabilities of failure vs. pga and the uncertainty of the analyst in estimating those probabilities. ("Failure", in this context, refers to inability to perform the assigned safety function.)

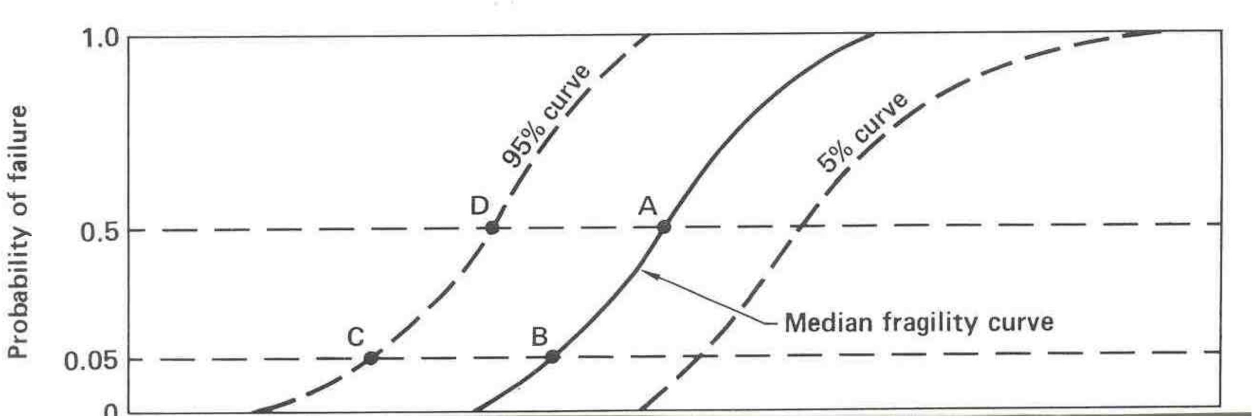
In the absence of variability and uncertainty, the capacity of an element could be defined by a single number, the precise pga at which the element would fail. Because of earthquake-to-earthquake variations in the dynamic response and capacity for the same nominal pga, one must recognize that the capacity can be represented only by a distribution -- specifically, a distribution of failure probability vs. pga. Further, because of incomplete technical knowledge (both theoretical and observational) about the probabilistic seismic behavior of elements and systems, it is necessary to describe the uncertainty in these fragility distributions.

Figure 4C-1 (Figure 2-1 of NUREG/CR-4334) presents one way of displaying such a full fragility description. The curves on this figure are very stylized and do not represent any particular functional form. The solid curve in the middle represents a "best-estimate" curve, the "median fragility curve." Corresponding to an ordinate of 0.50 is the ("best estimate" of the) median capacity, A_m, Point A. The pga corresponding to Point B is the ("best estimate" of the) pga at which there is only a 5% probability of failure.

The dashed lines in Figure 4C-1 reflect the uncertainty in the analyst's estimation of the probability distribution -- the uncertainty in the pga value corresponding to a given probability of failure, or conversely, the uncertainty in the probability of failure corresponding to a given pga. For example, Point D corresponds to the 95% (lower) confidence estimate of the median capacity. Specifically, the analyst is 95% confident that the median capacity exceeds this pga level. Similarly, Point C represents the high (95%) confidence estimate of the pga at which there is only a small (5%) probability of failure.

In those situations in which full fragility descriptions have been developed (mainly in full-scope seismic PRA studies), we have chosen the HCLPF to be represented by Point C. It is important to realize that this choice is only a convention, because the HCLPF point should not connote such numerical precision.

In most PRA practice, it has been conventional to assume a particular model for the fragility description. This is the (double) lognormal, in which the fragility can be fully described by only three parameters: the "best estimate" of) median capacity (A_m); a randomness measure, β_R that measures the slope or spread of the median fragility curve; and an uncertainty measure, β_U that is a measure of the separations between the median curve and the 95% and 5% curves in Figure 4C-1. Under these circumstances, and assuming that the lognormal model exactly characterizes the fragility at issue, it can be shown that Point B is below the median point by a factor of exp ($-1.65\ \beta_R$). Also, Point D is below the median by a factor of exp ($-1.65\ \beta_U$), and Point C is below the median by exp [$-1.65(\beta_R + \beta_U)$].



Peak ground acceleration (logarithmic scale)

Figure 4C-1 Fragility Curves. Peak ground acceleration corresponding to Point A represents the median capacity. Peak ground acceleration corresponding to Point C represents the HCLPF capacity.

Source: NUREG/CR-4334 (1985), Figure 2-1

Composite variability (β_c):

The composite variability includes the aleatory (randomness) uncertainty (β_R) and the epistemic (modeling and data) uncertainty (β_U). The logarithmic standard deviation of composite variability, (β_c), is expressed as $(\beta_R{}^2 + \beta_U{}^2)^{1/2}$.

HCLPF capacity:

The high confidence of low probability of failure (HCLPF) capacity is a measure of seismic margin. In seismic PRA, this is defined as the earthquake motion level at which there is a high (95 percent) confidence of a low (at most 5 percent) probability of failure. Using the lognormal fragility model, the HCLPF capacity is expressed as A_m [exp(-1.65($\beta_R + \beta_U$)]. When the logarithmic standard deviation of composite variability β_c is used, the HCLPF capacity could be approximated as the ground motion level at which the composite probability of failure is at most 1 percent. In this case, HCLPF capacity is expressed as A_m[exp-2.33 β_c]. In deterministic SMAs, the HCLPF capacity is calculated using the Conservative Deterministic Failure Methodology method.

Peak ground acceleration (PGA):

Maximum value of acceleration displayed on an accelerogram; the largest ground acceleration produced by an earthquake at a site.

Source: ANSI/ANS-58.21-2007 American National Standard External-Events PRA Methodology

Peak Ground Acceleration (http://earthquake.usgs.gov/faq/meas.html#14)

Acceleration is the rate of change in velocity of the ground shaking (how much the velocity changes in a unit time), just as it is the rate of change in the velocity of your car when you step on the accelerator or put on the brakes. Velocity is the measurement of the speed of the ground motion. Displacement is the measurement of the actual changing location of the ground due to shaking. All three of the values can be measured continuously during an earthquake. The peak ground acceleration (PGA) is the largest acceleration recorded by a particular station during an earthquake.

Spectral Acceleration

(http://earthquake.usgs.gov/image_glossary/spectral_accel.html)

PGA (peak acceleration) is what is experienced by a particle on the ground. SA (spectral acceleration) is approximately what is experienced by a building, as modeled by a particle on a massless vertical rod having the same natural period of vibration as the building.

Appendix 4D. Correspondence between PGA and Richter Scale

There are two methods of measurement for describing earthquakes. The Richter Scale measures magnitude, or the energy released by an earthquake. The Modified Mercalli Scale measures intensity, or an earthquake's impact or effect as felt at a particular location.

This section provides some information for the correspondence among Richter scale, PGA and modified Mercelli intensity scales for seismic events. The relation between modified Mercelli scale and PGA is taken from a paper which is based on regression analysis of eight significant California earthquakes.

Although there are some empirical relationships, no exact correlations of intensity, magnitude, and acceleration with damage are possible since many factors contribute to seismic behavior and structural performance.

Table 4D-1 Modified Mercalli Intensity Scale versus PGA

Mercalli Intensity	Equivalent Richter Magnitude	Witness Observations	Intensity Peak Accel. (% g)
I	1.0 to 2.0	Felt by very few people; barely noticeable.	<0.17
II	2.0 to 3.0	Felt by a few people, especially on upper floors.	0.17-1.4
III	3.0 to 4.0	Noticeable indoors, especially on upper floors, but may not be recognized as an earthquake.	0.17-1.4
IV	4	Felt by many indoors, few outdoors. May feel like heavy truck passing by.	1.4-3.9
V	4.0 to 5.0	Felt by almost everyone, some people awakened. Small objects moved. trees and poles may shake.	3.9-9.2
VI	5.0 to 6.0	Felt by everyone. Difficult to stand. Some heavy furniture moved, some plaster falls. Chimneys may be slightly damaged.	9.2-18
VII	6	Slight to moderate damage in well built, ordinary structures. Considerable damage to poorly built structures. Some walls may fall.	18-34
VIII	6.0 to 7.0	Little damage in specially built structures. Considerable damage to ordinary buildings, severe damage to poorly built structures. Some walls collapse.	34-65
IX	7	Considerable damage to specially built structures, buildings shifted off foundations. Ground cracked noticeably. Wholesale destruction. Landslides.	65-124
X	7.0 to 8.0	Most masonry and frame structures and their foundations destroyed. Ground badly cracked. Landslides. Wholesale destruction.	>124
XI	8	Total damage. Few, if any, structures standing. Bridges destroyed. Wide cracks in ground. Waves seen on ground.	>124
XII	8.0 or greater	Total damage. Waves seen on ground. Objects thrown up into air.	>124

Table 4D-2 PGA vs. Richter and Modified Mercalli Scales

Peak Ground Acceleration (% g)	PGA (representative)	Equivalent Richter Magnitude	Mercalli Intensity Scale
<0.17			I
0.17-1.4			II –III
1.4-3.9			IV
3.9-9.2			V
9.2-18	0.15g	5.0 to 6.0	VI
18-34	0.30g	6	VII
34-65	0.50g	6.0 to 7.0	VIII
65-124	1.00g	7	IX
>124	1.25g	7.0 or greater	X+

Table gives the peak ground motion ranges that correspond to each unit Modified Mercalli intensity value according to regression of the observed peak ground motions and intensities for California earthquakes. Equivalent Richter scales are also included.

The Modified Mercalli Scale of Earthquake Intensity

In seismology a scale of seismic intensity is a way of measuring or rating the *effects* of an earthquake at different sites. The Modified Mercalli Intensity Scale is commonly used in the United States by seismologists seeking information on the severity of earthquake effects. Intensity ratings are expressed as Roman numerals between I at the low end and XII at the high end.

The Intensity Scale differs from the Richter Magnitude Scale in that the effects of any one earthquake vary greatly from place to place, so there may be many Intensity values (e.g., IV, VII) measured from one earthquake. Each earthquake, on the other hand, should have just one Magnitude, although the several methods of estimating it will yield slightly different values (e.g., 6.1, 6.3).

Ratings of earthquake effects are based on the following relatively subjective scale of descriptions:

Table 4D-3 Modified Mercalli Intensity Scale (from FEMA)

Mercalli Intensity	Description
I	People do not feel any Earth movement.
II	A few people might notice movement if they are at rest and/or on the upper floors of tall buildings.
III	Many people indoors feel movement. Hanging objects swing back and forth. People outdoors might not realize that an earthquake is occurring.
IV	Most people indoors feel movement. Hanging objects swing. Dishes, windows, and doors rattle. The earthquake feels like a heavy truck hitting the walls. A few people outdoors may feel movement. Parked cars rock.
V	Almost everyone feels movement. Sleeping people are awakened. Doors swing open or close. Dishes are broken. Pictures on the wall move. Small objects move or are turned over. Trees might shake. Liquids might spill out of open containers.
VI	Everyone feels movement. People have trouble walking. Objects fall from shelves. Pictures fall off walls. Furniture moves. Plaster in walls might crack. Trees and bushes shake. Damage is slight in poorly built buildings. No structural damage.
VII	People have difficulty standing. Drivers feel their cars shaking. Some furniture breaks. Loose bricks fall from buildings. Damage is slight to moderate in well-built buildings; considerable in poorly built buildings.
VIII	Drivers have trouble steering. Houses that are not bolted down might shift on their foundations. Tall structures such as towers and chimneys might twist and fall. Well-built buildings suffer slight damage. Poorly built structures suffer severe damage. Tree branches break. Hillsides might crack if the ground is wet. Water levels in wells might change.
IX	Well-built buildings suffer considerable damage. Houses that are not bolted down move off their foundations. Some underground pipes are broken. The ground cracks. Reservoirs suffer serious damage.
X	Most buildings and their foundations are destroyed. Some bridges are destroyed. Dams are seriously damaged. Large landslides occur. Water is thrown on the banks of canals, rivers, lakes. The ground cracks in large areas. Railroad tracks are bent slightly.
XI	Most buildings collapse. Some bridges are destroyed. Large cracks appear in the ground. Underground pipelines are destroyed. Railroad tracks are badly bent.
XII	Almost everything is destroyed. Objects are thrown into the air. The ground moves in waves or ripples. Large amounts of rock may move.

As one can see from the list above, rating the Intensity of an earthquake's effects does not require any instrumental measurements. Thus seismologists can use newspaper accounts, diaries, and other historical records to make intensity ratings of past earthquakes, for which there are no instrumental recordings. Such research helps promote understanding of the earthquake history of a region, and estimate future hazards.

5.0 Other External Events Modeling and Risk Quantification

5.1 Objectives and Scope

This document is intended to provide a concise and practical handbook to NRC risk analysts who routinely use the SAPHIRE software and the Standardized Plant Analysis Risk (SPAR) probabilistic risk assessment (PRA) models to quantify event and plant condition importances, and other ad-hoc risk analyses. It is a complementary document to the handbook cited in Ref. 5-1.

NRC risk analysts encounter many plant conditions and events reported by such means as inspection reports, licensee event reports (LERs), generic risk issues that lend themselves to PRA quantification and evaluation, every year. The need for quantification of the event / condition importance in terms of the two common risk measures of core damage frequency (CDF) and large early release frequency (LERF) arise in many of these cases.

This handbook provides NRC risk analysts practical guidance for modeling "other external events" scenarios and quantifying their CDF using SPAR models and SAPHIRE software. Other External Events" are defined in Appendix A of Ref. 5-2, excluding internal fires, internal flooding, and seismic events. For those events, complementary handbooks are already prepared.

External flooding and extreme winds / tornadoes are the two other external events that most likely may appear as scenarios in some PRA studies (non-targeted transportation accidents, such as nearby chemical transport explosions or inadvertent on-site air crash, may appear in rare instances). This handbook will focus on these two events.

The handbook assumes that:

1. The user has hands-on experience with the SAPHIRE code;

2. The user has performed and documented event/condition importance analysis or plant risk assessment cases for a period of at least three months (this is a suggested period, not a firm limit) under the supervision of an experienced (qualified) senior PRA analyst. The user is the primary author of documentation packages for such analyses which are reviewed and accepted by an NRC program.

The current scope is limited to other external events during power operation and calculation of CDF only.

Mainstream PRA terms and abbreviations that are used in this document are not defined; the intended reader is assumed to be familiar with them.

5.2 Scenario Definition and Quantification

This handbook focuses on external flooding and extreme winds / tornadoes. These events share many common traits with area events (like internal flooding) and loss of offsite power (LOOP) events. However, two aspects in which they are distinctly different are (1) they originate from outside the facility; and (2) there is little, if any, opportunity for mitigation (e.g., one can suppress a fire or terminate an internal event, but one cannot readily block an external flooding event or tornado, other than to pre-harden the facility). In fact the initiating event frequency of LOOP includes weather related LOOP. Weather-related LOOP events involve hurricanes, strong winds greater than 125 miles per hour, tornadoes, thunderstorms, snow, and ice storms.

As in internal flooding and fire scenarios, a two-step process is discussed to model other external event scenarios and quantify their CDFs:

1. Define scenarios that lead to core damage. For this purpose, define initiating event, calculate its frequency; identify damaged structures, systems and components (SSCs) and evaluate their recovery (or lack of recovery) potential and means.

   Using a structured model, such as a small event tree, define scenarios that stem from the initiating event; calculate their scenario frequencies, and transfer each scenario to an existing event tree (such as LOOP).

   See example in the next section for an application of this process.

2. Quantify the CDF of the sequences stemming from these scenarios. For this purpose, first the scenario conditional core damage probability (CCDP) is calculated by using a SPAR model and the SAPHIRE software. Then this CCDP is multiplied by the scenario frequency calculated in Step 1.

The sequences defined can be summarized in terms of a matrix containing the minimum amount of information to be able to quantify the scenario frequency, the scenario CCDP, and thus the sequence CDF:

CDF = Scenario Frequency * CCDP.

5.2.1 Define Scenarios

Examine the event/condition characteristics and define scenarios that lead to core damage. Summarize those scenarios in terms of a table, such as Table 5-1. The columns of this table are discussed below. Note that, each of these scenarios is treated as an initiating event and will be assigned an event tree.

1. Scenario name (initiating event ID). This always starts with an appropriate prefix such as (FLE, HWD, TOR, etc.) and is used both for the event tree and the initiating event names.

2. Scenario description.

3. Scenario frequency IEfreq (initiating event frequency).

Table 5-1 Example Matrix Defining Other External Event Scenarios

	Name	Description	IE Frequency	Equipment Lost	IE Caused	HEPs / Basic Events Affected	New Basic Events (failures) Introduced
1	OEX-DAM	Dam failure – external flooding event	1.0E-05	LOOP; SWS; no LOOP recovery; no SWS recovery	IE-LOOP	None / None	IE-OEX-DAM
	Name	**Description**	**IE Frequency**	**Equipment Lost**	**IE Caused**	**HEPs / Basic Events Affected**	**New Basic Events (failures) Introduced**
1	OEX-HUR	LOOP due to hurricane during Mode 4 operation	N/A (1)	Offsite AC power	IE-LOOP	No RCP seal LOCA; event-specific LOOP recovery probabilities	None
Notes: (1) = event analysis is made; initiating event frequency is set equal to 1.0							

4. Equipment lost. Equipment credited in the PRA that is lost due to the external event is listed in this column. Include trains/system that caused the external event, if such is possible (unlikely for other than internal fires and floods, not being addressed here) and is also lost.

5. Initiating event caused. This is the initiating event caused by the external event. In most cases, it is one of the internal initiating event categories already defined (such as loss of main feedwater (LOMFW), TRANS, loss of service water system (LOSWS), etc.). However, due to the potential for structural damage similar to seismic, e.g., tornadoes/high winds or air crash, it may be necessary to consider new or merged event trees where multiple internal events initiators could be triggered concurrently.

6. Human error probabilities (HEPs), recovery actions, and other basic events affected. List the basic events and operator actions that are affected by the event (failed, degraded). This is in addition to equipment listed in item 5 above.

7. New basic events (failures) introduced. List any new basic events to model the scenarios.

Other columns may be introduced as needed.

5.2.2 Quantify Sequence CDFs

The CDF of each sequence can be calculated as a product of the scenario frequency and the CCDP given the scenario has occurred:

CDF = IEfreq * CCDP.

The scenario frequency IEfreq is already calculated in the earlier step. The CCDP can be calculated by using the SAPHIRE code and the SPAR models. For this purpose, either a change set or the Graphical Evaluation Module (GEM) software can be used.

Once the sequence CDF is known, it can be used to estimate event/condition importance.

5.2.3 Weather-Related LOOP Recovery Distributions

LOOP recovery distributions for weather-related events differ from other LOOP events. They are given in Table 5-2, as taken from Ref. 5-4, SPAR 3.12 models.

Table 5-2 LOOP Recovery Distributions

Failure to Recover Offsite Power in X hours		
X	**Composite**	**Weather Related**
1	0.53	0.66
2	0.32	0.52
2.5	0.26	0.48
3	0.22	0.44
4	0.16	0.38
5	0.12	0.34

Failure to Recover Offsite Power in X hours		
X	Composite	Weather Related
6	0.010	0.31
7	0.08	0.28
8	0.07	0.26

Composite = Composite of plant-, switchyard-centered, and grid-, and weather-related LOOP categories.

5.2.4 Weather-Related LOOP Frequencies

The weather-related LOOP frequencies (per reactor critical year or calendar year at power, and units for shutdown) are given in Table 5-3, as taken from Ref. 5-4.

Table 5-3 LOOP Frequencies

LOOP Category	Mean	95%
Critical Operation		
Plant-centered	2.07E-3	7.96E-3
Switchyard-centered	1.04E-2	3.98E-2
Grid-related	1.86E-2	7.16E-2
Weather-related	4.83E-3	1.86E-2
All	3.59E-2	9.19E-2

LOOP Category	Mean	95%
Shutdown Operation		
Plant-centered	5.09E-02	2.06E-01
Switchyard-centered	1.00E-01	2.83E-01
Grid-related	9.13E-03	3.51E-02
Weather-related	3.52E-02	1.35E-01
All	1.96E-01	4.33E-01

5.2.5 Treatment of Hurricane-Related Events

Plants susceptible to hurricane events have procedures to bring plant to a shutdown state prior to an expected hurricane event. Thus, a plant is expected to be in a Mode 3 or Mode 4 shutdown state when the site experiences a hurricane event. The most likely consequence of such an event is loss off offsite power, with a plant specific-recovery distribution for that particular event. See Example 5.3.2 for treatment of LOOP following a hurricane, while the plant is in a shutdown state.

If a SPAR shutdown model is available for the plant in question, it can be used for estimating the importance of the event or plant condition. If the SPAR-SD model does not provide enough modeling detail to address specific issues associated with the event, the LOOP/SBO model from SPAR internal events may be used, with certain modifications, which can be implemented by a change set in SAPHIRE. The following modifications can be considered:

i). RPS failure is removed (no ATWS);
ii). RCP seal LOCA (for PWRs) is most likely not applicable and should be removed;
iii). PORV LOCA likelihood is considerably reduced; may be removed;
iv). Availability of AFW and MFW recovery should be established and kept in the model;
v). Event-specific offsite power recovery distribution may need to be calculated and used; as a minimum, generic severe-weather recovery distribution should be used.
vi). Operator actions outside of the buildings, or those that require travel from one building to another via outside should not be modeled, at least for the first 2-4 hours following the onsite of the hurricane at the site.
vii). Introduction of an operator action to start a mitigating system (modeled in the LOOP / SBO event trees), which otherwise, would have started automatically.

viii). Since the plant has been shutdown for a period of 4-8 hours, the time windows available for operator actions, and also for time to core melt are expected to be longer (more favorable) than those used for at-power operations. Thus, the plant condition/event importance estimates using the at-power LOOP event tree are expected to be on the conservative side.

5.3 Examples

This section discusses examples for illustrative purposes; the values used in the examples are for illustration only.

5.3.1 Example Condition Analysis

An external flooding event is postulated to occur due to a catastrophic dam failure upstream in a river that provides the intake water for the service water system (SWS). Although the main plant buildings are not expected to be flooded, a non-recoverable failure of SWS during the first 24-hour mission time is postulated. The initiating event frequency is estimated as 1.0E-05/year. The event is postulated to occur only after extreme rains, and the plant has a procedure to shutdown under these severe conditions and remain at Mode 2 or lower. For modeling purposes, the event occurs during Mode 2 operation. LOOP is also expected and is postulated. No offsite power recovery is expected during the mission time.

This scenario was not considered in the plant PRA. This newly discovered condition importance can be calculated for an exposure time of 1 year as

COND-IMP = 1 year * IE-freq * CCDP

Where

IE-Freq = 1.0E-05/year

and CCDP is calculated as 3.2E-02 by using the SPAR model with LOOP event tree, no SWS, no LOOP recovery and no SWS recovery. It is assumed that the LOOP success criteria developed for at-power conditions bound the Mode 2 conditions.

This sequence represents additional risk and there is no need to process base plant risk. Thus the condition importance is given by

COND-IMP = 1.0E-05 * 3.2E-02 = 3.2E-07,

for a 1-year exposure time.

5.3.2 Example Event Analysis

A dual-unit LOOP occurred at a nuclear power plant (NPP) site. Earlier that day both units commenced an orderly shutdown to prepare for the arrival of Category-3 Hurricane J. At the time of the LOOP, the site was experiencing hurricane force winds with both units in Mode 4.

This event is modeled as a loss of alternating current (AC) power event leading to loss of RHR cooling during Mode 4 with a 24-hour mission time (no structural damage, other than that in the switchyard or offsite which could cause LOOP, is postulated).

Assumptions

1. The risk of this event can be estimated by assuming that the success criteria for a LOOP event at power applies.

    This assumption has both conservative and non-conservative aspects that are deemed to be balancing from a risk point of view. Namely,

    a) Since the units are already shutdown, the decay heat is lower than at power. This gives a larger time window for operator actions, both for starting systems, or recovering power.

    b) Some mitigating safety systems, if needed, may require operator action to start; they may not be available for automatic actuation in Mode 4. One example of this is AFW cooling by SGs for unit 1.

2. For AC recovery time distribution, an event-specific calculation is made using SPAR-H model.

3. Credit for crosstie to other unit emergency diesel generator (EDG), which is already modeled in SPAR, is retained.

4. Unit 1 is assumed to go to steam generator (SG) cooling by auxiliary feedwater (AFW), if residual heat removal (RHR) cooling failed.

5. Unit 1 SPAR model is used to estimate the event importance.

6. The reactor coolant system (RCS) temperature and pressure conditions are such that no reactor coolant pump (RCP) seal LOCA challenge exists.

For this category 3 hurricane event, event-specific offsite power non-recovery probabilities are calculated.

Although no attempt was made to restore offsite power to the startup transformers during the hurricane, if EDG power was lost, offsite power could have been restored through Bay 2. However, weather conditions did hamper the restoration of offsite power to the units' electrical buses. Therefore, during the hurricane, safe shutdown loads remained connected to the EDGs even after power was capable of being restored to the east electrical switchyard buses because conditions would not allow personnel to safely inspect the switchyard. AC power recovery was feasible during the mission time of interest and credible. It is modeled in the event importance assessment.

In the actual event, the offsite power was restored to the emergency buses in 11 hours; during that time, EDGs powered the buses.

The following AC power recovery distribution is used:

OPERATOR FAILS TO RECOVER OFFSITE POWER:
IN 1 HR 1.0
IN 2 HRS 0.5
IN 3 HRS 0.05
IN ≥4 HRS 0.005

When this AC power recovery distribution is used, the CCDP is calculated as 1.8E-05, which is the event importance.

Compare this with SPAR severe weather AC power recovery failure distribution. Namely

OPERATOR FAILS TO RECOVER OFFISTE POWER:

IN 1 HR 4.6E-001
IN 2 HRS 3.6E-001
IN 3 HRS 3.0E-001
IN 4 HRS 2.5E-001
IN 5 HRS 2.2E-001
IN 6 HRS 2.0E-001
IN 7 HRS 1.8E-001

With this recovery distribution, the event importance is calculated as CCDP = 3.4E-05.

5.4 References

5-1. U.S. Nuclear Regulatory Commission, “Risk Assessment of Operational Events Handbook: Volume 1 - Internal Events,” Revision 1.01. December 2007.

5-2. American National Standard, “External Events PRA Methodology,” ANSI/ANS-58.21-2003, 2003.

5-3. Idaho National Laboratory, “A Feasibility and Demonstration Study – Incorporating External Events into SPAR Models,” February 2005.

5-4. U.S. Nuclear Regulatory Commission, “Reevaluation of Station Blackout Risk at Nuclear Power Plants - Analysis of Loss of Offsite Power Events: 1986-2004,” NUREG/CR-6890, Volume 1, December 2005. http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6890/

Appendix 5A. Dam Failure Rates for External Flooding

External flooding is due to precipitation, storm surge, tsunami, or rupture of an impoundment. The precipitation can be in the form of extreme rainfall or a rapidly melting snow pack. The dam or dike rupture can be due to overtopping by flood or “blue sky” piping and collapse. Storm surge is typically a coastal phenomenon. Tsunamis and their fresh water cousins, seiches are seismic and shoreline geography phenomena.

The existing nuclear sites based their flood protection on the recommendations of SRP section 2.4, Regulatory Guides (RGs) 1.159 (Rev. 2) and 1.102, with RG 1.159 providing actual data for sites east of the Rockies. The accepted analysis methodology is outlined in ANSI/ANS-2.8-1992, “Determining Design Basis Flooding at Power Reactor Sites.”

Tsunamis

Tsunamis are treated as rare and mild events on eastern coasts and not analyzed. On the West coast, California plants claim a less than 10 foot tsunami but agreed to a design basis three times higher. There are no plants on the Pacific Northwest coast where there is an active plate boundary that can generate local large tsunamis.

Weather-Related Flood

Much of the country has already been analyzed by the U.S. Army Corps of Engineers and Federal Emergency Management Agency (FEMA) for flood and surge that are based on National Oceanic and Atmospheric Administration (NOAA) and U.S. Geological Survey (USGS) data. Virtually all licensees used the NRC recommended Refs.5-1 through 5-4 for input in Section 2.4 of their SARs.

The hurricane and precipitation data record only goes back 100 to 200 years. Probable maximum hurricane (PMH) values are developed from storm history over a wide stretch of coast extrapolated out about 2000 years. Probable maximum precipitation (PMP) values are developed from the100 year record maximum for the area. Both PMP and PMH are single value parameters that are deterministic and only of limited use for PRA. If more recent data is desired, precipitation frequency estimates are available out to 1000 years from NOAA at http://hdsc.nws.noaa.gov/hdsc/pfds/index.html and recent storm surge estimates are incomplete but available at the National Hurricane Study Program at http://chps.sam.usace.army.mil/USHESdata/HESHOME.htm .

Dam and Dike Failure

Dam failure is well documented and can be characterized by type of dam. Table 5A-1 is a summary of point estimate failure rates for dams that are broken down by large dams (>50 ft) and all sized dams. Characteristics of US dams and dam failures are available at the National Inventory of Dams, http://crunch.tec.army.mil/nid/webpages/nid.cfm and the National Performance of Dams Program, http://npdp.stanford.edu/index.html .

Of the 79,777 dams in the US, 72% are embankment type and 28% are concrete. Nineteenth century dams would fail at 5% in the first five years after construction but would settle out to a 1 to 4% additional failure by 20 years of life. This was reduced to 2% in the first 5 years for dams built after 1930. By 1960, dam failure rates were less than 0.01% due to better engineering. Whatever the era, half of all dams that ever fail, do so in the first five years. This high infant mortality is often due to piping in the soil around the dam or underneath it. Even concrete dams are not immune.

However, dam construction dropped dramatically after 1980 so that nearly all dams are older than 5 years.

Dams as far up or downstream as 300 miles should be considered for both flood and loss of heat sink. It is noteworthy that all forms of dams have a failure rate between 1E-4 and 4E-4, even for blue sky events. Determining flood levels, however, is a complex matter. The USACOE has software named HEC that when combined with GIS geographical data will model river flow and flooding in great detail.

Weather based floods remain in the deterministic world because the input conditions are always from the same source as was used in the original plant design basis. Besides, the growth of the maximum precipitation only increases about 20% when a 100 year interval is compared to a 1000 year interval. With only 100 years of data available in many locations, projecting beyond a 1E-3/yr event is very uncertain anyway.

Table 5A-1 Dam Failure Rates

	(all dams)	Failures	Dam-years	apost	bpost	Mean	5%	50%	95%
1	All Arch Dams	2	9101	2.5	12163.2644	2.055E-04	4.709E-05	1.789E-04	4.551E-04
2	All Buttress Dams	2	9819	2.5	12881.2644	1.941E-04	4.446E-05	1.689E-04	4.297E-04
3	All Concrete Dams	10	110227	10.5	113289.2644	9.268E-05	5.116E-05	8.976E-05	1.442E-04
4	All Earth Dams	366	2240403	366.5	2243465.2644	1.634E-04	1.496E-04	1.632E-04	1.776E-04
5	All Gravity Dams	28	122798	28.5	125860.2644	2.264E-04	1.615E-04	2.238E-04	3.004E-04
6	All Masonry Dams	5	21692	5.5	24754.2644	2.222E-04	9.240E-05	2.089E-04	3.974E-04
7	All Multi-Arch Dams	0	240	0.5	3302.2644	1.514E-04	5.954E-07	6.888E-05	5.816E-04
8	All Rockfill Dams	7	73806	7.5	76868.2644	9.757E-05	4.723E-05	9.327E-05	1.626E-04
9	All Stone Dams	2	11365	2.5	14427.2644	1.733E-04	3.970E-05	1.508E-04	3.837E-04
10	All Timber Crib Dams	3	6536	3.5	9598.2644	3.646E-04	1.129E-04	3.306E-04	7.328E-04
T	Total	425	2605987	0.5	3062.2644	1.633E-04	6.420E-07	7.428E-05	6.272E-04

No statistical difference among dam types. P-value = 0.15096. Empirical Bayes distribution does not exit since routine failed to converge. Prior distribution is obtained using the total values and obtaining using a Jeffreys' prior distribution. Then obtained uncertainty distribution using CNIP.

	(dams over 50 feet high)	Failures	Dam-years	apost	bpost	Mean	5%	50%	95%
1	Buttress Dams Over 50 Feet High	0	1876	2.4026	11970.7049	2.007E-04	4.410E-05	1.736E-04	4.497E-04
2	Arch Dams Over 50 Feet High	2	5667	4.4026	15761.7049	2.793E-04	1.018E-04	2.585E-04	5.280E-04
3	Concrete Dams Over 50 Feet High	0	19215	2.4026	29309.7049	8.197E-05	1.801E-05	7.092E-05	1.837E-04
4	Earth Dams Over 50 Feet High	56	144810	58.4026	154904.7049	3.770E-04	2.997E-04	3.749E-04	4.617E-04
5	Gravity Dams Over 50 Feet High	7	19542	9.4026	29636.7049	3.173E-04	1.683E-04	3.061E-04	5.044E-04
6	Masonry Dams Over 50 Feet High	0	1987	2.4026	12081.7049	1.989E-04	4.370E-05	1.721E-04	4.456E-04
7	Multi-Arch Dams Over 50 Feet High	0	77	2.4026	10171.7049	2.362E-04	5.190E-05	2.044E-04	5.293E-04
8	Rockfill Dams Over 50 Feet High	4	20010	6.4026	30104.7049	2.127E-04	9.568E-05	2.017E-04	3.671E-04
T	Total	69	213184	2.4026	10094.7049	2.380E-04	5.230E-05	2.059E-04	5.333E-04

Prior distribution obtained using empirical Bayes method in SAS.

Notes:
Dams constructed with mixed materials are not counted; dams with no construction dates available are not counted.

Appendix 1. Frequencies of Seismically-Induced LOOP Events for SPAR Models

1. Objective

This report provides frequencies of seismically-induced loss of offsite power (LOOP) events for U.S. nuclear power plants (NPPs). These LOOP frequencies could be used for external events scenarios in event importance calculations. The intended user is the U.S. NRC senior reactor analysts (SRAs).

2. Input

The inputs to these calculations are:

i). seismic initiating event frequencies (seismic hazard distribution) as a function of seismic g level (NUREG-1488, April 1994);

ii). structures, systems and components (SSCs) (for example ceramic insulator) fragilities as a function of g level (NUREG-6544, April 1998).

Attachment A provides the details.

3. Summary of Results

The input data is combined as a weighted average over the g levels to obtain mean value estimates, as shown in Attachment A. The following information is provided as shown in Table 1:

1. Seismic initiating event mean frequency of a 0.05g or higher earthquake per year;

2. Given an earthquake occurs, the conditional LOOP probability caused by the earthquake (based on failure of ceramic insulators);

3. Frequency of seismically induced LOOP event (per year).

Tables 2 and 3 compare the seismically induced LOOP frequency with frequencies of other “internal LOOP events.” Average durations of the LOOP events are also provided in the same tables.

4. Comments

i). These results show that the seismically-induced LOOP frequencies are at least two orders of magnitude lower than LOOP frequencies calculated for internal events. However, the power recovery may not be feasible for an extended time period, following a seismic event. This fact should be factored into the calculation of plant risk due to seismically-induced LOOP events.

ii). A small fraction of these LOOP events (at high seismic g values) will have additional SSC failures that would cause other initiating events, such as small loss-of-coolant accident (LOCA), large LOCA, etc.

iii). For the sites to the east of the Rocky Mountains, a calculational tool is set up in terms of an MS EXCEL workbook and is used repeatedly to calculate the seismically-induced LOOP frequencies for 61 sites. The same generic ceramic insulator seismic fragility distribution is used for these calculations.

iv). For the four sites west of the Rocky Mountains, plant-specific seismic event frequency distributions (seismic hazard curves) are obtained from Individual Plant Examination of External Events (IPEEE) submittals (they are not given in the reference NUREG). The seismic fragility distributions for LOOP are also obtained from the same source. Then, the same calculational tool is used for LOOP frequency calculations.

v). The calculations can be readily customized for plant-specific SSC fragilities (e.g., ceramic insulators) and/or hazard curves. The MS EXCEL workbook named Seismically-Induced LOOP – Tables.xls is available for this purpose.

Table 1 Frequencies of Seismically-Induced LOOP Events

	Plant	Seismic IEV Frequency	Cond. Prob. of LOOP	Seis. Indu. LOOP Frequency	Plant Type	# of Units
		A	B	A*B		
1-2	ANO 1 & 2	1.27E-03	6.59E-02	8.39E-05	B&W/CE	2
3-4	Beaver Valley 1 & 2	8.78E-04	8.52E-02	7.48E-05	W	2
5	Braidwood 1 & 2	4.30E-04	6.64E-02	2.85E-05	W	2
6-7	Browns Ferry 2 & 3	9.12E-04	5.63E-02	5.14E-05	BWR	2
8	Brunswick 1 & 2	1.53E-03	6.95E-02	1.06E-04	BWR	2
9	Byron 1 & 2	5.09E-04	7.23E-02	3.68E-05	W	2
10	Callaway	1.08E-03	3.82E-02	4.14E-05	W	1
11	Calvert Cliffs 1 & 2	7.67E-04	8.24E-02	6.33E-05	CE	2
12	Catawba 1 & 2	1.20E-03	6.27E-02	7.52E-05	W	2
13	Clinton	1.55E-03	6.87E-02	1.06E-04	BWR	1
14	Columbia (ex-WNP-2)	1.30E-03	1.37E-01	1.78E-04	BWR	1
15	Comanche Peak 1 & 2	1.41E-04	5.52E-02	7.78E-06	W	2
16	Cook 1 & 2	5.01E-04	7.77E-02	3.89E-05	W	2
17	Cooper	1.16E-03	1.15E-01	1.33E-04	BWR	1
18	Crystal River 3	1.48E-04	7.76E-02	1.15E-05	B&W	1
19	Davis-Besse	1.07E-03	6.12E-02	6.55E-05	B&W	1
20	Diablo Canyon	1.85E-02	5.71E-02	1.06E-03	W	2
21	Dresden	4.58E-04	7.23E-02	3.31E-05	BWR	2
22	Duane Arnold	1.55E-04	6.28E-02	9.72E-06	BWR	1
23	Farley 1 & 2	2.00E-04	7.17E-02	1.43E-05	W	2
24	Fermi 2	6.01E-04	5.29E-02	3.18E-05	BWR	1
25	Fitzpatrick	7.34E-04	4.95E-02	3.63E-05	BWR	1
26	Fort Calhoun	8.78E-04	1.21E-01	1.06E-04	CE	1
27	Ginna	8.48E-04	7.07E-02	6.00E-05	W	1
28	Grand Gulf	3.31E-04	7.12E-02	2.35E-05	BWR	1
29	Hatch 1 & 2	6.13E-04	6.83E-02	4.19E-05	BWR	2
30	Hope Creek	9.72E-04	8.09E-02	7.86E-05	BWR	1
31-32	Indian Point 2	1.15E-03	7.54E-02	8.69E-05	W	2
33	Kewaunee	3.04E-04	9.38E-02	2.85E-05	W	1
34	LaSalle 1 & 2	8.25E-04	8.66E-02	7.14E-05	BWR	2
35	Limerick 1 & 2	1.22E-03	7.66E-02	9.35E-05	BWR	2
36	McGuire 1 & 2	1.08E-03	5.86E-02	6.35E-05	W	2
37-38	Millstone 2 & 3	9.97E-04	7.49E-02	7.46E-05	CE/W	2
39	Monticello	3.56E-04	1.01E-01	3.58E-05	BWR	1
40-41	Nine Mile Point 1 & 2	7.30E-04	4.98E-02	3.63E-05	BWR	2
42	North Anna 1 & 2	1.15E-03	7.42E-02	8.55E-05	W	2
43	Oconee 1, 2, & 3	1.28E-03	6.69E-02	8.57E-05	B&W	3
44	Oyster Creek	8.53E-04	6.98E-05	6.98E-05	BWR	1
45	Palisades	3.92E-04	7.78E-02	3.05E-05	CE	1
46	Palo Verde 1, 2, & 3	3.00E-02	1.79E-03	5.37E-05	CE	3
47	Peach Bottom 2 & 3	1.06E-03	7.63E-02	8.08E-05	BWR	2
48	Perry	4.48E-04	6.96E-02	3.12E-05	BWR	1
49	Pilgrim	2.81E-03	1.16E-01	3.25E-04	BWR	1

	Plant	Seismic IEV Frequency	Cond. Prob. of LOOP	Seis. Indu. LOOP Frequency	Plant Type	# of Units
		A	B	A*B		
50	Point Beach 1 & 2	3.13E-04	9.32E-02	2.91E-05	W	2
51	Prairie Island 1 & 2	3.15E-04	1.04E-01	3.28E-05	W	2
52	Quad Cities 1 & 2	3.66E-04	6.37E-02	2.33E-05	BWR	2
53	River Bend	1.97E-04	7.45E-02	1.46E-05	BWR	1
54	Robinson 2	2.72E-03	8.78E-02	2.39E-04	W	1
55-56	Saint Lucie 1 & 2	1.47E-04	9.02E-02	1.33E-05	CE	2
57	Salem 1 & 2	9.59E-04	8.06E-02	7.73E-05	W	2
58	San Onofre 2 & 3	5.20E-03	4.71E-01	2.45E-03	CE	2
59	Seabrook	2.34E-03	8.08E-02	1.89E-04	W	1
60	Sequoyah 1 & 2	1.33E-03	7.46E-02	9.92E-05	W	2
61	Shearon Harris	5.85E-04	6.25E-02	3.65E-05	W	1
62	South Texas 1 & 2	1.63E-04	8.59E-02	1.40E-05	W	2
63	Surry 1 & 2	6.03E-04	7.83E-02	4.72E-05	W	2
64	Susquehanna 1 & 2	8.46E-04	7.12E-02	6.02E-05	BWR	2
65	TMI-1	1.11E-03	7.68E-02	8.51E-05	B&W	1
66	Turkey Point 3 & 4	1.23E-04	7.97E-02	9.78E-06	W	2
67	V.C. Summer	1.83E-03	6.36E-02	1.17E-04	W	1
68	Vermont Yankee	1.29E-03	6.20E-02	8.02E-05	BWR	1
69	Vogtle 1 & 2	2.50E-03	7.05E-02	1.76E-04	W	2
70	Waterford	2.86E-04	8.56E-02	2.45E-05	CE	1
71	Watts Bar	1.26E-03	7.44E-02	9.36E-05	W	1
72	Wolf Creek	3.29E-04	5.70E-02	1.87E-05	W	1
			Average =	**1.20E-04**	**Sum =**	**103**

Note:
Bold numbers in the first column identify the four sites to the West of Rocky Mountains.

Table 2 LOOP Frequency Comparisons - Power Operation

		Mean Frequency	95%	Mean Duration (hrs)	95% Duration
1	Plant centered	2.38E-03	9.15E-03	0.5	2
2	Switchyard centered	8.74E-03	3.36E-02	1.3	5
3	Grid related	1.67E-02	6.41E-02	2.7	9.3
4	Severe weather related	2.98E-03	1.15E-02	5.4	25.1
5	Extreme weather related	2.32E-03	8.91E-03	78	187.4
6	Seismically induced	1.2E-04		78	187.4
	Sum =	**3.32E-02**			

Table 3 LOOP Frequency Comparisons - Shutdown Operation

		Mean Frequency	95%	Mean Duration (hrs)	95% Duration
1	Plant centered	5.16E-02	2.03E-01	0.5	2
2	Switchyard centered	1.02E-01	2.92E-01	1.3	5
3	Grid related	9.26E-03	3.56E-02	2.7	9.3
4	Severe weather related	2.51E-02	9.65E-02	5.4	25.1
5	Extreme weather related	1.32E-03	5.08E-03	78	187.4
6	Seismically induced	1.2E-04		78	187.4
	Sum =	**1.89E-01**			

Source = INEEL/EXT-04-02326, October 20004

Attachment A - Calculations

This attachment documents the calculational details of the frequencies of seismically-Induced LOOP events given in the main body of the report.

A-1 Input-1: Seismic Event Frequencies

The seismic event frequencies for 69 NPP sites east of the Rocky Mountains are given in NUREG-1488 (April 1994). Data taken from this source for seven example plants east of the Rock Mountains is given in Table AA-1. Similar data for plants to the West of the Rock Mountains may be obtained from the utilities, or their IPEEEs.

A-2 Input-2: SSC Fragilities leading to LOOP

Generally, the ceramic insulators with the lowest fragilities among the SSCs modeled in the PRAs govern the occurrence of LOOP following a seismic event. The generic fragility data for ceramic insulators is taken from NUREG-6544 (April 1998) as shown in Table AA-2. The mean failure probabilities at different g level earthquakes are calculated by using the equation:

$$P_{fail}(a) = \Phi\ [\ \ln(a/a_m)\ /\ \mathrm{sqrt}(\beta_r^2 + \beta_u^2)]$$

Where Φ is the standard normal cumulative distribution function and

a = median acceleration level of the seismic event;
a_m = median of the component fragility (or median capacity);
β_r = logarithmic standard deviation representing random uncertainty;
β_u = logarithmic standard deviation representing systematic or modeling uncertainty.

Fragilities of SSCs that would cause LOOP for the plants west of the Rocky Mountains can also be calculated by using the information taken from their IPEEEs.

Calculations of mean failure probabilities of SSCs as a function of g level for various cases are shown in Tables AA-2 and AA-3.

A-3 Calculation of LOOP Frequency

Once the initiating event frequencies at different g levels and their corresponding conditional LOOP probabilities are known, as given in Tables AA-1 through AA-3, the frequency of seismically-induced LOOP event can be calculated as a weighed average of frequencies at different g intervals. This is shown for seven plants in Tables A-1 through A-7. The summary Table 1 has the seismically induced LOOP frequencies for all SPAR models.

A-4 Summary of Results

The summary of results for

1. Seismic initiating event frequencies
2. Conditional probability of LOOP given seismic event
3. Frequency of seismically-induced LOOP event

for SPAR models is given in Table A-1.

The calculations can be readily customized for plant-specific SSC fragilities and/or hazard curves.

The seismically-induced LOOP frequency calculations for the 72 SPAR model plants are performed in a MS EXCEL workbook, which can be found by ADAMS accession number ML062540239.

Table AA-1 Seismic Initiating Event Frequencies

g value	mean frequency of exceedance (per year)						
	Clinton	**Comanche Peak**	**Duane Arnold**	**Limerick**	**Pilgrim**	**Robinson**	**Vogtle**
0.05	1.55E-03	1.41E-04	1.55E-04	1.22E-03	2.81E-03	2.72E-03	2.50E-03
0.08	8.08E-04	6.79E-05	8.11E-05	6.99E-04	1.78E-03	1.57E-03	1.36E-03
0.15	2.46E-04	1.88E-05	2.38E-05	2.29E-04	7.15E-04	5.47E-04	4.15E-04
0.25	9.42E-05	6.42E-06	8.21E-06	8.35E-05	3.27E-04	2.26E-04	1.55E-04
0.30	6.54E-05	4.19E-06	5.36E-06	5.55E-05	2.41E-04	1.60E-04	1.06E-04
0.40	3.57E-05	2.02E-06	2.58E-06	2.75E-05	1.44E-04	8.99E-05	5.68E-05
0.50	2.17E-05	1.10E-06	1.40E-06	1.52E-05	9.38E-05	5.57E-05	3.42E-05
0.65	1.17E-05	5.08E-07	6.42E-07	7.10E-06	5.45E-05	3.06E-05	1.83E-05
0.80	6.89E-06	2.66E-07	3.34E-07	3.73E-06	3.43E-05	1.85E-05	1.09E-05
1.00	3.79E-06	1.28E-07	1.59E-07	1.79E-06	2.02E-05	1.04E-05	6.18E-06
Seismic IE Freq. =	**1.55E-03**	**1.41E-04**	**1.55E-04**	**1.22E-03**	**2.81E-03**	**2.72E-03**	**2.50E-03**

For 69 NPP sites east of Rocky Mountains, NUREG-1488 provides Seismic IEV frequencies.
For other plants West of Rocky mountains, this information can be obtained either from the plant, or from the literature, as needed.

Table AA-2 Fragilities of SSCs causing seismically induced LOOP

	median capacity	β_r	β_u	**HCLPF**	
Generic Ceramic Insulators	0.3	0.3	0.45	0.1	used for all sites except those West of the Rocky Mountains
Switchyard Fragility	0.31	0.25	0.43	0.1	Columbia
Offsite Power	1.40	0.2200	0.2	0.7	Diablo Canyon
Ceramic Insulators	0.3	0.3	0.45	0.1	Palo Verde

Table AA-3 Calculation of mean failure probability of SSCs (causing LOOP) as a function of g level

Ceramic Insulators					
	median capacity	β_r	β_u	**HCLPF**	**HCLPF (calculated)**
	0.3	0.3	0.45	0.1	0.087
g value	**pf (median)**	**pf(mean)**		**g value**	**pf(mean)**
0.05	1.17E-09	4.62E-04		0.05	0.0005
0.08	5.27E-06	7.26E-03		0.1	0.0211
0.15	1.04E-02	1.00E-01		0.15	0.1000
0.25	2.72E-01	3.68E-01		0.2	0.2267
0.3	5.00E-01	5.00E-01		0.25	0.3680
0.4	8.31E-01	7.03E-01		0.3	0.5000
0.5	9.56E-01	8.28E-01		0.35	0.6122
0.65	9.95E-01	9.24E-01		0.4	0.7026
0.8	9.99E-01	9.65E-01		0.45	0.7733
1	1.00E+00	9.87E-01		0.5	0.8275
				0.55	0.8688
				0.6	0.9000
pf = probability of failure				0.65	0.9236
median pf is not used; for comparison only.				0.7	0.9414
Note that median overshoots mean above 0.3g				0.75	0.9549
				0.8	0.9651
				0.85	0.9729
				0.9	0.9789
				0.95	0.9835
				1	0.9870

For SSC fragilities, a simple generic list is available in NUREG-6544 Table 6-1.

Columbia Switchyard Fragility					
	median capacity	β_r	β_u	**HCLPF**	**HCLPF (calculated)**
	0.31	0.25	0.43	0.1	0.101
g value	**pf (median)**	**pf(mean)**			
0.05	1.47E-13	1.22E-04			
0.1	1.25E-04	1.15E-02			
0.2	8.83E-02	1.89E-01			
0.3	5.00E-01	4.74E-01			
0.4	8.31E-01	6.96E-01			
0.5	9.56E-01	8.32E-01			
0.6	9.90E-01	9.08E-01			
0.7	9.98E-01	9.49E-01			
0.8	9.99E-01	9.72E-01			
0.9	1.00E+00	9.84E-01			
1	1.00E+00	9.91E-01			

Diablo Canyon	Table 3-8 (page 3-53) of IPEEE Submittal				
Offsite Power	**median capacity**	**β_r**	**β_u**	**HCLPF**	**HCLPF (calculated)**
	1.40	0.2200	0.2	0.7	0.702
g value		**pf(mean)**			
0.2		2.99E-11			
0.5		2.67E-04			
0.8		2.99E-02			
1		1.29E-01			
1.2		3.02E-01			
1.5		5.92E-01			
2		8.85E-01			
2.5		9.74E-01			
3		9.95E-01			
4		1.00E+00			

San Onofre	Table 3.6-1 (page 3.83) of IPEEE Submittal				
Switchyard	**SA(g)**	**β_r**	**β_u**	**HCLPF**	**HCLPF (calculated)**
	0.74	0.2	0.34		0.304

IPEEE reports fragility in spectral acceleration; use generic fragility in units of PGA from Table AA-3 for failure probability calculations.

Ceramic Insulators used for Palo Verde					
	median capacity	**β_r**	**β_u**	**HCLPF**	**HCLPF (calculated)**
	0.3	0.3	0.45	0.1	0.087
g value	**pf (median)**	**pf(mean)**		**g value**	**pf(mean)**
0.01	0.00E+00	1.61E-10		0.05	0.0005
0.02	0.00E+00	2.77E-07		0.1	0.0211
0.05	1.17E-09	4.62E-04		0.15	0.1000
0.07	6.15E-07	3.56E-03		0.2	0.2267
0.1	1.25E-04	2.11E-02		0.25	0.3680
0.15	1.04E-02	1.00E-01		0.3	0.5000
0.2	8.83E-02	2.27E-01		0.35	0.6122
0.3	5.00E-01	5.00E-01		0.4	0.7026
0.5	9.56E-01	8.28E-01		0.45	0.7733
1	1.00E+00	9.87E-01		0.5	0.8275
				0.55	0.8688
				0.6	0.9000
pf = probability of failure				0.65	0.9236
median pf is not used; for comparison only.				0.7	0.9414
Note that median overshoots mean above 0.3g				0.75	0.9549
				0.8	0.9651
				0.85	0.9729
				0.9	0.9789
				0.95	0.9835
				1	0.9870

Table A-1 Clinton

g value	mean f per year	LOOP Probability	EQ g interval	Interval IEV Frequency	Interval Conditional LOOP Probability	Weighted Average
0.05	1.55E-03	4.62E-04	.05-.08	7.39E-04	1.83E-03	1.35E-06
0.08	8.08E-04	7.26E-03	.08-.15	5.63E-04	2.70E-02	1.52E-05
0.15	2.46E-04	1.00E-01	.15-.25	1.51E-04	1.92E-01	2.91E-05
0.25	9.42E-05	3.68E-01	.25-.30	2.88E-05	4.29E-01	1.23E-05
0.30	6.54E-05	5.00E-01	.30-.40	2.97E-05	5.93E-01	1.76E-05
0.40	3.57E-05	7.03E-01	.40-.50	1.40E-05	7.63E-01	1.07E-05
0.50	2.17E-05	8.28E-01	.50-.65	1.01E-05	8.74E-01	8.79E-06
0.65	1.17E-05	9.24E-01	.65-.80	4.76E-06	9.44E-01	4.49E-06
0.80	6.89E-06	9.65E-01	.80-1	3.10E-06	9.76E-01	3.03E-06
1.00	3.79E-06	9.87E-01	>1	3.79E-06	1.00E+00	3.79E-06
			Sum =	**1.55E-03**		**1.06E-04**
SE Initiating Event Frequency =				1.55E-03	CCDP =	6.87E-02
Seismically induced LOOP probability =				6.87E-02		
Seismically induced LOOP frequency =				1.06E-04		

Table A-2 Comanche Peak

g value	mean f per year	LOOP Probability	EQ g interval	Interval IEV Frequency	Interval Conditional LOOP Probability	Weighted Average
0.05	1.41E-04	4.62E-04	.05-.08	7.31E-05	1.83E-03	1.34E-07
0.08	6.79E-05	7.26E-03	.08-.15	4.91E-05	2.70E-02	1.32E-06
0.15	1.88E-05	1.00E-01	.15-.25	1.24E-05	1.92E-01	2.37E-06
0.25	6.42E-06	3.68E-01	.25-.30	2.23E-06	4.29E-01	9.57E-07
0.30	4.19E-06	5.00E-01	.30-.40	2.17E-06	5.93E-01	1.29E-06
0.40	2.02E-06	7.03E-01	.40-.50	9.20E-07	7.63E-01	7.02E-07
0.50	1.10E-06	8.28E-01	.50-.65	5.92E-07	8.74E-01	5.18E-07
0.65	5.08E-07	9.24E-01	.65-.80	2.42E-07	9.44E-01	2.28E-07
0.80	2.66E-07	9.65E-01	.80-1	1.38E-07	9.76E-01	1.35E-07
1.00	1.28E-07	9.87E-01	>1	1.28E-07	1.00E+00	1.28E-07
			Sum =	**1.41E-04**		**7.78E-06**
SE Initiating Event Frequency =			1.41E-04		CCDP =	5.52E-02
Seismically induced LOOP probability =			5.52E-02			
Seismically induced LOOP frequency =			7.78E-06			

Table A-3 Duane Arnold

g value	mean f per year	LOOP Probability	EQ g interval	Interval IEV Frequency	Interval Conditional LOOP Probability	Weighted Average
0.05	1.55E-04	4.62E-04	.05-.08	7.38E-05	1.83E-03	1.35E-07
0.08	8.11E-05	7.26E-03	.08-.15	5.73E-05	2.70E-02	1.54E-06
0.15	2.38E-05	1.00E-01	.15-.25	1.56E-05	1.92E-01	2.99E-06
0.25	8.21E-06	3.68E-01	.25-.30	2.85E-06	4.29E-01	1.22E-06
0.30	5.36E-06	5.00E-01	.30-.40	2.78E-06	5.93E-01	1.64E-06
0.40	2.58E-06	7.03E-01	.40-.50	1.19E-06	7.63E-01	9.05E-07
0.50	1.40E-06	8.28E-01	.50-.65	7.55E-07	8.74E-01	6.60E-07
0.65	6.42E-07	9.24E-01	.65-.80	3.08E-07	9.44E-01	2.91E-07
0.80	3.34E-07	9.65E-01	.80-1	1.74E-07	9.76E-01	1.70E-07
1.00	1.59E-07	9.87E-01	>1	1.59E-07	1.00E+00	1.59E-07
			Sum =	**1.55E-04**		**9.72E-06**
SE Initiating Event Frequency =			1.55E-04		CCDP =	6.28E-02
Seismically induced LOOP probability =			6.28E-02			
Seismically induced LOOP frequency =			9.72E-06			

Table A-4 Limerick

g value	mean f per year	LOOP Probability	EQ g interval	Interval IEV Frequency	Interval Conditional LOOP Probability	Weighted Average
0.05	1.22E-03	4.62E-04	.05-.08	5.21E-04	1.83E-03	9.54E-07
0.08	6.99E-04	7.26E-03	.08-.15	4.70E-04	2.70E-02	1.27E-05
0.15	2.29E-04	1.00E-01	.15-.25	1.46E-04	1.92E-01	2.79E-05
0.25	8.35E-05	3.68E-01	.25-.30	2.80E-05	4.29E-01	1.20E-05
0.30	5.55E-05	5.00E-01	.30-.40	2.80E-05	5.93E-01	1.66E-05
0.40	2.75E-05	7.03E-01	.40-.50	1.23E-05	7.63E-01	9.38E-06
0.50	1.52E-05	8.28E-01	.50-.65	8.10E-06	8.74E-01	7.08E-06
0.65	7.10E-06	9.24E-01	.65-.80	3.37E-06	9.44E-01	3.18E-06
0.80	3.73E-06	9.65E-01	.80-1	1.94E-06	9.76E-01	1.89E-06
1.00	1.79E-06	9.87E-01	>1	1.79E-06	1.00E+00	1.79E-06
			Sum =	**1.22E-03**		**9.35E-05**
SE Initiating Event Frequency =			1.22E-03		CCDP =	7.66E-02
Seismically induced LOOP probability =			7.66E-02			
Seismically induced LOOP frequency =			9.35E-05			

Table A-5 Pilgrim

g value	mean f per year	LOOP Probability	EQ g interval	Interval IEV Frequency	Interval Conditional LOOP Probability	Weighted Average
0.05	2.81E-03	4.62E-04	.05-.08	1.04E-03	1.83E-03	1.90E-06
0.08	1.78E-03	7.26E-03	.08-.15	1.06E-03	2.70E-02	2.86E-05
0.15	7.15E-04	1.00E-01	.15-.25	3.88E-04	1.92E-01	7.45E-05
0.25	3.27E-04	3.68E-01	.25-.30	8.62E-05	4.29E-01	3.70E-05
0.30	2.41E-04	5.00E-01	.30-.40	9.69E-05	5.93E-01	5.74E-05
0.40	1.44E-04	7.03E-01	.40-.50	5.03E-05	7.63E-01	3.84E-05
0.50	9.38E-05	8.28E-01	.50-.65	3.93E-05	8.74E-01	3.44E-05
0.65	5.45E-05	9.24E-01	.65-.80	2.02E-05	9.44E-01	1.90E-05
0.80	3.43E-05	9.65E-01	.80-1	1.41E-05	9.76E-01	1.38E-05
1.00	2.02E-05	9.87E-01	>1	2.02E-05	1.00E+00	2.02E-05
			Sum =	**2.81E-03**		**3.25E-04**
SE Initiating Event Frequency =			2.81E-03		CCDP=	1.16E-01
Seismically induced LOOP probability =			1.16E-01			
Seismically induced LOOP frequency =			3.25E-04			

Table A-6 Robinson

g value	mean f per year	LOOP Probability	EQ g interval	Interval IEV Frequency	Interval Conditional LOOP Probability	Weighted Average
0.05	2.72E-03	4.62E-04	.05-.08	1.15E-03	1.83E-03	2.11E-06
0.08	1.57E-03	7.26E-03	.08-.15	1.02E-03	2.70E-02	2.74E-05
0.15	5.47E-04	1.00E-01	.15-.25	3.21E-04	1.92E-01	6.16E-05
0.25	2.26E-04	3.68E-01	.25-.30	6.56E-05	4.29E-01	2.81E-05
0.30	1.60E-04	5.00E-01	.30-.40	7.01E-05	5.93E-01	4.15E-05
0.40	8.99E-05	7.03E-01	.40-.50	3.42E-05	7.63E-01	2.60E-05
0.50	5.57E-05	8.28E-01	.50-.65	2.51E-05	8.74E-01	2.20E-05
0.65	3.06E-05	9.24E-01	.65-.80	1.21E-05	9.44E-01	1.15E-05
0.80	1.85E-05	9.65E-01	.80-1	8.06E-06	9.76E-01	7.87E-06
1.00	1.04E-05	9.87E-01	>1	1.04E-05	1.00E+00	1.04E-05
			Sum =	**2.72E-03**		**2.39E-04**
SE Initiating Event Frequency =			2.72E-03		CCDP=	8.78E-02
Seismically induced LOOP probability =			8.78E-02			
Seismically induced LOOP frequency =			2.39E-04			

Table A-7 Vogtle

g value	mean f per year	LOOP Probability	EQ g interval	Interval IEV Frequency	Interval Conditional LOOP Probability	Weighted Average
0.05	2.50E-03	4.62E-04	.05-.08	1.14E-03	1.83E-03	2.09E-06
0.08	1.36E-03	7.26E-03	.08-.15	9.41E-04	2.70E-02	2.54E-05
0.15	4.15E-04	1.00E-01	.15-.25	2.61E-04	1.92E-01	5.00E-05
0.25	1.55E-04	3.68E-01	.25-.30	4.86E-05	4.29E-01	2.08E-05
0.30	1.06E-04	5.00E-01	.30-.40	4.92E-05	5.93E-01	2.92E-05
0.40	5.68E-05	7.03E-01	.40-.50	2.26E-05	7.63E-01	1.72E-05
0.50	3.42E-05	8.28E-01	.50-.65	1.59E-05	8.74E-01	1.39E-05
0.65	1.83E-05	9.24E-01	.65-.80	7.36E-06	9.44E-01	6.95E-06
0.80	1.09E-05	9.65E-01	.80-1	4.76E-06	9.76E-01	4.64E-06
1.00	6.18E-06	9.87E-01	>1	6.18E-06	1.00E+00	6.18E-06
			Sum =	**2.50E-03**		**1.76E-04**
SE Initiating Event Frequency =				2.50E-03	CCDP =	7.05E-02
Seismically induced LOOP probability =				7.05E-02		
Seismically induced LOOP frequency =				1.76E-04		