Risk Assessment of Operational Events

Handbook

Volume 2 – External Events

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

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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
HCLPF	high confidence of low probability of failure
HEP	human error probability
HVAC	heating, ventilation and air conditioning
IE	initiating event
IE_{freq}	initiating event frequency
IMC	inspection manual chapter
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
PRA	probabilistic risk assessment
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 assessment
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 in accordance with Management Directive 8.3, "NRC Incident Investigation Program." 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 "P-302 – Risk Assessment in Event Evaluation" or has related experience.

1.4 Handbook Content

The revised handbook includes four volumes, designed to address Internal Events (Volume 1), External Events (Volume 2), SPAR Model Reviews (Volume 3), and Shutdown Events (Volume 4). 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 Modeling

  – Failure Modeling

  – Mission Time Modeling

  – Common-Cause Failure Modeling

  – Modeling Recovery and Repair Actions

  – Multi-Unit Considerations

  – Initiating Event Analyses

  – Human Reliability Analysis

  – Loss of Offsite Power Event Analysis

  – Support System Initiating Events

  Appendix A, "Road Map for Risk Analysis of Operational Events," provides a general overview on the risk analysis of initiating events and conditions in event assessment.

  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 shutdown operations.

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.

- ***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, 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

- ***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 NUREG/CR-3485, "PRA Review Manual," American Society of Mechanical Engineers (ASME) RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," and Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and experiences and lessons learned from SDP and ASP analyses.

  In addition, this volume summarizes key assumptions in a SPAR model and unresolved technical issues that may produce 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.

- ***Volume 4, Shutdown Events.*** Volume 4, "Shutdown Events," provides methods and practical guidance for modeling shutdown scenarios and quantifying their core damage frequency using SPAR models and SAPHIRE software. The current scope includes the following plant operating states for boiling-water reactors (BWRs) and pressurized-water reactors (PWRs): hot shutdown, cold shutdown, refueling outage, and mid-loop operations for PWRs.

1.5 Companion References to the Handbook

Guidance in this handbook often refers to other references, as applicable to the application. Documents referenced in this volume of the handbook are provided in Section 7.0 of this document.

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.

1.7 Questions, Comments, and Suggestions

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

Internal NRC staff and NRC contractors should contact:

- Volume 1, Internal Events
  - Christopher Hunter, 301-415-1394, Christopher.Hunter@nrc.gov
  - Michael Montecalvo, 301-415-1678, Michael.Montecalvo@nrc.gov
  - Don Marksberry, 301-415-1543, Don.Marksberry@nrc.gov
- Volume 2, External Events and Volume 4, Shutdown Events
  - Selim Sancaktar, 301-415-2391, Selim.Sancaktar@nrc.gov
  - Ching Ng, 301-415-8054, Ching.Ng@nrc.gov
- Volume 3, SPAR Model Reviews
  - Jeffery Wood, 301-415-0953, Jeffery.Wood@nrc.gov

External NRC stakeholders (e.g., public, licensees) should contact:

- All Handbook Volumes
  - Candace Spore, 301-415-8537, Candace.Spore@nrc.gov

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 standardized plant analysis risk (SPAR) models and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. This volume of the handbook complements Volume 1 for internal events.

- ***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 Significance Determination Process (SDP), as documented in Inspection Manual Chapter (IMC) 0609 Appendix F, “Fire Protection Significance Determination Process,” 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,” 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 may be used as an alternative to this volume of the RASP Handbook. This document also simplifies the detailed guidance in NUREG/CR-6850. The analysis process in the Handbook for Phase 3 Fire Protection SDP Analysis 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 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, and
  - Containment fires.

  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.

[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).

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 (IE_{freq}) of a fire scenario can be simply defined as

$$IE_{freq} = F_{fi} * SF * P_{ns}, \text{ where}$$

F_{fi} = fire ignition frequency
SF = Severity factor
P_{ns} = 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 IE_{freq}.* This is calculated using models discussed in Appendix 2A.
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).

Table 2-1. Example Summary of Fire Scenarios

	Name	Description	IEf_{req}	Equipment Lost	Initiating Event Caused	HEPs/ Other BEs Affected	New BEs/ 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	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 and B	LOMFW	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**	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	IE-FRI-FI8
9	FRI-FI10	Fire in MCR Bus 5 Switches Occurs	2.02E-04	Bus 5	FRI-MCR-E-0-07	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	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.

The guidance in the following table can be used to determine the 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.	TRANS
Turbine building fire that damages MFW or condenser system equipment	LOMFW
Spurious opening of RCS system valve(s) (e.g., power-operated relief valve (PORV), solenoid relief valve)	LOCA (size dependent on the number and size of valves)
Equipment damage (e.g., bus, transformer) leading to LOOP; self-induced LOOP by operators by fire procedures	LOOP
Reactor shutdown from remote shutdown panel after main control room evacuation	Make special event tree model

1. *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.
2. *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.

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:

$$CDF = IE_{freq} \times CCDP$$

The scenario IE_{freq} is already calculated in the earlier step, using Appendix 2A. 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 Event and Condition Assessment (ECA) Workspace 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.

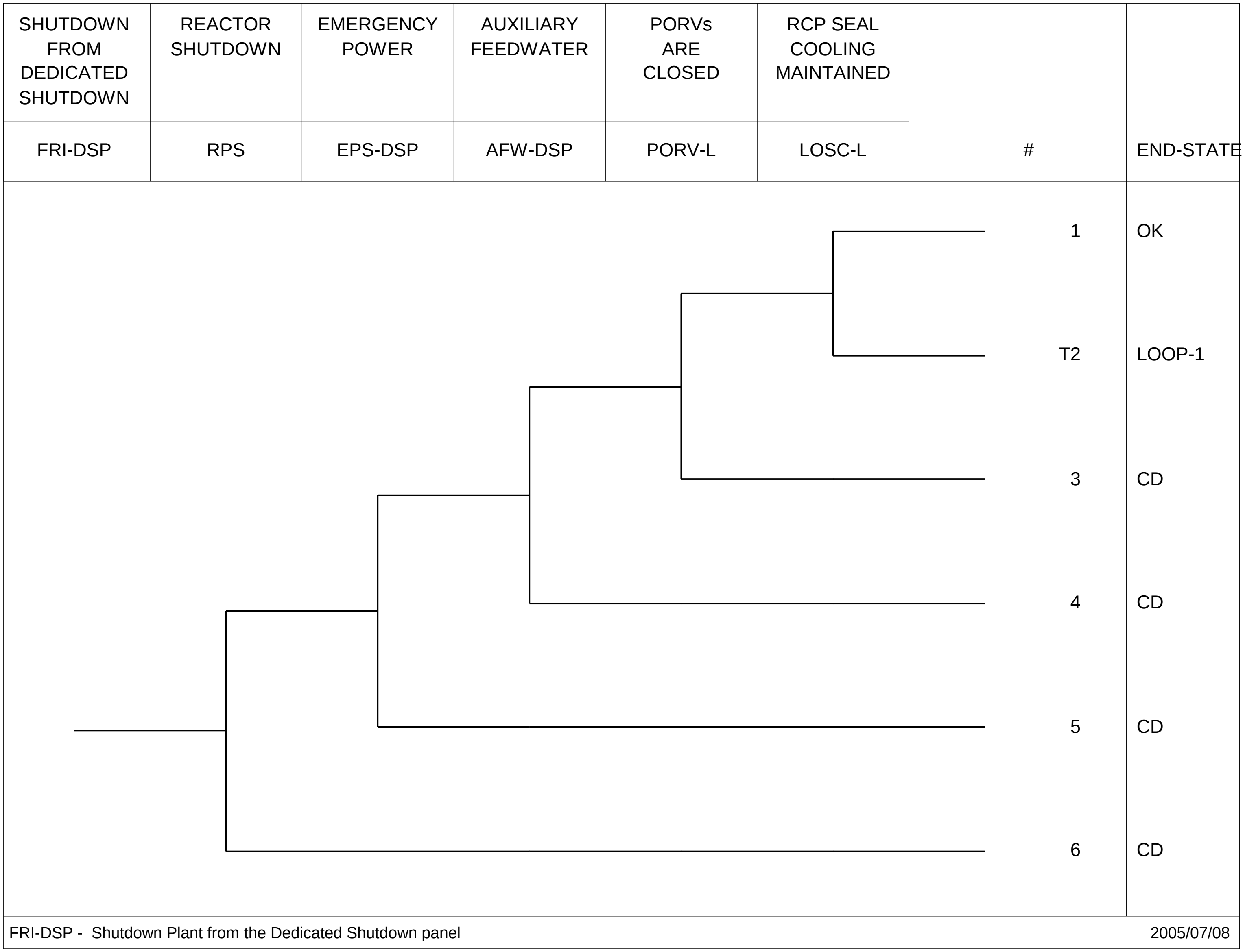

Figure 2-1. An Example New Event Tree Model

Table 2-2. An Example Calculation of Sequence CDFs

	Event	Description	IE_{freq}	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**

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 volt alternating-current (AC) 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 volt AC bus SPAR model event tree, with an IE_{freq} event frequency of 1.0, calculate the event CCDP using SAPHIRE as:

$CCDP = 4.3\times10^{-4}$

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.

Define base and condition case fire scenarios as in Figure 2-2 and Figure 2-3 (note that $F_{fi} = 3.25\times10^{-3}$ per year and SF =1).

Use SAPHIRE and SPAR to calculate the CCDPs for plant trips with loss of either one or two 4160 volt buses as 4×10^{-4} and 5×10^{-2}, respectively.

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

Detection (DET). 0.05 failure probability (NUREG/CR-6850)

Suppression (SUP). 0.05 failure probability (NUREG/CR-6850)

If the event tree nodes DET and SUP are successful, only the affected component is assumed damaged (and the CCDP = 4×10^{-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.

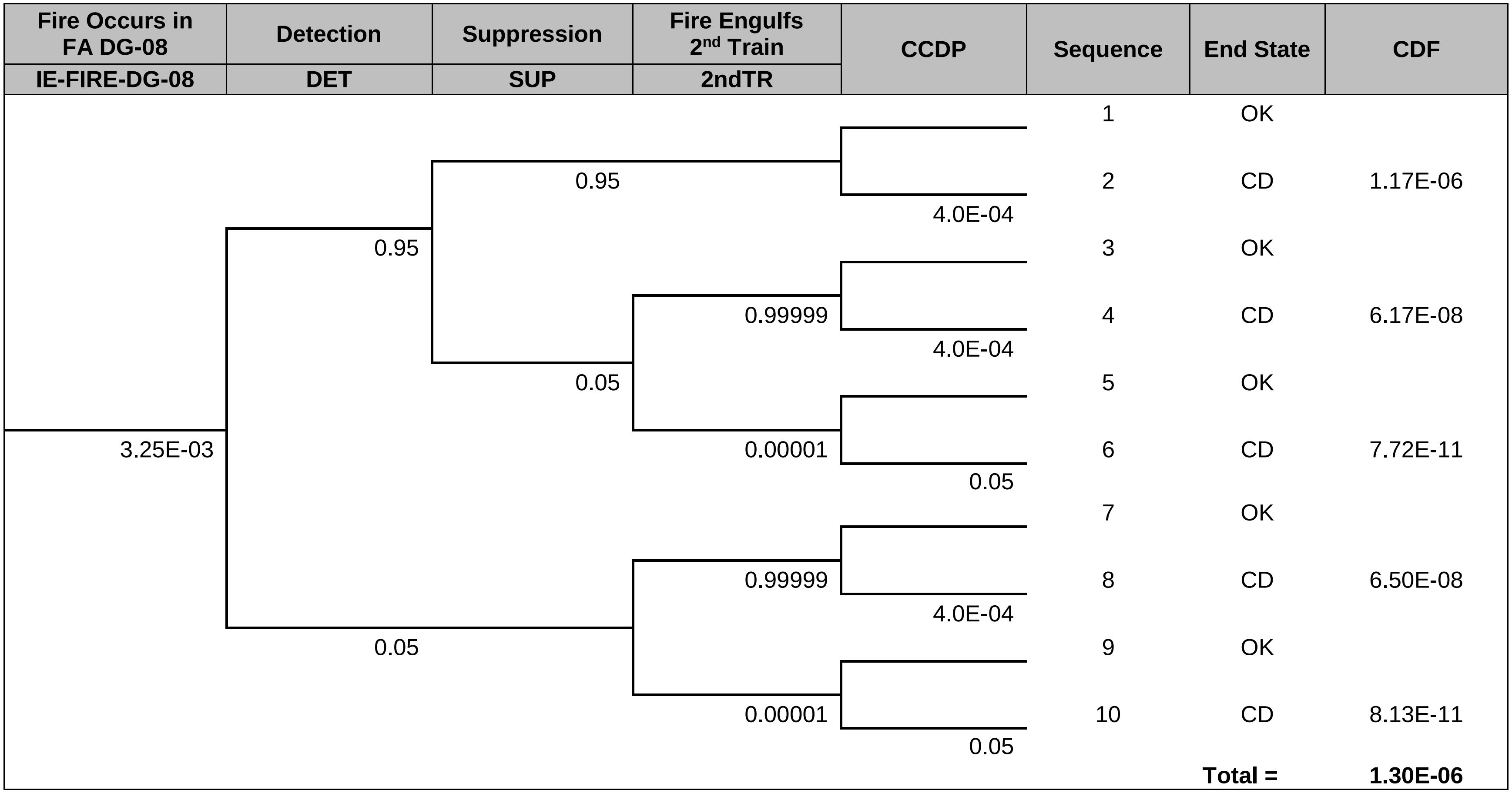

Figure 2-2. Fire in DG-08 Base Case (Example 2)

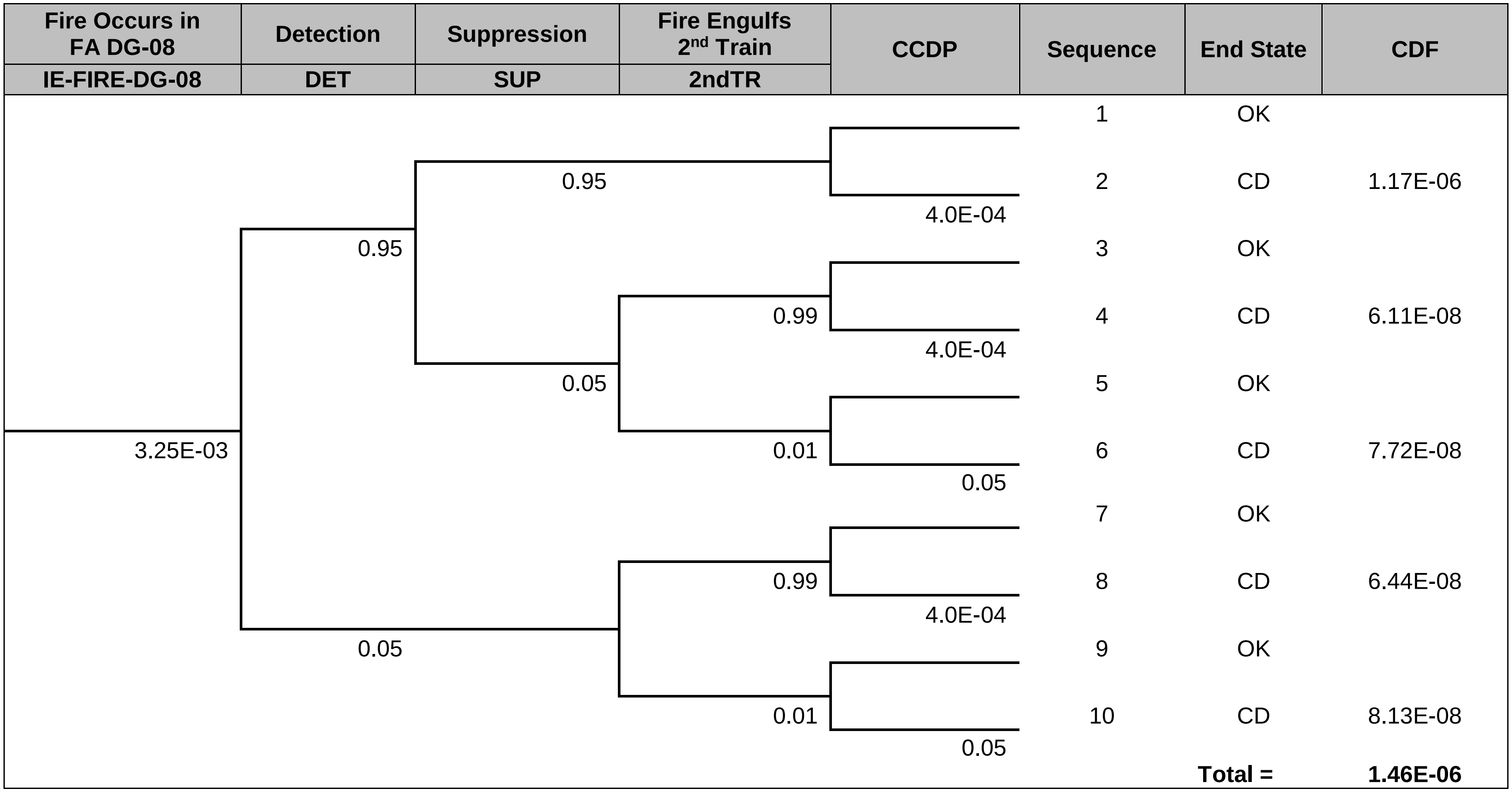

Figure 2-3. Fire in DG 08 with Plant Condition in Effect (Example 2)

Fire engulfs 2nd train (2ndTR). Fire engulfs second train (conditional upon unsuccessful detection or suppression). Effectively zero for the base case (1×10^{-5} 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, 1×10^{-2} is assigned, corresponding to 87 hours per 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 480 volt bus assumed unavailable, the CCDP is 4×10^{-4} (ECA Workspace output).

For the condition with both 480V buses unavailable, CCDP is calculated to be 4.8×10^{-2} (ECA Workspace output), 5×10^{-2} is used for calculations.

Base case CDF is calculated as shown in Figure 2-2 as 1.3×10^{-6} per year, as a sum of five fire sequences defined by the same figure.

The condition CDF is calculated as shown in Figure 2-3 as 1.46×10^{-6} per year.

The condition importance, defined as the difference between CDFs for the plant condition case and the base case, is calculated for a 1-year exposure time as

Condition Importance = (1.46×10^{-6} per year − 1.3×10^{-6} per year) * 1 year = 1.6×10^{-6}.

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 (P_{2ndTR}). The scenario frequency is:

$$IE_{freq} = F_{fi} * SF * P_{ns} * P_{2ndTR}$$

With SF = 1 and P_{ns} = 0.1 (approximate Boolean sum for failure of detection or suppression), the scenario frequency is:

$IE_{freq} = 3.25\times10^{-3} \times 1 \times 0.1 \times 1\times10^{-2} = 3.25\times10^{-6}$ per year

CCDP with loss of two trains is 5×10^{-2}. Thus the scenario CDF is:

$CDF = IE_{freq} \times CCDP = 3.25\times10^{-6}$ per year $\times 5\times10^{-2}$

$CDF = 1.6\times10^{-7}$ per year

This result matches that of the example in Section 2.3.2.

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 9.45×10^{-5} per year per 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 [note 1]	6.90E-03	Transient without non-safety systems
Fire in safety-related electrical cabinets	30 × 9.45E-05 [note 1]	2.83E-03	Transient without one safety train
MCR evacuation	[note 2]	[notes 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 and the Handbook for Phase 3 Fire Protection SDP Analysis. The Handbook for Phase 3 Fire Protection SDP Analysis also contains information about methods that can be used to perform Phase 3 SDP analysis of a sample of fire protection issues. Handbook for Phase 3 Fire Protection SDP Analysis is a specific application of those methods detailed in NUREG/CR-6850.

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 probabilistic risk assessment (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 2-4 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 Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. Also see the example in Section 2.3.3 where a shortcut is used in lieu of developing an event tree.

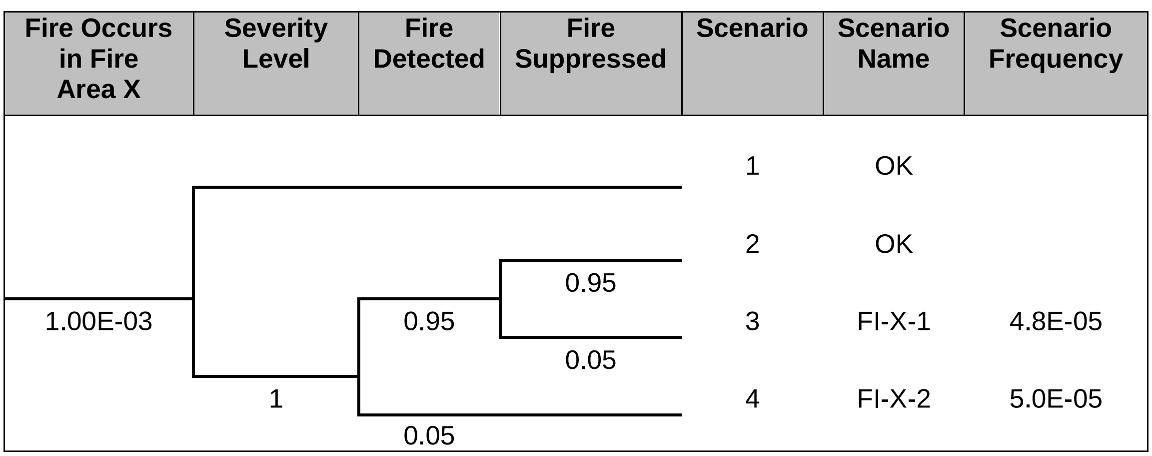

Figure 2-4. Example Event Tree Model Showing Fire Scenario Definitions

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.

- ***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
  - Appendix 2B: Generic Fire Ignition Frequencies
  - Appendix 2C: Severity Factors Data
  - Appendix 2D: Detection Failure Data
  - Appendix 2E: Suppression Failure Data
  - Appendix 2F: Spurious Actuations (due to hot shorts) Probabilities
  - Appendix 2G: Operator Actions
  - Appendix2H: 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 Appendix2B.

- ***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 core damage frequency (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 2-4 is shown in Figure 2-5. The reverse propagation from Y into X must also be modeled in a similar manner.

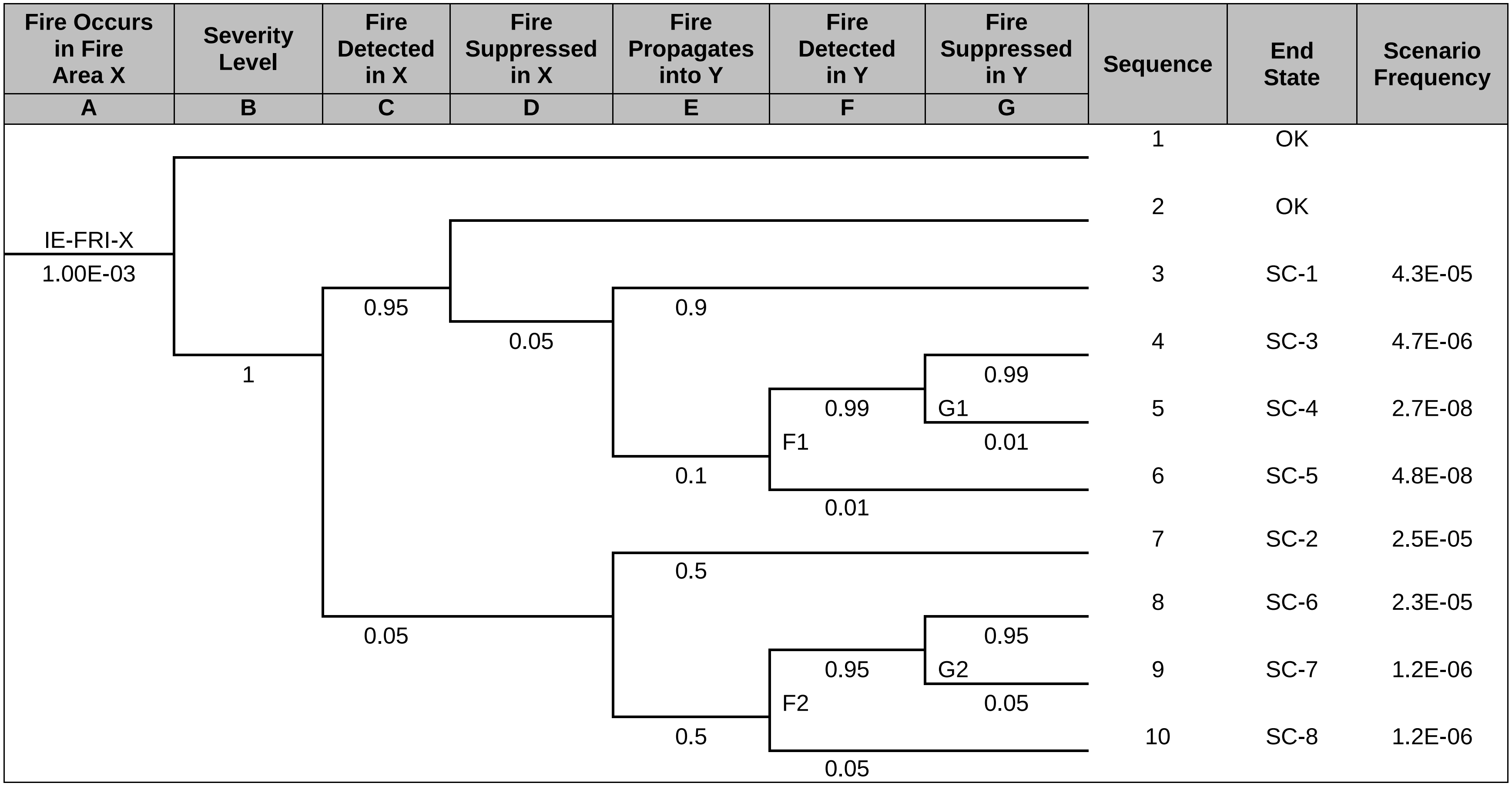

Figure 2-5. An Example Event Tree Model with Possible Propagation

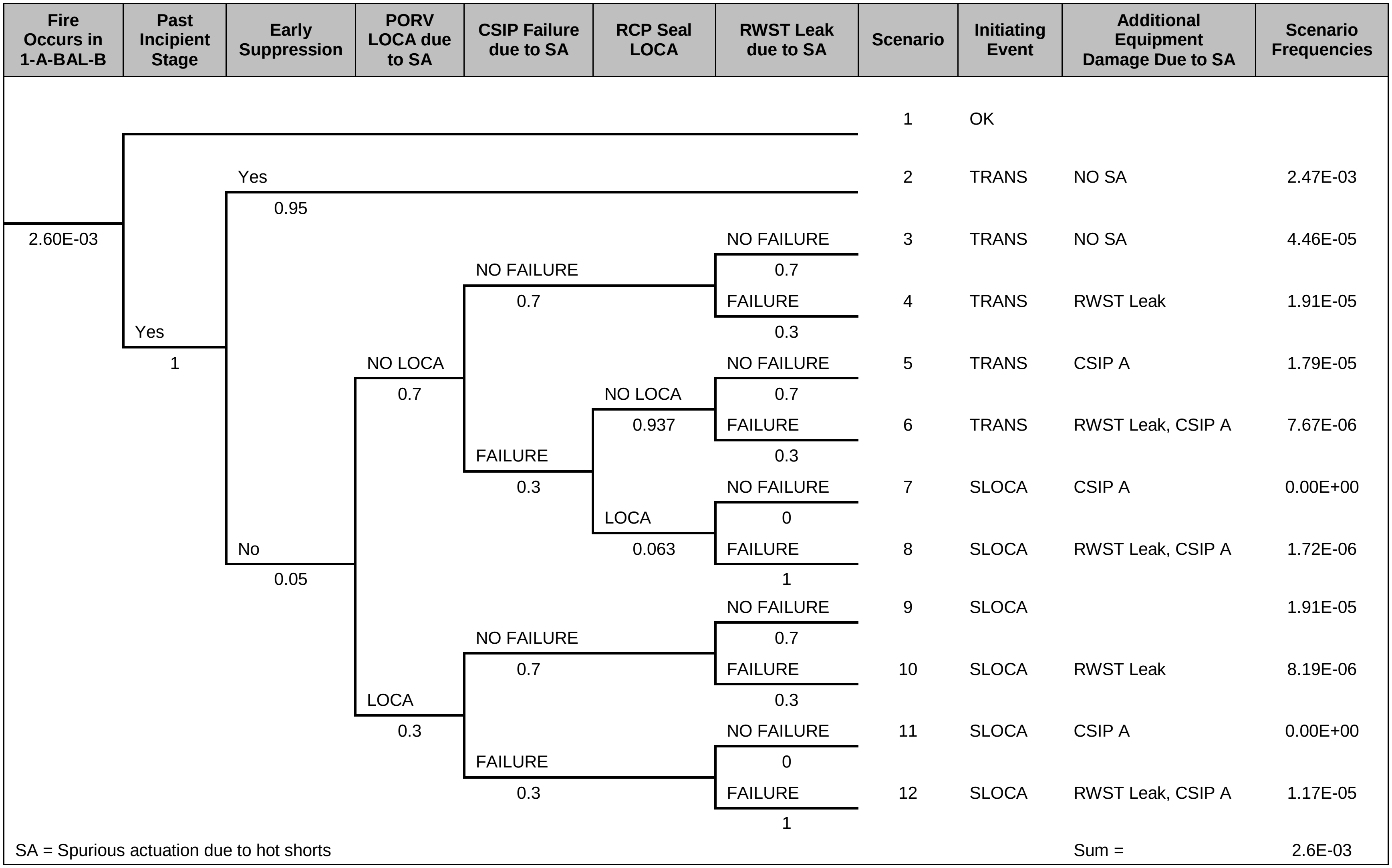

Figure 2-6. An Example Event Tree Model with Possible Spurious Actuations Due to Hot Shorts

- ***Example 1.*** From the example depicted in Figure 2-5, 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 5×10^{-2}.

    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 5×10^{-2}.

    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 (5×10^{-2}) 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 (5×10^{-2} x $0.1 = 5 \times 10^{-3} \sim 1 \times 10^{-2}$) is the product of random unavailability of the automatic fire detection system for fire area Y (5×10^{-2}) 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 2-5 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 feet 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 (Table 11-3) are given in Table 2-3. Note that these probabilities do not represent the failure of a barrier given a challenge by a particular fire hazard.

Table 2-3. Random Barrier Failure

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 2-6. 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 loss-of-coolant accident (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 main control room (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, and
- 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, and
- 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.

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 (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 2-4.

- ***Plant Area-Based Ignition Frequencies.*** Use generic fire area frequencies as provided in Table 2-5. 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 RES/OERAB/S02-01, "Fire Events – Update of U.S. Operating Experience, 1986–1999."

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.

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) × 1 year (duration) = 5.0×10^{-3}

Switchgear room B fire frequency (F_{iB}) = F_i ÷ 2 = 2.5×10^{-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 NUREG/CR-6850).

Table 2-4. Fire Frequency Bins and Generic Frequencies from NUREG/CR 6850 (Table 6 1)

ID	Location	Ignition Source (Equipment Type)	Mode	Generic Frequency (per reactor-year)
1	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	Miscellaneous Hydrogen Fires	All	2.50E-03
20	Plant-Wide Components	Off-gas/Hydrogen 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**):
See Appendix M for a description of high-energy arcing fault (HEAF) fires.
See Section 6.5.6.
The event should be considered either as an electrical or oil fire, whichever yields the worst consequences.

Table 2-5. Fire Ignition Frequencies for Power Operation

Plant Location	No. of Fires	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:
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.
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.
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.
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 1 year, the frequency will be multiplied by the fractional year.

Appendix 2C Severity Factors Data

Be cautious in assigning severity factors other than one, 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 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 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 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 = 5×10^{-2}

Source: NUREG/CR-6850, Appendix P

See also Figure P-4 of NUREG/CR-6850 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 2-6, below.

Table 2-6. Generic Failure Probabilities of Suppression

Fixed Suppression System	Unavailability
Carbon dioxide	4×10^{-2}
Halon system	5×10^{-2}
Wet pipe sprinkler systems	2×10^{-2}
Deluge or pre-action sprinkler systems	5×10^{-2}

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

$$P_{ms} = e^{(-\lambda - \Delta T)}, \text{ where}$$

ΔT (minutes) = (Time to target damage) − (Response time of the brigade) − (Time to detection)

Appendix P of NUREG/CR-6850 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 2-7 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 that contain two fixed ignition sources, the more conservative suppression curve (λ) should be utilized. For fire areas which contain many ignition sources, the "all fires" suppression curve (λ) should be utilized. The control room type of fire should be applied to evaluate the control room. Exceptions should be justified.

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:

ΔT = 30 minutes − 7 minutes− 3 minutes = 20 minutes

Then, the probability of manual suppression failure before the target is damaged is 9×10^{-2}.

Table 2-7. Manual Suppression Probability per Unit Time (Lambda) and Failure

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

Note:
Minimum $P_{ms} = 1 \times 10^{-3}$

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. The following tables, taken from NUREG/CR-6850, are provided below for convenience. See table notes following the last table.

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

Table 2-8. Failure Mode Probability Estimates Given Cable Damage Thermoset

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	

Notes:
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.

Table 2-9. Failure Mode Probability Estimates Given Cable Damage

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	

Notes:
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.

Table 2-10. Failure Mode Probability Estimates Given Cable Damage

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	

Notes:
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.

Table 2-11. Failure Mode Probability Estimates Given Cable Damage

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	

Notes:
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.

Table 2-12. Failure Mode Probability Estimates Given Cable Damage

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

Notes:
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.

Table 2-13. Screening Criteria to Assess the Ignition and Damage

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 (Tables 2-8 to 2-12), 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:

$$P_{\text{Component Failure}} = (P_{\text{Failure Cable A}}) + (P_{\text{Failure Cable B}}) - (P_{\text{Failure Cable A}})\,(P_{\text{Failure Cable B}})$$

Appendix 2G Operator Actions

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

- 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.
- 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 discusses the smoke damage due to a fire event. It concludes that the current state of knowledge cannot support detailed quantitative assessment.

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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 Volume 1 of this handbook.

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:

- The user has hands-on experience with the SAPHIRE code, and
- 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 that 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:

- They can fail multiple components in the same area, and
- 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:

- ***Step 1: Define Flooding Scenarios that Lead to Core Damage.*** For this purpose, an event tree logic structure such as the one given in Figure 3-1 may be used. Using such a modeling structure, calculate scenario frequencies. Definition of a flooding scenario is discussed in Section 3.2.1.

- ***Step 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:

$$\text{CDF} = \text{Scenario Frequency} \times \text{CCDP}$$

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 firewater actuation by steam or fire, in the following systems:

- Fire water system,
- Emergency service water (ESW)/component cooling water (CCW) system, and
- 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, and

– 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 Table 3-4 and Table 3-6. 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 3-1. 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 3-2. 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 initiating event frequency (IE_{freq}).* This is calculated using models such as the one discussed in Figure 3-1.
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:

$$CDF = IE_{freq} * CCDP$$

The scenario frequency IE_{freq} 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 Event and Condition Assessment (ECA) Workspace 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	IE_{freq}	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 "A Feasibility and Demonstration Study – Incorporating External Events into SPAR Models," (*Not Publicly Available,* ADAMS Accession No. ML15174A003) 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 1.0×10^{-2}. If this failure occurs, the AFW pump supported by ruptured SWS train will fail.

The event importance can be calculated as:

$$\text{Importance} = (1 - 1.0\times10^{-2}) \times CCDP_1 + 1.0\times10^{-2} \times CCDP_2$$

where $CCDP_1$ and $CCDP_2$ are the conditional core damage probabilities with or without success of isolation, respectively.

If the isolation is successful, use the existing SPAR model transient event tree, with an IE_{freq} of 1.0. Also fail the MFW system and the one train of SWS. Calculate event $CCDP_1$ using SAPHIRE as:

$CCDP_1 = 1.0\times10^{-4}$

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,

$CCDP_2 = 1.0\times10^{-4}$

The event importance is 1.0×10^{-4}. (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 volt AC 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 $CCDP_2$ value of:

$CCDP_2 = 1.0\times10^{-3}$

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

$$\text{Importance} = 0.99 \times 1.0{\times}10^{-4} + 1.0{\times}10^{-2} \times 1.0{\times}10^{-3}$$

Therefore, the event importance is calculated to be = $1.1{\times}10^{-4}$. (Based on the above discussion, with $CCDP_1$ reduced by at least a factor of 100, this value will be no greater than $1.1{\times}10^{-5}$)

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 volt AC emergency buses inoperable (P_{pr}). 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 (HEP_{iso}). The total IE_{freq} from different potential large flood sources in area X is calculated to be $5{\times}10^{-3}$ per 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 $1.0{\times}10^{-6}$ 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:

$$\text{Importance} = \text{exposure time} \times IEf_{req} + (CCDP_{cond} - CCDP_{base})$$

$$\text{Importance} = 0.25 \text{ year} \times 5{\times}10^{-3} \text{ per year} \times (0.2 - 1.0{\times}10^{-6})$$

$$\text{Importance} = 2.5{\times}10^{-4}$$

Note that HEP_{iso} and Ppr both equal 1.0 in this example. Thus, the scenario frequency is equal to flood IE_{freq} 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 IE_{freq} 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 (F_1),
2. Rupture of the piping and components in the system (F_2), and
3. Maintenance errors (F_3).

$$\text{IE-FLI-X} = F_1 + F_2 + F_3$$

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:

$F1 = 4.5\times10^{-5} \times 4 = 1.80\times10^{-4}$ per year,

where the expansion joint failure is taken from Table 3-4.

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

$$F_2 = F_{piping} + F_{valves}$$

$F2 = 8760$ hours/year × ((10 pipe segments) × (1.39×10^{-10} per pipe segment-hour) + (4 valves) × (4.0×10^{-10} per valve-hour)) × 0.5

$F_{piping} = 1.31\times10^{-5}$ per year,

where data is taken from Table 3-6 (generic PWR pipe rupture in "Other Safety-Related Systems" for $D \geq 6$ inches), Table 3-4 (valve non-PCS rupture) and Table 3-5 (0.5 for large failure given a break in large piping ($D \geq 6$ inches).

Flooding events initiated by maintenance on the circulating water system are considered negligible contributors to the overall IE_{freq} (assume an upper bound F_3 for completeness):

$F3 = 1.0\times10^{-6}$ per year

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

$IE\text{-}FLI\text{-}X = 1.8\times10^{-4} + 1.31\times10^{-5} + 1.0\times10^{-6}$
$IE\text{-}FLI\text{-}X = 1.9\times10^{-4}$ per year

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. 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 IE_{freq} of a limiting FLI scenario can be defined as

$$IE_{freq} = F_{if} * HEP_{iso} * P_{pr}, \text{ where}$$

F_{if} = FLI frequency
HEP_{iso} = Failure to terminate the flood source
P_{pr} = 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 3-2. 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 3-3 shows another table where the scenario information is tabulated for CCDP calculation.

3A-2 Initiating Event Frequency Data

Table 3-4 provides pipe and other equipment rupture frequencies assembled from different sources.

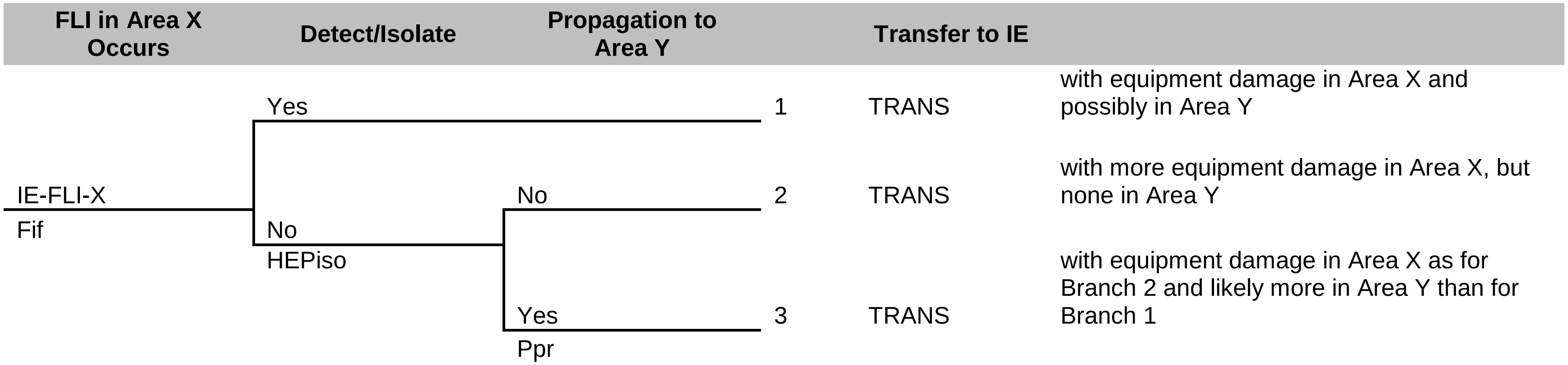
In medium and large diameter pipes, the breaks of smaller equivalent sizes can occur. The fraction of smaller sizes of breaks, given a failure in a larger pipe, can be calculated by using data in Table 3-5.

A more recent data set for pipe failures by system and reactor type is also given in Table 3-6 in units of per hour-per segment. Use of this data requires knowing the number of segments in question.

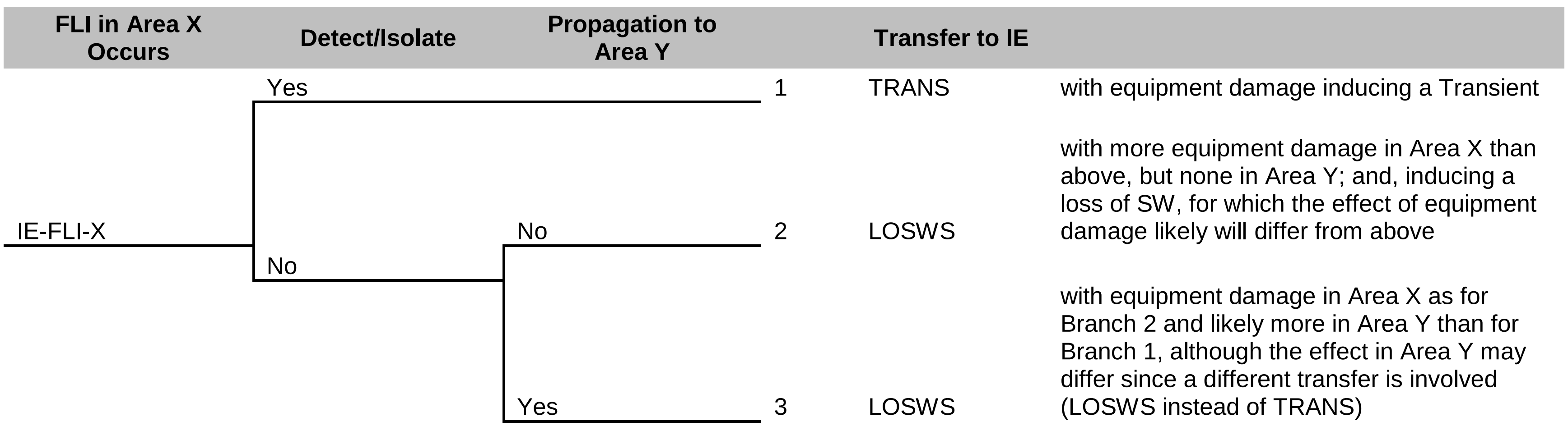
Finally, the initiating event frequencies of steam and feedline breaks are given in Table 3-7.

3A-3 Quantification of Internal Flooding Initiating Event Frequencies

To calculate flooding initiating event frequencies, data from Tables 3A-3 through 3A-6 may be used. This requires knowing the number of segments or feet 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-HEP_{iso})
Frequency of Scenario 2 = Fif * HEP_{iso}* (1-P_{pr})
Frequency of Scenario 3 = Fif * HEP_{iso}* P_{pr}

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

Table 3-2. Example Matrix Defining Internal Flooding Scenarios

	Name	Description	IE_{freq}	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	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	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	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	IE-FLI-FL4
5	FLI-FL5	Relay Room Potable Water Flood	1.50E-04	None	IE-TRANS	None	IE-FLI-FL5
6	FLI-FL6	Control Rod Drive Equipment Room Service Water Flood	1.50E-04	None	IE-TRANS	None	IE-FLI-FL6
		Sum =	**1.50E-03**				

Table 3-3. Example Summary of a Plant X Turbine Building Flood Scenario

IE Name	Description	Flood Damage	Detection/Isolation Means	Failed Gate or Component Basic Events[1]
CI06B	Rupture of an Inlet Condenser Expansion Joint in TU-22-1	<u>Propagate</u>: TU-94 TU-95B-1 <u>Damage</u>: 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 Plant Equipment Water Pump 1A Plant Equipment Water Pump 1B MCC-32D MCC-42B MCC-42D AOV PW-52	<u>Detect</u>: Reactor Trip due to Loss of Condenser Vacuum <u>Isolate</u>: Trip both Circulating Water Pumps	<u>Initiating Event:</u> IE-CI06B <u>Failed BEs:</u> 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 3-4. Data for Calculating Internal Flooding Initiating Event Frequencies

Component Type	Rupture/Leakage [note 4]	Rate (/hour)	Error Factor [note 2]
Generic Piping (including elbows)	Leakage	3.0E-09 /hour-foot	10
	Non-PCS Rupture	1.2.0E-10 /hour-foot	30
	PCS Rupture	3.0E-11 /hour-foot	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 /year	

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 EGG-SSRE-9639, "Component External Leakage and Rupture Frequency Estimates."
[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 3-5.
[4] Leakage <50 gpm; rupture ≥ 50 gpm.

Table 3-5. Conditional Probability of Small,

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 EPRI TR-102266, "Pipe Failure Study Update." Breaks include all ruptures.

Table 3-6. Rupture Failure Rates for Generic System Groups for Piping [note 1]

System	Failure rate (per Section-hour) for Pipe Size Groups [note 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 and Condensate Systems	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 >50 gpm. Use together with Table 3-5 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.
Data from EPRI TR-102266.

Table 3-7. 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 1-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 1-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 NUREG/CR-5750, “Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995,” 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 Volume 1 of this handbook.

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:

– The user has hands-on experience with SAPHIRE, and

– 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 that 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 seismic margins analysis (SMA). See Section 4.2.8 for additional information.

4.2 Seismic Event Scenario Definition

4.2.1 Minimum Input Requirements

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

- ***Seismic hazard vector (frequencies of seismic events).*** The seismic hazard vectors for all 61 U.S. nuclear power plants are obtained from licensees' submittals as part of the effort to address Near- Term Task Force (NTTF) Recommendation 2.1 in 2014 and

2015. These seismic hazard vectors are given in Appendix 4A. Uncertainty information for each of the seismic hazard vector can also be obtained from the licensees' submittals.

- ***Seismic fragilities of major structures, systems, and components (SSCs).*** Seismic fragilities can be found in plants with SPRAs, and some of this information may be available for plants with seismic margins analyses. Section 4.2.4 provided additional discussion with respect to the usage of a more extensive collection of SSC seismic fragilities, which contains proprietary information that is available in an NRC document (non-public) with ADAMS Accession No ML071220070.

- ***An event tree model representing the seismic sequences.*** Such an event tree model is provided as a default in a later section.

4.2.2 Example Seismic Hazard Vector

The example seismic hazard vector in Table 4-1 is taken from NUREG-1488, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," and is presented graphically in Figure 4-1:

Table 4-1. Example Seismic Hazard Vector

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

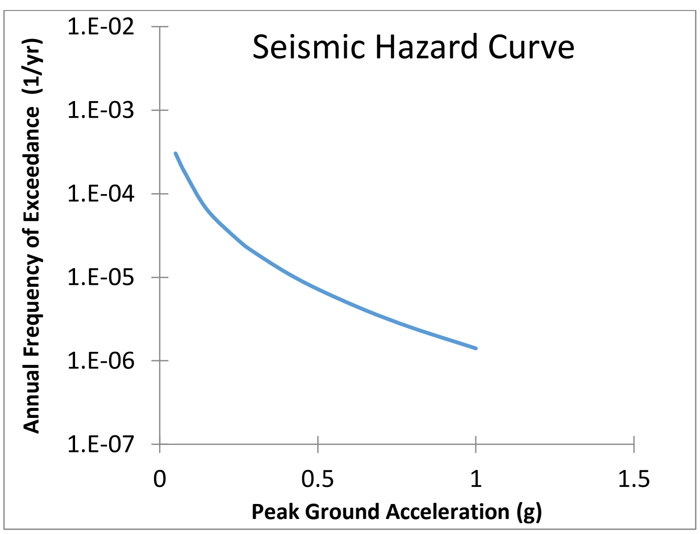


Figure 4-1. Example Seismic Hazard Vector

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 3.04×10^{-4} per 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 to five seismic bins as defined below, unless plant-specific considerations require more bins.

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

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

For 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 (0.05g–0.3g)	0.122g
BIN-2 (0.3g–0.5g)	0.387g
BIN-3 (>0.5g)	0.707g

The frequency of each bin, which is calculated as the difference of the frequencies of two bin range limits, is calculated as shown in Table 4-2.

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)	0.122	2.84E-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)	0.387	1.26E-05
0.40	1.141E-05			
0.50	7.212E-06	3 (>0.5g)	0.707	7.21E-06
0.65	4.043E-06			
0.80	2.474E-06			
1.00	1.409E-06			
			Sum =	**3.04E-04**

The three seismic bins chosen here follow, “A Feasibility and Demonstration Study – Incorporating External Events into SPAR Models,” (*Not Publicly Available,* ADAMS Accession No. ML15174A003) for Limerick. The first bin is driven by seismically induced loss-of-offsite power (LOOP) events. The second bin captures other modeled events (small loss-of-coolant accident (SLOCA), large loss-of-coolant accident (LLOCA), LOOP, and structural failures). The third bin is driven by the seismic failure of major structures, leading to direct core damage.

A larger number of bins can be readily introduced into the SPAR models without significantly affecting their quantification times. The current SPAR-AHZ models use five seismic bins. A larger number of bins may be appropriate for the sites to the West of the Rocky Mountains. 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).

After the next step (Section 4.2.4) is completed and if plant-specific low fragility SSCs are identified, redefinition of the seismic event categories (i.e., number of bins, or the bin ranges) may be required to provide better resolution at the lower g level or at the frequency range that is of interest.

4.2.4 SSC Seismic Fragilities

The fragilities of the major SSCs must be obtained to calculate seismic failure probabilities. Preferably, the analyst should use the plant-specific fragility value if one exists for the plant. In the absence of plant-specific SSC fragilities, fragility values from power plants of similar vintage may be used as surrogates by NRC risk analysts when obtaining risk insights for operational events via the SDP, the ASP Program, Notice of Enforcement Discretion (NOED) evaluations, and event assessments under the Management Directive 8.3, “NRC Incident Investigation Program.” For plant-specific risk-informed licensing applications, the fragility values should be developed by meeting the appropriate Standard and guidance.

A more extensive collection of SSC seismic fragilities is available in an NRC document (*Not Publicly Available*, ADAMS Accession No. ML071220070), which contains proprietary information. Many of the values in the collection are obtained from the Individual Plant Examination of External Events (IPEEE) vintage and older compilations. In the case that plant-specific fragilities are not available, the analyst should review this collection along with more recent results to select appropriate surrogate values for the situation being analyzed. In addition, as seen from the collection, the recorded fragility values may have a wide range for a given component. For example, the median capacity values for a RHR pump ranges from 0.62g to 2.00g. Also, there have been some significant revisions of fragility guidance as described in the following:

– Electric Power Research Institute (EPRI) TR-103959, “Methodology for Developing Seismic Fragilities,”
– EPRI 1002988, “Seismic Fragility Application Guide,”
– EPRI 1002989, “Seismic Probabilistic Risk Assessment Implementation Guide,” and
– EPRI 1019200, “Seismic Fragility Application Guide Update.”.

A number of seismic PRAs are being performed in connection with the implementation of the NTTF Recommendation 2.1. These PRAs will provide a more current estimates of fragilities using the recent guidance.

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

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

High confidence of low probability of failure (HCLPF) capacity is a term that is commonly-used in a SPRA or SMA. HCLPF capacity is a measure of seismic ruggedness and it 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 of a single SSC or of an ensemble of them. The HCLPF value is calculated by the equation:

$$HCLPF = a_m\ e^{(-1.645(\beta r + \beta u))}$$

EPRI 1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," discusses a methodology that uses HCLPF values in connection with generic β values to develop fragilities. However, a caution needs to be exercised in using such an approach, as undue large β values can be nonconservative. As discussed in EPRI 1025287, those important fragilities should be checked by using the method separation of variable.

The fragilities of key SSCs can be ordered from lowest to highest in a table. The lower fragilities will determine the number of bins and their ranges while 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.)

Generally, ceramic insulators have one of the lowest median capacities among the SSCs modeled in a seismic PRA. Therefore, the failure of ceramic insulator is assumed to trigger the occurrence of LOOP following a seismic event in many plants.

As previously discussed, seismic event categories (i.e., number of bins or bin ranges) definitions may be revisited/revised after SSC fragilities are modeled.

Table 4-3 and Table 4-4 show some examples of how SSC fragilities are used in two SPAR-AHZ models. These values should not be taken as NRC staff-endorsed values and the values for a specific situation should be determined using the collection of data and other relevant information as described above. Other appropriate NRC endorsed or NRC guidance,

Table 4-3. SSC Fragilities and Their Treatment in SPAR AHZ

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	

SSC Description	Median Capacity (g)	β_c OR β_r	β_u	SSC Failure Probability	Comment	HCLPF
Turbine Building	1.1	0.3	3.50E-01	1.69E-01	CD	
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	May be assumed failed in SPRA due to low fragility.					
CST	Assumed failed due to low fragility in SPRA. SWS is credited as alternate.					
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 NUREG/CR-4840, "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150" 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 three buildings (Containment, Aux., Turbine)
Plant-specific SPRA assignments are used when available

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

	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	
		0.3	0.3	0.35	9.69E-01	LOOP-EQ-3	

	SSC Description	Median Capacity (g)	β_r	β_u	SSC Failure Probability	Comment	HCLPF
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 NUREG/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

and applicable codes and standards should be considered for specific regulatory applications. Please refer to Section 1.2 regarding the scope of methods and guidance in this handbook.

The following list illustrates the candidate SSCs that may need to be modeled in 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 Offsite 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 Figure 4-2, Figure 4-3, and Figure 4-4.

The example SPRA also modeled medium loss-of-coolant accident (MLOCA) event, but its CDF contribution was not dominant. Therefore, the MLOCA event is left out of the current SPAR-AHZ models. 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.

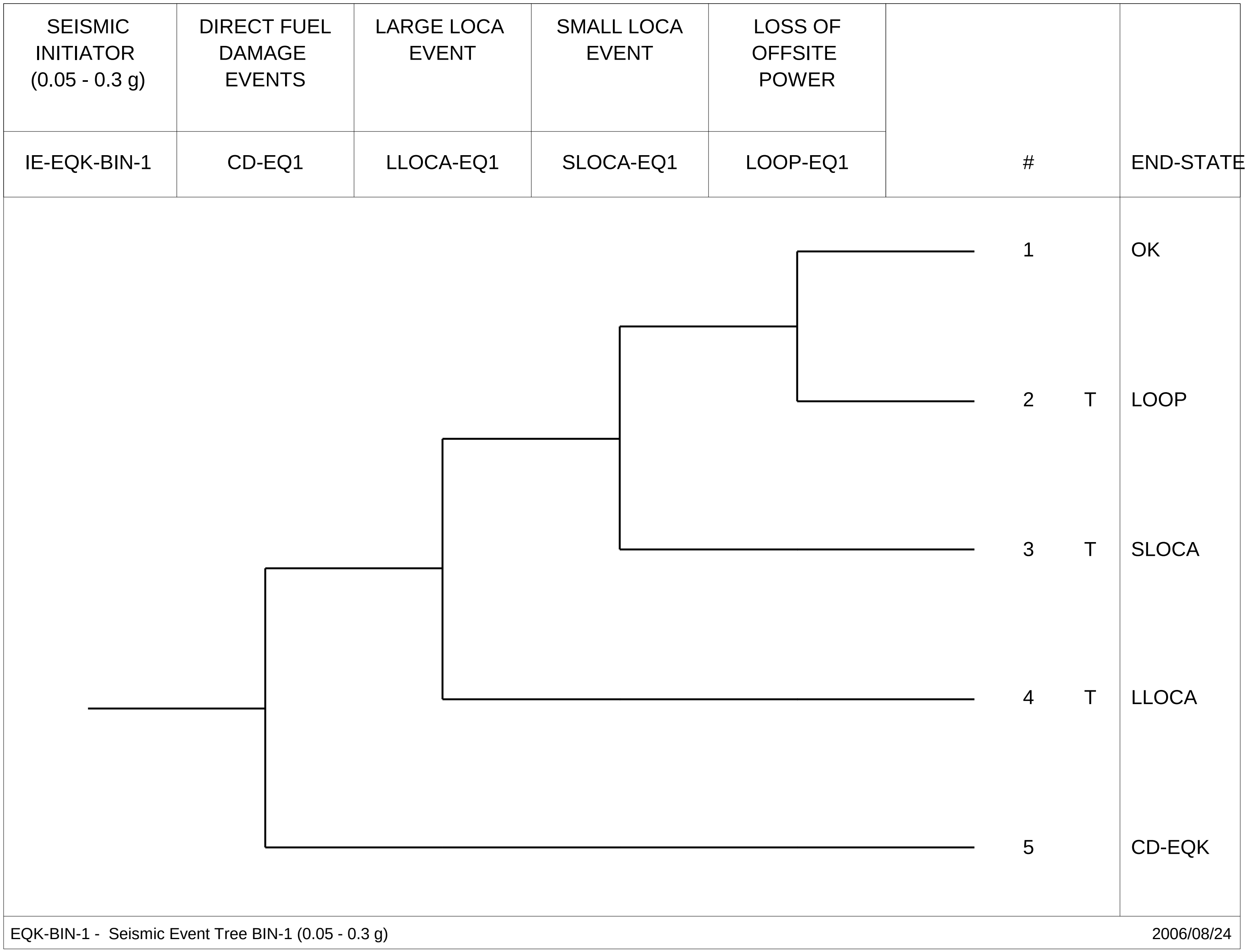


Figure 4-2. Seismic Event BIN 1 Event Tree

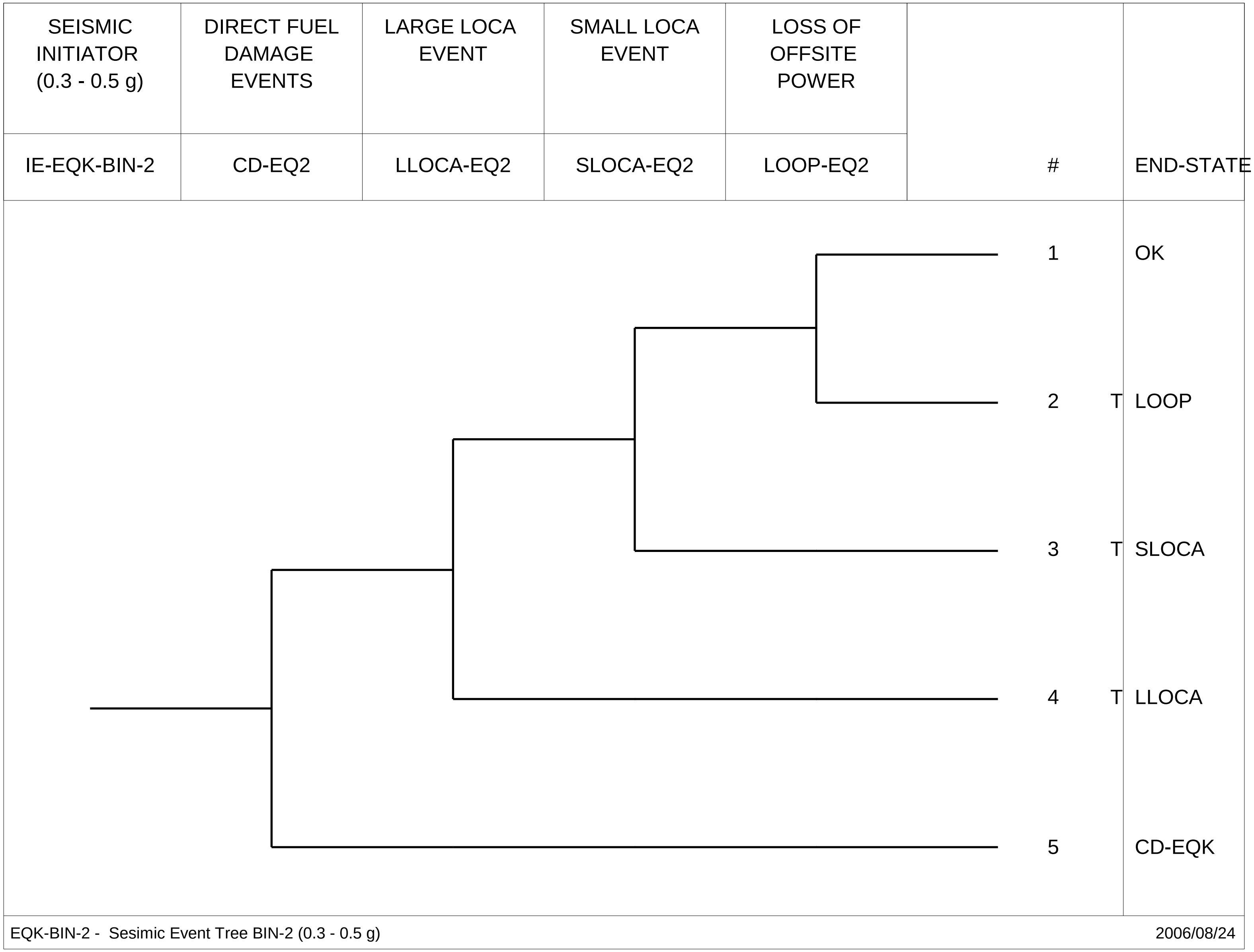


Figure 4-3. Seismic Event BIN 2 Event Tree

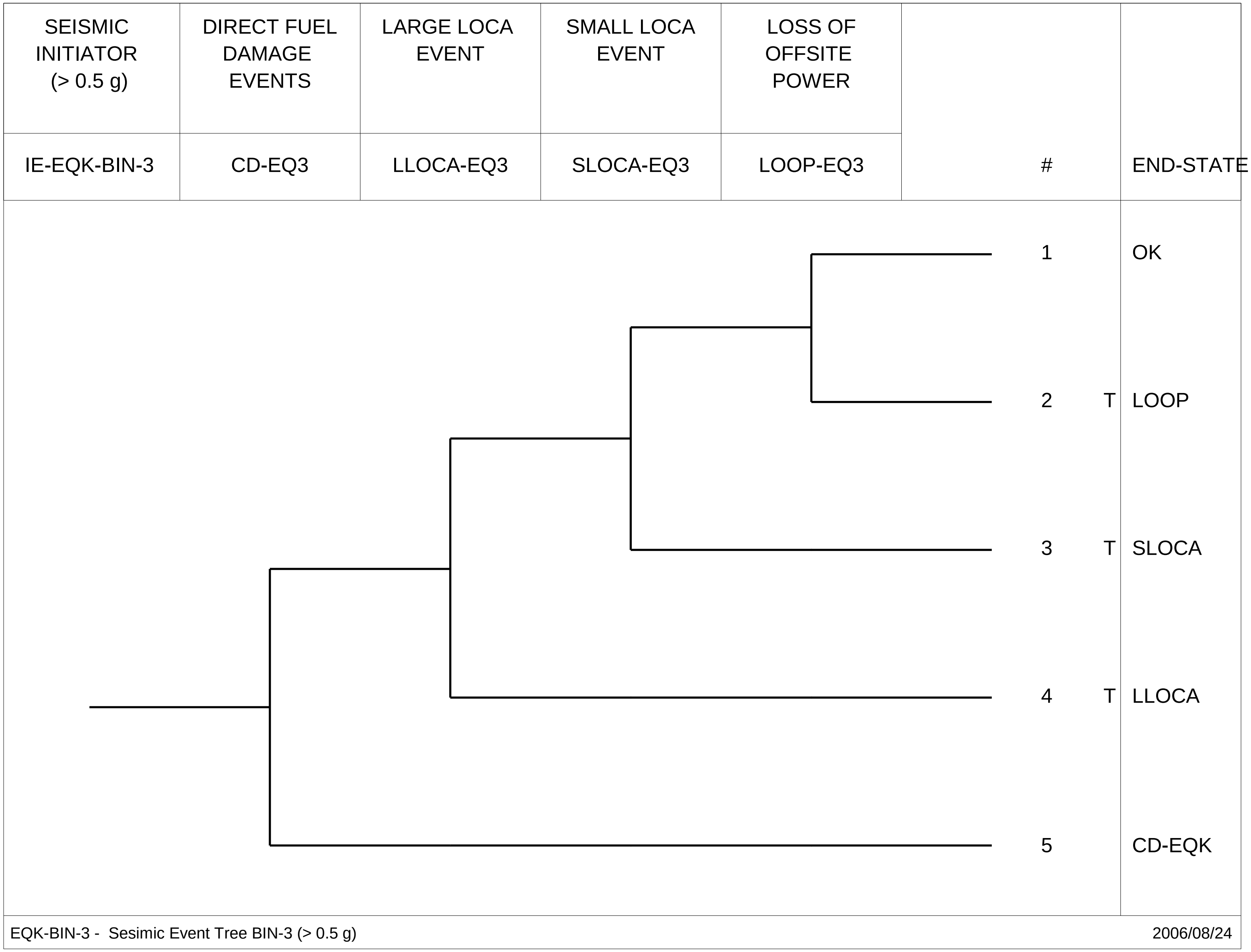


Figure 4-4. Seismic Event BIN 3 Event Tree

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 4-5 shows an example for a front line system in which 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 4-6.

Figure 4-7, Figure 4-8, and Figure 4-9 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 the associated house event (such as “EQ-BIN-1-OCCURS”) is set to TRUE

4.2.7 New Basic Events

Four types of new basic events are introduced in SPAR-AHZ models:

1. Initiating event frequencies,
2. Basic events,
3. Flags – house events, and
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 SPAR-AHZ models are given in Table 4-5.

For some basic events represented by the FT value, the process flags in the SAPHIRE “Edit Basic Event” dialog 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.

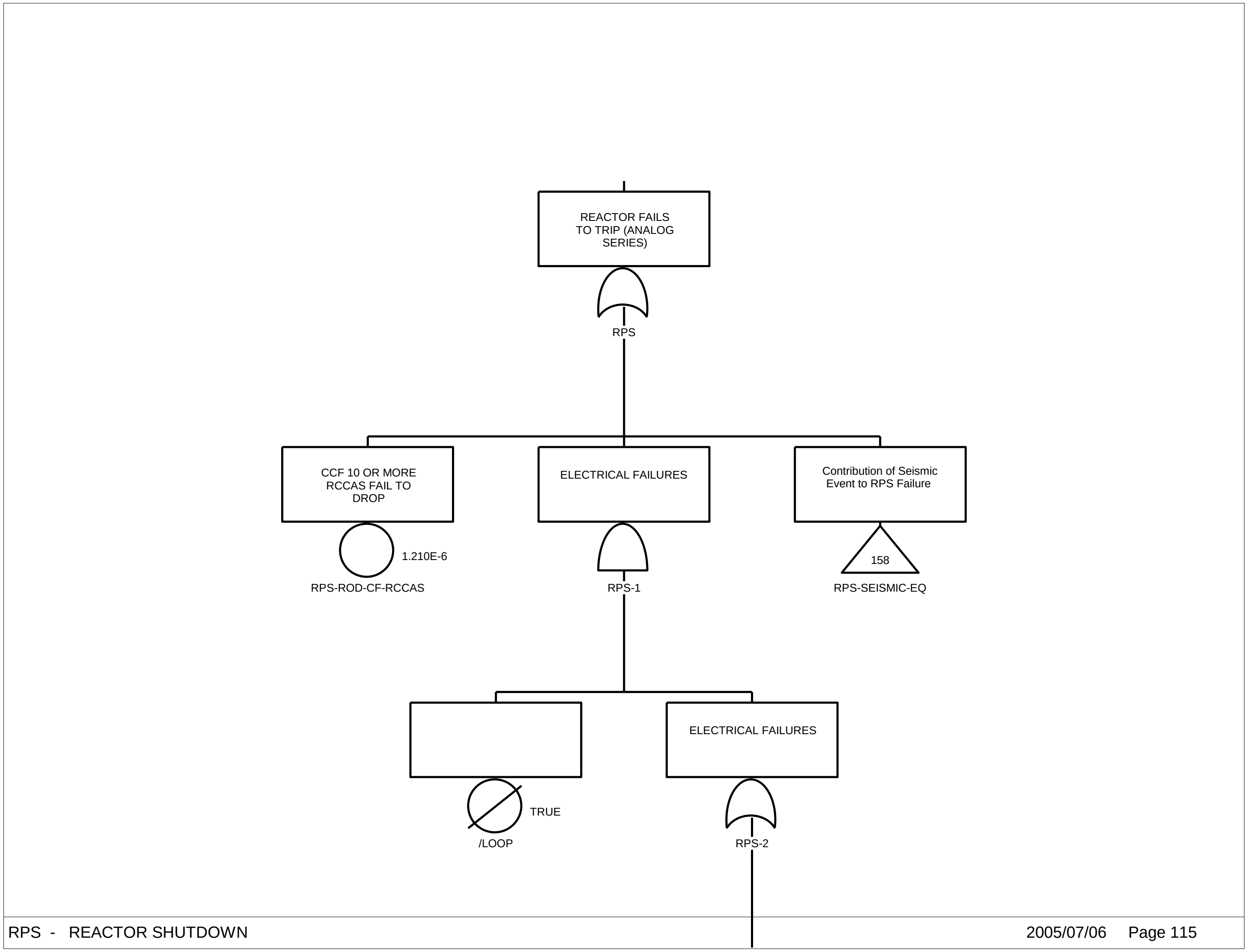


Figure 4-5. RPS Fault Tree (partial top showing introduction of seismic faults)

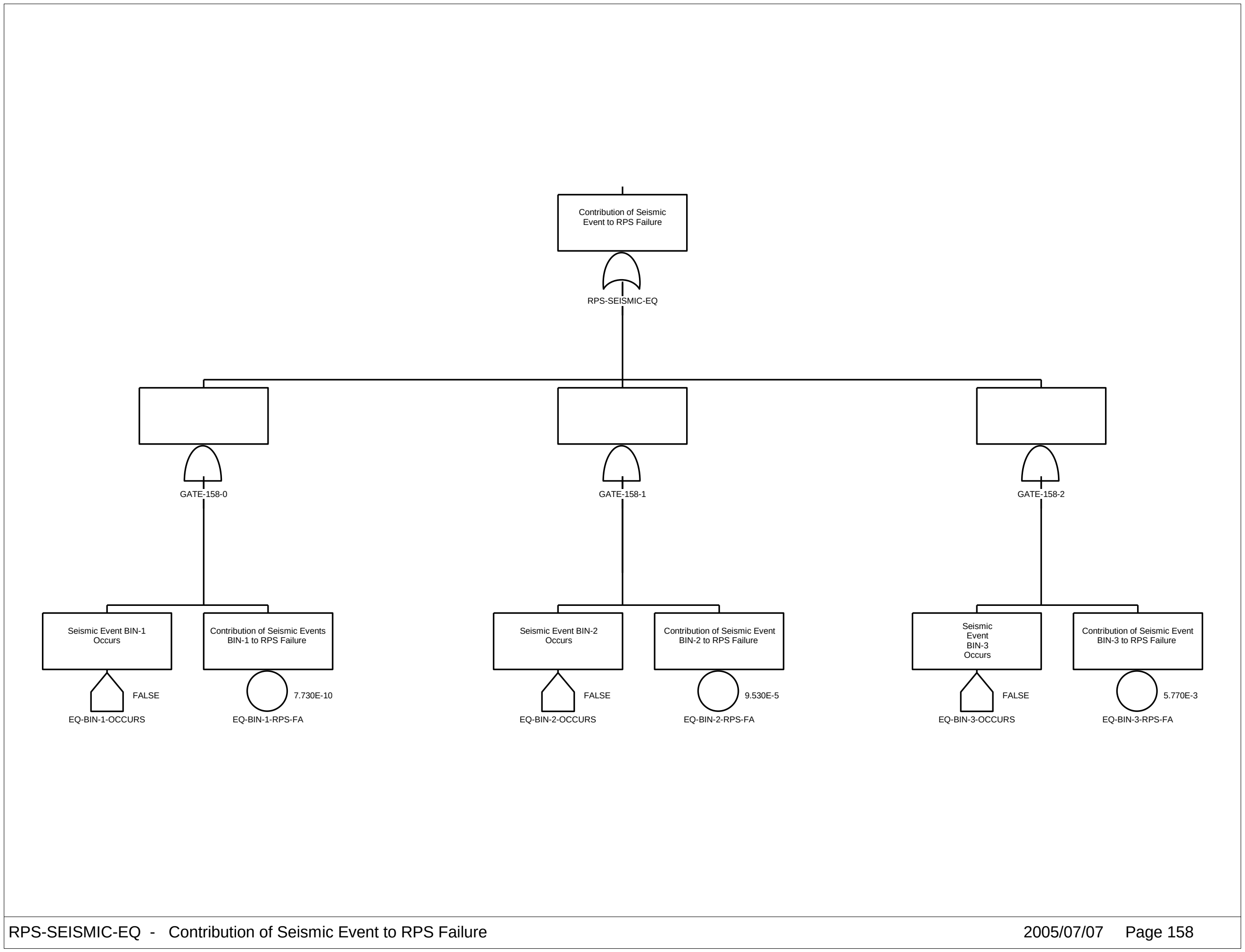


Figure 4-6. RPS SEISMIC EQ Fault Tree

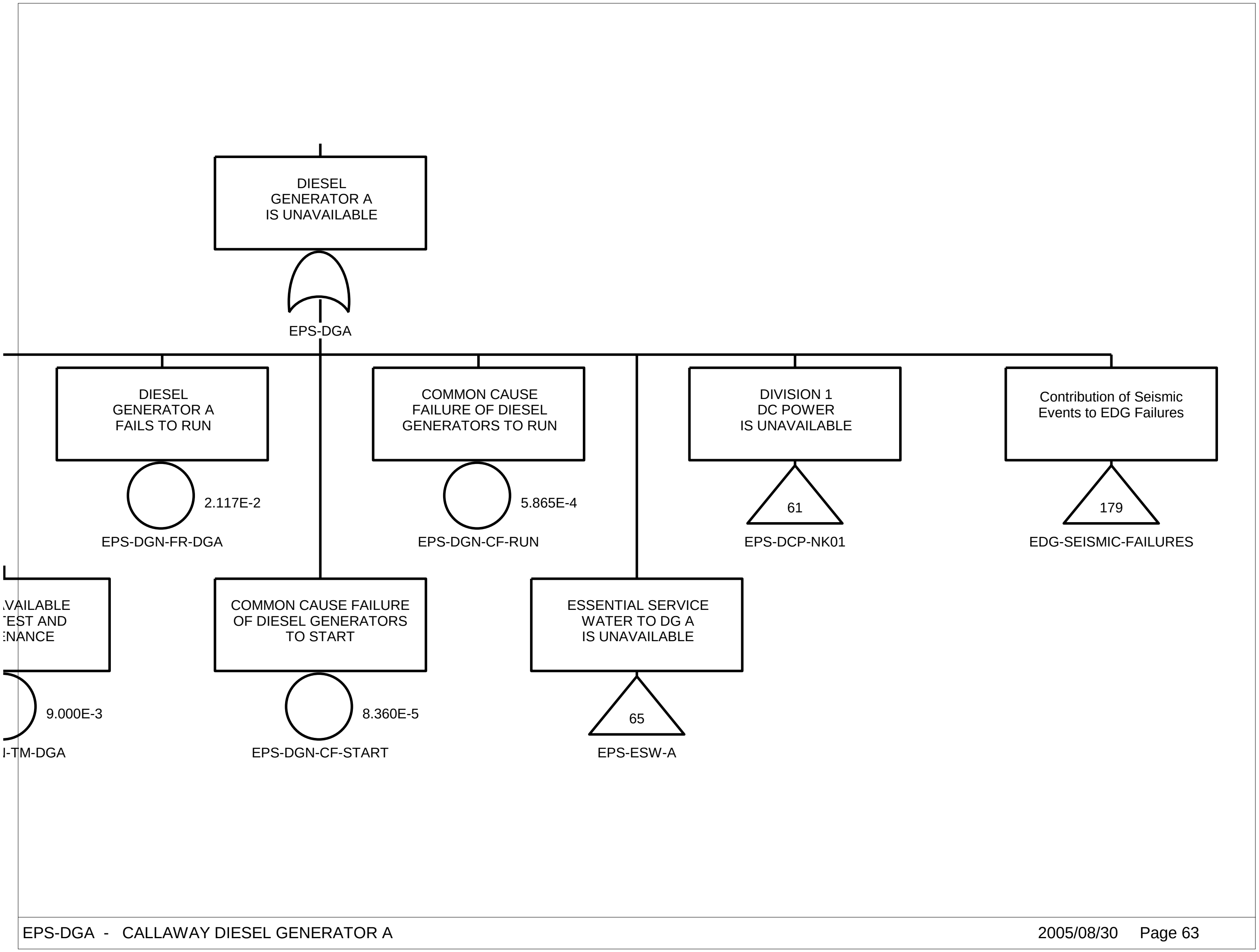


Figure 4-7. Adding Seismic Failures to a Support System (Figure 1 of 3)

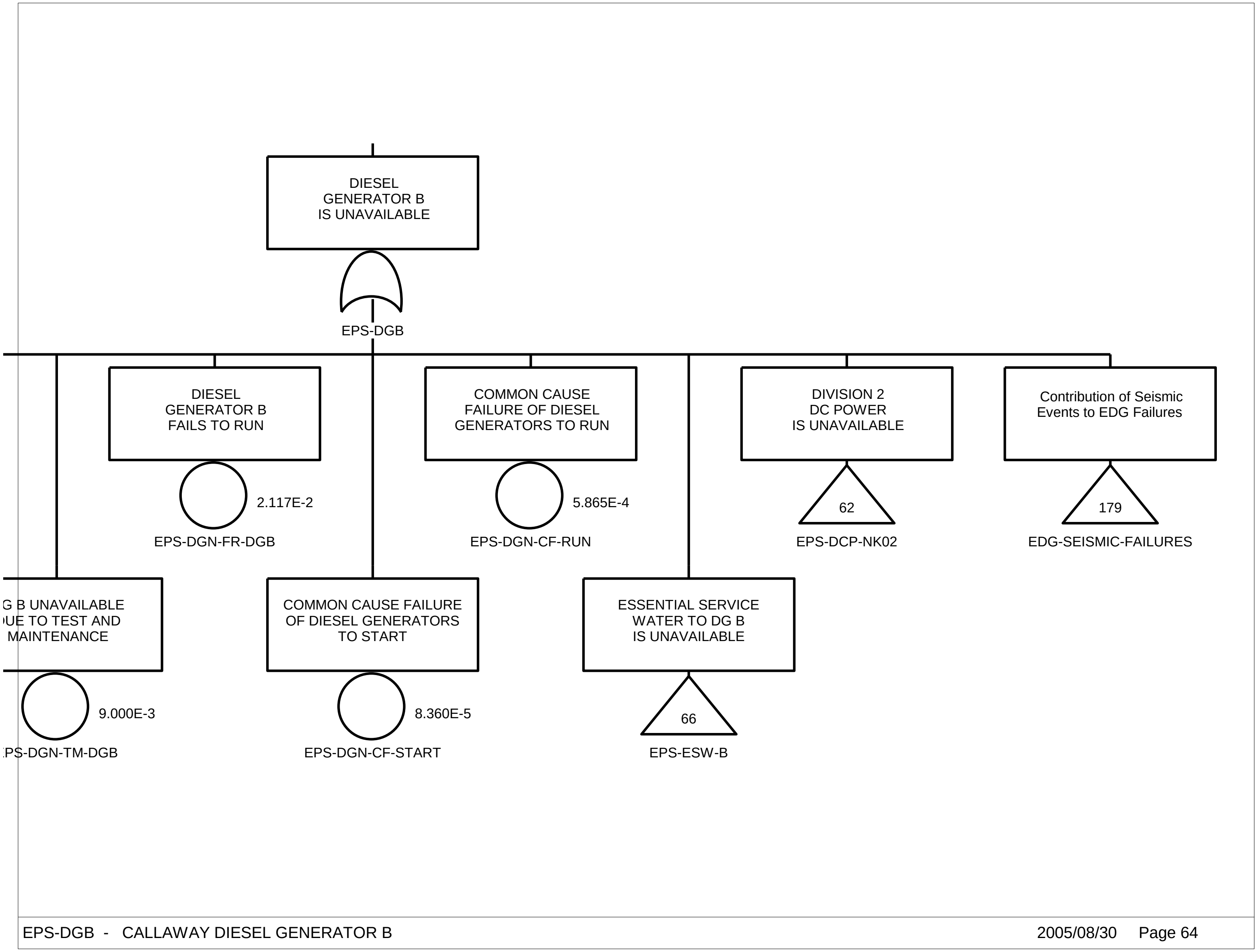


Figure 4-8. Adding Seismic Failures to a Support System (Figure 2 of 3)

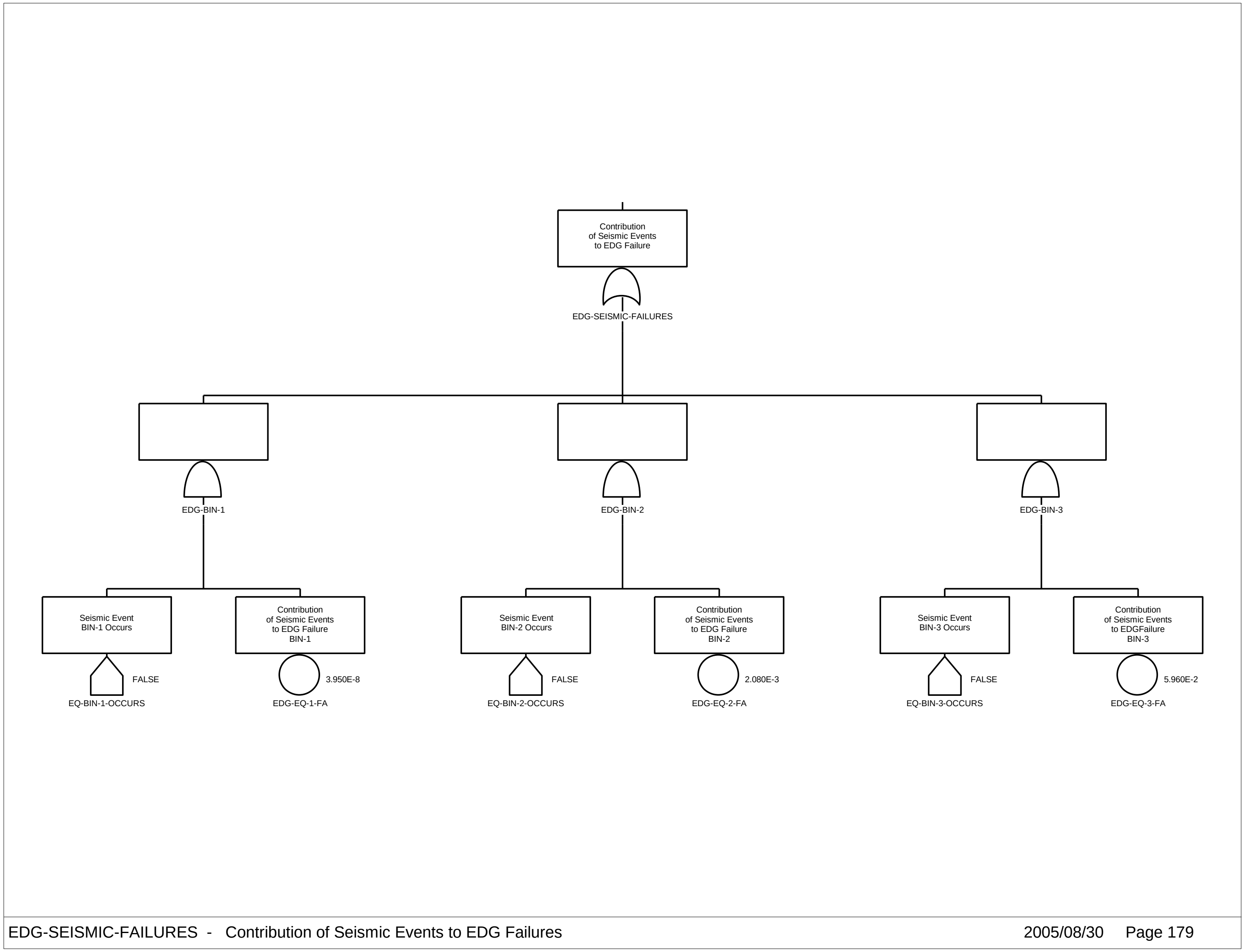


Figure 4-9. Adding Seismic Failures to a Support System (Figure 3 of 3)

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	1.00E+00	FT name

4.2.8 Application to SMA Plants

The model described from Section 4.2.5 to Section 4.2.7 can be adapted to develop limited SPRA for plants that have an SMA. For an SMA plant, the following process applies:

1. Obtain the seismic hazard vector from Appendix 4A. Calculate BIN frequencies and assign bin acceleration levels.
2. Examine the SMA documentation to locate any SSC fragilities and/or HCLPFs. It should be noted that most SMA from IPEEEs have very limited fragilities. Supplement that information with additional SSC fragilities from the collection of SSC seismic fragilities in ADAMS Accession No. ML071220070.

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

The nonsafety systems credited in the at-power PRA have high likelihood of failure in BINs 2 and 3. Therefore as a precaution, these non-safety systems should not be credited in BINs 2 and 3. Examples of 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 Appendix 4A. Such a calculation is done for all 61 U.S. nuclear power plants and is given in Appendix 1.

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 station black out (SBO) analysis, such credit for any of those power sources may need to be reconsidered for seismically-induced LOOP because those power sources may not be seismically qualified.

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 human error probabilities (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 a detailed human error analysis for operator actions credited during a seismic event, a model for adjustment of human error probabilities in a SPRA is given in an NRC document that is used during the construction of SPAR-AHZ models. This document, which contains proprietary information, is available (non-public) in ADAMS as ML13280A056 and it may be used as needed.

Furthermore, sensitivity analyses may be performed to understand the effect of dominant HEPs and the adjustments to the HEPs due to a seismic event.

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 malfunctioning relays have an adequate seismic capacity.

Relay chatter may introduce system actuation failures or spurious actuations. Operator actions may be needed for starting otherwise auto-start safety systems. This handbook does not provide guidance to address modeling of relay chatter problems explicitly. However, it should be noted that generic relay seismic fragilities are typically lower than that of other SSCs. See NUREG/CR-4840, page 3-32 for a discussion.

Unless the IPEEE or similar reports identified relay chatter vulnerabilities, this issue needs not be pursued for evaluation purposes.

In 2014, as part of the efforts to address lessons learned from the Fukushima events, industry conducted high-frequency seismic testing of typical plant control components. The following component categories are tested with averaged spectral accelerations over the 20 to 40 Hz range:

– Control and protective relays,
– Contactors and motor starters,
– Molded case circuit breakers,
– Control switches,
– Process switches and transmitters,
– Low- and medium-voltage circuit breakers, and
– Potentiometers and proximity switches.

The results of this test program are documented in the publicly-available report EPRI 3002002997, “High Frequency Program: High Frequency Testing Summary.”

4.3.5 Seismically-Induced Internal Flooding and Fires

For seismically-induced internal flooding scenarios, 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 in seismic BINs 2 and 3. Even for safety-related systems, seismically-induced internal flooding issues may arise if a plant vulnerability or a plant condition is observed.

For 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.

After the Fukushima events, consideration of concurrent events (or induced events) became a subject of renewed interest. Recommendation 3 of the NTTF's report, which is classified as a Tier 3 activity, concluded that the staff should evaluate potential enhancements to the capability to prevent or mitigate seismically-induced fires and floods (SIFFs). A publicly available NRC report on investigating the feasibility for modeling and quantitatively evaluating seismically-induced fires and floods in a PRA is available in ADAMS (Accession No. ML16004A250). As part of the SECY-15-0137, the staff indicated that broad regulatory activities pertaining to seismic, fire, and flooding events, operating experience involving SIFFs, and actions taken in response to the Fukushima accident, the staff's conclusion is that additional requirements related to SIFF are not needed. In the SRM-SECY-15-0137 dated February 8, 2016, the Commission has approved staff's recommendation to close NTTF Recommendation 3.

Therefore, the issues of seismically-induced flooding and fires are mentioned but not further pursued in this handbook at this time.

4.3.6 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 4-10 of this handbook shows the calculations of the SLOCA and MLOCA probabilities for the PGA values for the three seismic bins discussed in Section 4.2.3. An MS EXCEL file containing these values is placed in ADAMS with Accession No. ML071220066 as well as in the RASP Tool Box Web site (internal use only). The EXCEL file can also be used to calculate the SLOCA and MLOCA probabilities for more than three seismic bins.

4.3.7 Seismic Correlation Coefficients

One of the important elements of SPRA, which is different from the internal events PRA, is the treatment of dependencies or correlations in the seismic capacities of SSCs and in their responses to earthquakes. Specifically, the major dependence arises from the earthquake itself since it subjects all the components in the plant to the effects of vibratory motion. The questions of interest include whether the failures of component are somehow correlated or dependent and how the analyst can quantitatively account for that correlation or dependency. This issue is important because whether these capacities and responses are independent or partially (or even totally) dependent, especially for identical or nearly identical SSCs that are co-located or nearly so, can make a difference to the insights derived from many seismic PRAs.

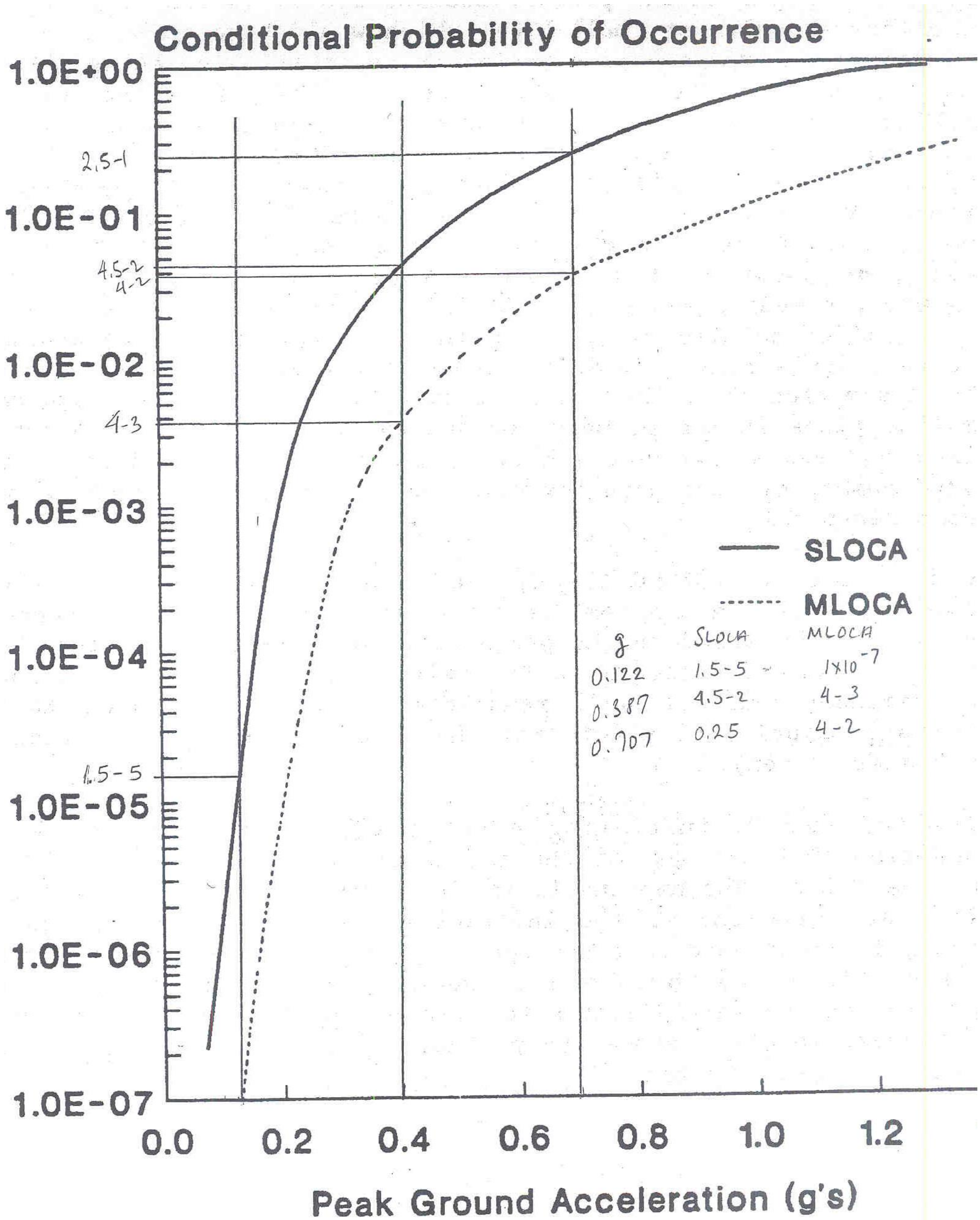


Figure 4-10. Estimation of Seismically induced SLOCA and MLOCA Probabilities (NUREG/CR 4840, Figure 3 6)

NUREG/CR-4840 provides simple rules for assigning the response correlation so that the tedious response correlation task could be avoided. These rules are given in Table 4.6. These rules include situations for which the recommended correlation is 0.5 or 0.75. For practical reasons, it is common practice to exclusively use correlation of one or zero, which simplifies the SPRA modeling.[3] For identical, redundant equipment, a correlation of one should be assumed. For all other equipment, a correlation of zero should be assumed.[4] If excessive conservatism in the results is observed due to this shortcut, other correlation coefficients may be introduced for a selected few SSC groups at the cost of model complication.[5] As an example, in the seismic modeling in Figure 4-7 and Figure 4-8, the seismic correlation is assumed to be one because the "EDG Seismic Failures" gate for both EDG A and EDG B transfer to the same fault tree in Figure 4-9.

Table 4-6. Rules for Assigning Response Correlation

Rule #	Description
1	Components on the same floor slab and sensitive to the same spectral frequency range (i.e., ZPA, 5-10 Hz. or 10-15 Hz) will be assigned response correlation = 1.0.
2	Components on the same floor slab sensitive to different ranges of spectral acceleration will be assigned response correlation = 0.5.
3	Components on different floor slabs (but in the same building) and sensitive to the same spectral frequency range (ZPA, 5–10 Hz or 10–15 Hz) will be assigned response correlation = 0.75.
4	Components on the ground surface (outside tanks, etc.) shall be treated as if they were on the grade floor of an adjacent building
5	"Ganged" valve configurations (either parallel or series) will have response correlation = 1.0.
6	All other configurations will have response correlation equal to zero.

4.3.8 Multi-Unit Effects

The effect of a seismic event on sites with multiple NPPs should be considered at least from the following aspects:

- ***Credit for cross-ties between two units.*** In many PRAs for an NPP, credit is taken for cross ties to a second unit on the same site. Examples of this credit are:
  - Electrical ties between units,
  - Ties of emergency feed-water supply (*i.e*. CST) between units,
  - Ties of refueling water (in RWST) between units, and
  - Ties of service water (or seawater) systems between units.

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3 NUREG/CR-4840 proposed to assign perfect (100 percent) correlation or dependency to the seismic response and capacity of identical SSCs if they are co-located or nearly so, and zero correlation or dependency otherwise. It was recognized early-on (a) that the 100-percent-correlation assignment is surely conservative for most situations in which it is applied, albeit perhaps not by much for many situations; and (b) that "zero correlation otherwise" is likely to be non-conservative in some situations. However, it was also generally thought that the differences are typically not likely to be important nor to compromise the major safety insights."

4 Also see Appendix D *Correlation between Seismic Failures* in publicly-available report EPRI 3002000709, "Seismic Probabilistic Risk Assessment Implementation Guide."

5 As of 2016, NRC/RES has a draft NUREG/CR report titled "Correlation of Seismic Performance in Similar SSCs (Structures, Systems, and Components)," ADAMS No. ML16035A002.

Such credit during seismic events should be either eliminated, or at least discounted. The reason of discounting the credit is that the SSCs of the second unit will either be dedicated to the second unit due to the likely trip of the unit, or at least will have additional (and maybe correlated) failures due to the nature of the event.

- ***Credit for an offsite emergency AC power source.*** In SBO scenarios, credit is often taken for offsite AC power sources and their transmission lines to the site. These SSCs are likely not seismically qualified to the same level as the conventional AC power sources that are already modeled. Credit for offsite AC power sources should not be taken in SBO scenarios if such sources and their transmission lines are deemed to be affected by the seismic event.

- ***Magnitude of fission product release.*** When a multi-unit site experiences a high-intensity seismic event, a multi-unit trip leading to core damage for multiple units is considerably more likely than that due to other random events (such as internal events at-power). This would increase the potential fission product release magnitude and frequency compared to a single-event core damage.

  Since LERF is not in the scope of this document, this subject is mentioned but not pursued further.

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 cut sets are identified using the SAPHIRE software. Table 4-7, Table 4-8, and Table 4-9 provide an illustration of the results and output for a plant-specific SPAR-AHZ seismic model.

Table 4-7. Seismic Event BIN Frequencies

Bin	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 4-8. Seismic Event Sequence Frequencies

Event Tree	Sequence	CDF	Cut Sets	End State	Notes
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

Event Tree	Sequence	CDF	Cut Sets	End State	Notes
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
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

Event Tree	Sequence	CDF	Cut Sets	End State	Notes
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**		

4.5 LERF Quantification for Seismic Events

LERF modeling and quantification is not currently addressed.

Table 4-9. Seismic Event CDF Cut Sets

#	% Cut Set	CDF	Basic Event	DESCRIPTION	Event Prob./ Freq.
1	82.97	3.80E-6	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5g)	7.212E-06
			CD-EQ3	DIRECT FUEL DAMAGE EVENTS	5.270E-01
2	11.73	5.37E-7	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5g)	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 (0.3–0.5g)	1.258E-05
			CD-EQ2	DIRECT FUEL DAMAGE EVENTS	3.550E-02
4	0.73	3.33E-8	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5g)	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 (0.3–0.5g)	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 (> 0.5g)	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 (> 0.5g)	7.212E-06
			AFW-TDP-TM-1C	AFW TDP 1C UNAVAILABLE DUE TO TEST AND MAINTENANCE	5.000E-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
8	0.14	6.00E-9	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5g)	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

#	% Cut Set	CDF	Basic Event	DESCRIPTION	Event Prob./ Freq.
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
9	0.12	5.37E-9	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5g)	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 (> 0.5g)	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 (> 0.5g)	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
12	0.05	2.02E-9	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3–0.5g)	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 (0.3–0.5g)	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 (> 0.5g)	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 (0.3–0.5g)	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

#	% Cut Set	CDF	Basic Event	DESCRIPTION	Event Prob./ Freq.
			LOOP-EQ2	LOSS OF OFFSITE POWER	5.720E-01
16	0.03	9.23E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05–0.3g)	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
			/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 (0.3–0.5g)	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 (> 0.5g)	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 (> 0.5g)	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
20	0.02	7.87E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05–0.3g)	2.842E-04
			CD-EQ1	DIRECT FUEL DAMAGE EVENTS	2.770E-06
21	0.02	7.87E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05–0.3g)	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 (0.05–0.3g)	2.842E-04

#	% Cut Set	CDF	Basic Event	DESCRIPTION	Event Prob./ Freq.
			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 (0.3–0.5g)	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 (0.3–0.5g)	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 (0.05–0.3g)	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
26	0.02	6.45E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3–0.5g)	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 (> 0.5g)	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 (> 0.5g)	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 (0.3–0.5g)	1.258E-05

#	% Cut Set	CDF	Basic Event	DESCRIPTION	Event Prob./ Freq.
			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 (0.05–0.3g)	2.842E-04
			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 (0.3–0.5g)	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 (> 0.5g)	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 (> 0.5g)	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-TRAINB-ALIGNED	SW TRAIN B ALIGNED TO TURBINE BLDG	5.000E-01
34	0.01	4.18E-10	IE-EQK-BIN-3	SEISMIC INITIATOR (> 0.5g)	7.212E-06
			/CD-EQ3	DIRECT FUEL DAMAGE EVENTS	4.730E-01

#	% Cut Set	CDF	Basic Event	DESCRIPTION	Event Prob./ Freq.
			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 (> 0.5g)	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-TRAINB-ALIGNED	SW TRAIN B ALIGNED TO TURBINE BLDG	5.000E-01
36	0.01	3.53E-10	IE-EQK-BIN-1	SEISMIC INITIATOR (0.05–0.3g)	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 (0.3–0.5g)	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 (0.3–0.5g)	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 (0.3–0.5g)	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

#	% Cut Set	CDF	Basic Event	DESCRIPTION	Event Prob./ Freq.
			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 (> 0.5g)	7.212E-06
			/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 (0.05–0.3g)	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 (0.05–0.3g)	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 (0.3–0.5g)	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
			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 (0.3–0.5g)	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

#	% Cut Set	CDF	Basic Event	DESCRIPTION	Event Prob./ Freq.
			/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 (> 0.5g)	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 (> 0.5g)	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
			/SLOCA-EQ3	SMALL LOCA EVENT	7.500E-01
47	0.01	2.65E-10	IE-EQK-BIN-2	SEISMIC INITIATOR (0.3–0.5g)	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 (> 0.5g)	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 (> 0.5g)	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 (> 0.5g)	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

#	% Cut Set	CDF	Basic Event	DESCRIPTION	Event Prob./ Freq.
			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

Appendix 4A Generic Seismic Hazard Vectors

The generic hazard vectors for 58 sites east of the Rocky Mountains are obtained from licensees' submittals in 2014 as part of the effort to address NRC Near-Term Task Force (NTTF) Recommendation 2.1.

The hazard vectors for the remaining three sites (Columbia, Diablo Canyon, and Palo Verde) are obtained from licensees' submittals in 2015 as part of the effort to address NRC NTTF Recommendation 2.1.

The submittals are available at the following NRC SharePoint Site.

Table 4-10 provides the seismic hazard vectors for the 61 U.S. nuclear power plants. Uncertainty information for each of the seismic hazard vector can also be obtained from the licensees' submittals.

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

Table 4-10. 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	12/13	14	15/16	17/18
	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
0.03	2.60E-03	1.83E-03	1.52E-03	2.95E-03	1.56E-03	1.35E-03	8.25E-03	8.08E-04	2.49E-03
0.05	1.22E-03	7.68E-04	6.31E-04	1.46E-03	8.00E-04	6.41E-04	4.54E-03	2.81E-04	1.22E-03
0.075	6.12E-04	3.81E-04	3.07E-04	7.68E-04	4.17E-04	3.49E-04	2.69E-03	1.18E-04	6.51E-04
0.1	3.57E-04	2.19E-04	1.83E-04	4.62E-04	2.45E-04	2.26E-04	1.78E-03	6.39E-05	4.06E-04
0.15	1.55E-04	1.35E-04	8.70E-05	2.09E-04	1.05E-04	1.20E-04	9.17E-04	2.64E-05	2.01E-04
0.3	3.21E-05	1.99E-05	2.26E-05	4.37E-05	1.99E-05	3.68E-05	2.28E-04	5.39E-06	5.54E-05
0.5	9.43E-06	5.08E-06	7.39E-06	1.26E-05	5.13E-06	1.37E-05	6.54E-05	1.51E-06	2.00E-05
0.75	3.38E-06	1.40E-06	2.76E-06	4.63E-06	1.67E-06	5.66E-06	2.17E-05	5.04E-07	8.41E-06
1.00	1.55E-06	4.59E-07	1.29E-06	2.21E-06	7.32E-07	2.85E-06	9.66E-06	2.18E-07	4.36E-06

	19	20	21/22	23/24	25	26	27/28	29/30	31
	Clinton	Columbia	Comanche Peak	Cook	Cooper	Davis-Besse	Diablo Canyon	Dresden	Duane Arnold
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
0.03	5.15E-03	[redacted]	2.24E-04	2.10E-03	7.73E-04	9.11E-04	[redacted]	1.60E-03	3.04E-04
0.05	2.20E-03	[redacted]	7.27E-05	9.06E-04	2.94E-04	4.14E-04	[redacted]	7.21E-04	1.12E-04
0.075	9.94E-04	[redacted]	2.87E-05	4.46E-04	1.38E-04	2.11E-04	[redacted]	3.75E-04	5.23E-05
0.1	5.40E-04	[redacted]	1.48E-05	2.67E-04	8.15E-05	1.25E-04	[redacted]	2.32E-04	3.13E-05
0.15	2.16E-04	[redacted]	5.86E-06	1.29E-04	3.83E-05	7.94E-05	[redacted]	1.16E-04	1.54E-05
0.3	4.09E-05	[redacted]	1.28E-06	3.37E-05	9.16E-06	1.43E-05	[redacted]	3.19E-05	4.29E-06
0.5	1.12E-05	[redacted]	4.06E-07	1.08E-05	2.67E-06	4.31E-06	[redacted]	1.08E-05	1.51E-06
0.75	3.71E-06	[redacted]	1.52E-07	3.85E-06	8.85E-07	1.50E-6	[redacted]	4.17E-06	5.98E-07
1.00	1.61E-06	[redacted]	7.15E-08	1.70E-06	3.79E-07	6.42E-07	[redacted]	2.00E-06	2.92E-07

	32/33	34	35	36	37	38	39/40	41	42/43
	Farley	Fermi	Fitzpatrick	Fort Calhoun	Ginna	Grand Gulf	Hatch	Hope Creek	Indian Point
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
0.03	1.93E-04	9.04E-04	7.84E-04	1.61E-03	5.19E-04	7.53E-04	1.45E-03	9.93E-04	1.20E-03
0.05	7.25E-05	3.91E-04	2.91E-04	6.95E-04	2.07E-04	2.31E-04	6.87E-04	4.46E-04	7.04E-04
0.075	3.57E-05	2.00E-04	1.28E-04	3.35E-04	9.91E-05	8.28E-05	2.87E-04	2.30E-04	4.52E-04
0.1	2.24E-05	1.24E-04	7.04E-05	1.97E-04	5.87E-05	4.02E-05	1.47E-04	1.40E-04	3.25E-04
0.15	1.19E-05	6.18E-05	2.99E-05	9.23E-05	2.78E-05	1.55E-05	5.69E-05	6.67E-05	1.97E-04
0.3	3.84E-06	1.70E-05	6.65E-06	2.24E-05	7.24E-06	3.29E-06	7.25E-06	1.51E-05	7.45E-05
0.5	1.52E-06	5.73E-06	2.07E-06	6.61E-06	2.45E-06	9.35E-07	1.11E-06	4.04E-06	3.17E-05
0.75	6.60E-07	2.18E-06	7.77E-07	2.14E-06	9.52E-07	3.07E-07	2.69E-07	1.23E-06	1.45E-05
1.00	3.44E-07	1.03E-06	3.70E-07	8.72E-07	4.61E-07	1.30E-07	8.79E-08	4.89E-07	7.79E-06

	44/45	46/47	48/49	50/51	52	53/54	55/56	57/58/59	60
	LaSalle	Limerick	McGuire	Millstone	Monticello	Nine Mile Point	North Anna	Oconee	Oyster 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
0.03	4.52E-03	9.50E-04	2.07E-03	8.60E-04	5.28E-04	8.04E-04	**	2.71E-03	2.34E-03
0.05	1.98E-03	4.47E-04	9.66E-04	5.615E-04	2.44E-04	3.00E-04	1.07E-03	1.31E-03	1.01E-03
0.075	9.50E-04	2.37E-04	5.06E-04	2.08E-04	1.30E-04	1.32E-04	9.55E-04	7.14E-04	4.43E-04
0.1	5.57E-04	1.48E-04	3.14E-04	1.24E-04	8.28E-05	7.31E-05	6.51E-04	4.60E-04	2.32E-04
0.15	2.61E-04	7.34E-05	1.56E-04	8.1E-05	4.27E-05	3.11E-05	3.71E-04	2.43E-04	8.88E-05
0.3	6.66E-05	1.95E-05	4.50E-05	1.84E-05	1.23E-05	6.92E-06	1.37E-04	7.59E-05	1.50E-05
0.5	2.01E-05	6.44E-06	1.70E-05	5.97E-06	4.33E-06	2.16E-06	5.70E-05	2.95E-05	3.54E-06
0.75	6.42E-06	2.44E-06	7.41E-06	2.43E-06	1.70E-06	8.08E-07	2.54E-05	1.28E-05	1.06E-06
1.00	2.59E-06	1.15E-06	3.92E-06	1.08E-06	8.18E-07	3.85E-07	1.39E-05	6.71E-06	4.39E-07

** Information not provided in the licensee's submittal

	61	62/63/64	65/66	67	68	69/70	71/72	73/74	75
	Palisades	**Palo Verde**	**Peach Bottom**	**Perry**	**Pilgrim**	**Point Beach**	**Prairie Island**	**Quad Cities**	**River Bend**
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**
0.03	2.87E-03	[redacted]	1.15E-03	7.68E-04	2.47E-03	9.36E-04	1.43E-04	7.92E-04	5.20E-04
0.05	1.29E-03	[redacted]	6.70E-04	3.86E-04	1.39E-03	3.69E-04	5.48E-05	3.21E-04	1.80E-04
0.075	6.34E-04	[redacted]	4.27E-04	2.17E-04	8.57E-04	1.67E-04	2.64E-05	1.57E-04	8.15E-05
0.1	3.77E-04	[redacted]	3.04E-04	1.38E-04	5.99E-04	9.41E-05	1.58E-05	9.50E-05	4.80E-05
0.15	1.78E-04	[redacted]	1.83E-04	9.05E-05	3.50E-04	4.14E-05	7.62E-06	4.71E-05	2.26E-05
0.3	4.61E-05	[redacted]	6.82E-05	1.90E-05	1.24E-04	9.28E-06	1.99E-06	1.33E-05	5.30E-06
0.5	1.48E-05	[redacted]	2.93E-05	5.74E-06	5.08E-05	2.66E-06	6.49E-07	4.65E-06	1.50E-06
0.75	5.18E-06	[redacted]	1.38E-05	1.95E-06	2.23E-05	8.43E-07	2.39E-07	1.84E-06	4.86E-07
1.00	2.21E-06	[redacted]	7.63E-06	8.34E-07	1.15E-05	3.36E-07	1.10E-07	8.96E-07	2.05E-07

	76	77/78	79/80	81	82/83	84	85/86	87/88	89/90
	Robinson	**Saint Lucie**	**Salem**	**Seabrook**	**Sequoyah**	**Shearon Harris**	**South Texas**	**Surry**	**Susquehanna**
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**
0.03	6.20E-03	1.38E-04	8.46E-04	1.90E-03	6.64E-04	9.03E-04	8.67E-05	1.37E-03	5.08E-04
0.05	3.72E-03	5.21E-05	3.67E-04	1.11E-03	2.39E-04	3.31E-04	4.16E-05	4.36E-04	2.17E-04
0.075	2.38E-03	2.40E-05	1.84E-04	7.14E-04	9.04E-05	1.37E-04	2.31E-05	1.57E-04	1.09E-04
0.1	1.66E-03	1.40E-05	1.11E-04	5.13E-04	4.18E-05	6.99E-05	1.52E-05	7.50E-05	6.59E-05
0.15	9.09E-04	6.36E-06	5.13E-05	3.11E-04	1.30E-05	2.60E-05	8.28E-06	2.66E-05	3.20E-05
0.3	2.33E-04	1.43E-06	1.15E-05	1.15E-04	1.68E-06	4.70E-06	2.71E-06	4.44E-06	8.57E-06
0.5	6.44E-05	3.82E-07	3.24E-06	4.78E-05	4.15E-07	1.37E-06	1.07E-06	1.13E-06	2.94E-06
0.75	1.94E-05	1.12E-07	1.07E-06	2.14E-05	1.45E-07	5.01E-07	4.69E-07	3.71E-07	1.15E-06
1.00	7.51E-06	4.32E-08	4.58E-07	1.14E-05	6.86E-08	2.36E-07	2.45E-07	1.64E-07	5.62E-07

For the three sites West of Rocky Mountains, this information is obtained from licensees' submittals as part of the effort to address Near-Term Task Force (NTTF) Recommendation 2.1 in 2015. (See ML15078A243 for Columbia, ML15070A607 and ML15070A608 for Diablo Canyon, and ML15076A073 for Palo Verde).

20		27/28		62/63/64	
Columbia		**Diablo Canyon**		**Palo Verde**	
g value	**mean f per year**	**g value**	**mean f per year**	**g value**	**mean f per year**
0.03	7.03E-03	0.05	2.30E-02	0.03	1.07E-03
0.05	3.94E-03	0.1	8.40E-03	0.05	3.97E-04
0.075	2.41E-03	0.25	2.00E-03	0.075	1.84E-04
0.10	1.67E-03	0.5	4.30E-04	0.1	1.07E-04
0.20	6.46E-04	0.7	1.70E-04	0.15	5.00E-05
0.30	3.53E-04	1	4.90E-05	0.3	1.23E-05
0.50	1.54E-04	1.2	2.60E-05	0.5	3.70E-06
0.75	7.45E-05	1.6	8.20E-06	0.75	1.24E-06
1.0	4.22E-05	2.0	3.20E-06	1.0	5.30E-07
		3.0	5.00E-07	1.5	1.43E-07
		4.0	1.30E-07		

Appendix 4B Seismic Fragility/PGA/HCLPF

The complete fragility description of any particular structure, system, component (SSC) includes a representation of both the probabilities of failure vs. peak ground acceleration (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 4-11, which is Figure 2-1 of NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," 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 percent probability of failure.

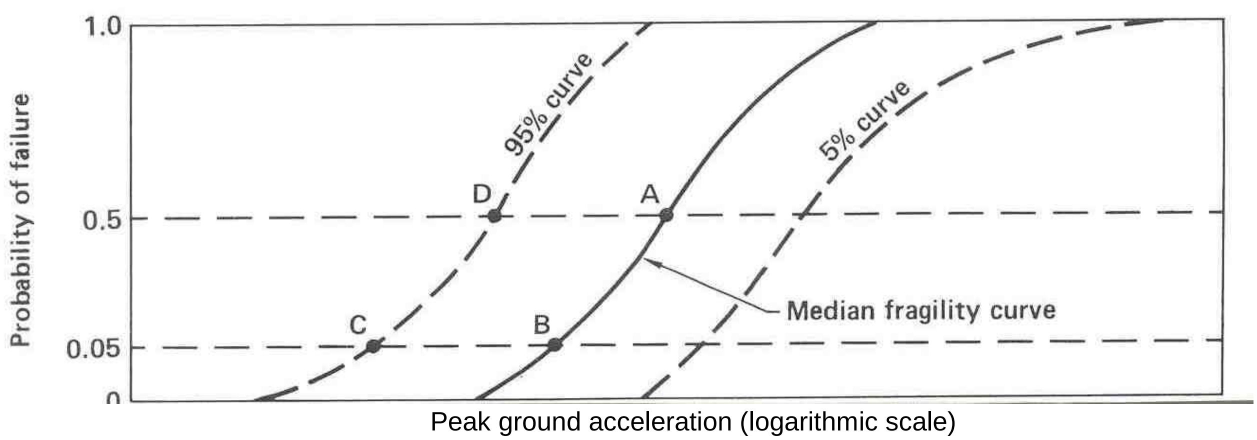

Figure 4-11. Fragility Curves

The dashed lines in Figure 4-11 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 percent (lower) confidence estimate of the median capacity. Specifically, the analyst is 95 percent confident that the median capacity exceeds this PGA level. Similarly, Point C represents the high (95 percent) confidence estimate of the PGA at which there is only a small (5 percent) 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 current 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 percent and 5 percent curves in Figure 4-11. 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 $e^{(-1.65\,\beta r)}$. Also, Point D is below the median by a factor of $e^{(-1.65\,\beta r)}$, and Point C is below the median by $e^{[-1.65(\beta r + \beta u)]}$.

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 ruggedness. 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 ($e^{[-1.65(\beta r + \beta u)]}$). 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 ($e^{-2.33\,\beta c}$). In deterministic SMAs, the HCLPF capacity is calculated using the conservative deterministic failure methodology method.

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”

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 PGA is the largest acceleration recorded by a particular station during an earthquake.

Appendix 4C Correspondence between PGA and Severity of Earthquakes

There are two methods of measurement for describing the effects of 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.

In seismology, the scale of seismic intensity is a way of measuring or rating the *effects* of an earthquake at different locations. The Modified Mercalli 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 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 relatively subjective scale of descriptions. As one can see from the list in Table 4-11, 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.

Table 4-11 also provides some information for the use of the Richter Scale, PGA, and Modified Mercalli scales for seismic events. The relation between Modified Mercalli 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 4-11. Modified Mercalli Intensity Scale vs. PGA

Mercalli Intensity	Equivalent Richter Magnitude	Witness Observations	Intensity Peak Acceleration (% 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

Mercalli Intensity	Equivalent Richter Magnitude	Witness Observations	Intensity Peak Acceleration (% g)
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 4-12 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.

Table 4-12. 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+

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 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 Volume 1 of this handbook.

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 ANSI/ANS-58.21-2003, "External Events PRA Methodology," excluding internal fires, internal flooding, and seismic events. For those events, complementary handbooks are already prepared.

Extreme winds/tornadoes are an example of 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 focuses on these hazards.

The handbook assumes that:

– The user has hands-on experience with SAPHIRE, and

– 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 that 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 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 a tornado, other than to pre-harden the facility). In fact the initiating event frequency for (internal events) LOOPs 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:

$$\text{CDF} = \text{Scenario Frequency} \times \text{CCDP}$$

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 (HWD, TOR, etc.) and is used both for the event tree and the initiating event names.
2. *Scenario description.*
3. *Scenario initiating event frequency (IE_{freq}).*

Table 5-1. Example Matrix Defining Other External Event Scenarios

	Name	Description	IE_{freq}	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 [note 1]	Offsite AC power	IE-LOOP	No reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA); event- specific LOOP recovery probabilities	None

Note:
[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:

$$CDF = IE_{freq} \times CCDP$$

The scenario frequency IE_{freq} 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 Event and Condition Assessment (ECA) Workspace 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 Volume 1 of NUREG/CR 6890, “Reevaluation of Station Blackout Risk at Nuclear Power Plants.”

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
6	0.010	0.31
7	0.08	0.28
8	0.07	0.26

Composite = Composite of plant-, switchyard-centered, and grid-, 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, from Volume 1 of NUREG/CR 6890.

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 Section 5.3.2 for an example on the treatment of a 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/station blackout (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:

- Reactor protection system failure is removed (no anticipated transient without scram);
- RCP seal LOCA (for pressurized-water reactors) is most likely not applicable and should be removed;
- Power-operated relief valve LOCA likelihood is considerably reduced; may be removed;
- Availability of auxiliary feedwater (AFW) and main feedwater (MFW) recovery should be established and kept in the model;
- Event-specific offsite power recovery distribution may need to be calculated and used; as a minimum, generic severe-weather recovery distribution should be used.
- 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.

- Introduction of an operator action to start a mitigating system (modeled in the LOOP/SBO event trees), which otherwise, would have started automatically.
- 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 Example

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

5.3.1 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 a Category 3 hurricane. 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 residual heat removal (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, and
   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 steam generators (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 cooling by AFW, if RHR removal cooling fails.
5. Unit 1 SPAR model is used to estimate the event importance.
6. The reactor coolant system temperature and pressure conditions are such that no RCP seal LOCA challenge exists.

For this Category 3 hurricane event, event-specific offsite power nonrecovery 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 hour = 1.0
In 2 hours = 0.5
In 3 hours = 5×10^{-2}
In ≥ 4 hours = 5×10^{-3}

When this AC power recovery distribution is used, the CCDP is calculated as 1.8×10^{-5}, which is the event importance.
Compare this with SPAR severe weather AC power recovery failure distribution. Namely,

Operator Fails to Recover Offsite Power:
In 1 hour = 0.46
In 2 hours = 0.36
In 3 hours = 0.30
In 4 hours = 0.25
In 5 hours = 0.22
In 6 hours = 0.20
In 7 hours = 0.18

With this recovery distribution, the event importance is calculated as CCDP = 3.4×10^{-5}.

6.0 External Flood Modeling and Risk Quantification

6.1 Objectives and Scope

The objective of this guidance is to improve and ensure continued consistency in external flooding risk assessments. This objective is accomplished by providing references to methods and datasets along with discussions on common issues related to key aspects of external flooding assessment. This guidance also discusses experience from recent Significance Determination Process (SDP) analyses that could be useful to risk analysts. This guidance does not provide a step-by-step guide that covers all aspects of external flooding assessments. At the time of development, a step-by-step guide was impractical because of the diverse nature of external flooding events and the lack of widely accepted methods for aspects of modeling and quantifying the risk of these events. Nevertheless, consistency in external flooding assessments can be maintained and enhanced by identifying potential issues and discussing their treatment in previous analyses. The scope of this guidance includes evaluations of those events that could not be dispositioned with simple screening methods and, therefore, require detailed assessments.

This guidance discusses sources of information related to flooding analyses (Section 6.2), considerations in using methods and datasets used for external flood hazard assessments (Section 6.3), and considerations in evaluation of flood protection features and human reliability (Section 6.4). Brief descriptions of several findings related to external flooding are provided in Appendix 6A and Appendix 6B. In addition, Appendix 6C provides a summary of point estimate failure rates for dams that are broken down by all sized dams.

Risk analysts can use this guidance as a source of information for insights and considerations related to some aspects of external flooding risk assessments. The guidance is meant to be reference material to inform risk assessments, it is not meant to be a procedure with a defined process and outcome.

6.2 Sources of Information

The objective of this section is to provide risk analysts with a list of possible sources of information for their use in performing an external flooding risk assessment. These sources are not listed in order of importance and do not include all possible sources of information. The risk analyst should determine which sources are needed, or would be beneficial, to support their specific assessment.

Section 2.4 of final safety analysis reports (FSARs) describes, among other subjects, design-basis floods for nuclear power plants. These reports provide valuable information regarding a flood caused by one or an appropriate combination of several hydrometeorological, geoseismic, or structural-failure phenomena, which results in the most severe hazards such as flooding due to precipitation, storm surge, 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 seiches are seismic and shoreline geography phenomena. The NRC has developed a database that contains information provided in FSARs relevant to external flooding such as flood mechanisms, heights and durations along with sources of data utilized for design-basis floods evaluations by the licensees. In addition to FSARs, plant procedures also describe the flood mitigation actions, which at times should be evaluated and quantified as a part of the overall risk assessment. Licensees also performed evaluations of external flood hazards, with varying degrees, as part of the individual plant examinations of external events (IPEEEs). External flood hazards are mostly evaluated deterministically in IPEEEs. Furthermore, IPEEEs provide no, or limited, information on external flood plant impact assessments and equipment fragilities. If a licensee has a peer-reviewed flooding probabilistic risk assessment (PRA), this could also be utilized as a source of information for an assessment.

NUREG/CR-7046, "Design-Basis Flood Estimation for Site Characterization at Nuclear Power Plants in the United States of America," describes general approaches, including some probabilistic aspects, for evaluating flood hazards consistent with present-day guidance and methods applicable to new reactors. Appendix A to NUREG/CR-7046 identifies and discusses the data and data sources that should be considered for collection depending on specific modeling tasks and levels of detail required. This Appendix places special emphasis on using nationally available datasets. Moreover, NUREG/CR-7046 contains appendices that describe currently available hydrometeorological datasets and geographical information system techniques that are useful in data preprocessing and synthesis of model inputs along with flood estimation techniques for various flood-causing mechanisms, such as local intense precipitation, dam breaches and failures, storm surges and seiche. In addition to NUREG/CR-7046, Electric Power Research Institute (EPRI) Report 3002005292, "External Flooding Hazard Analysis: State of Knowledge Assessment," examines probabilistic methods currently available to assess the external flooding risks and their uncertainties for local intense precipitation, riverine flooding, dam failure, and storm surge.

JLD-ISG-2012-06, "Interim Staff Guidance for Performing a Tsunami, Surge, or Seiche Hazard Assessment," describes methods acceptable to NRC staff for performing a tsunami, surge, or seiche hazard assessment in response to the 50.54(f) letter. While this interim staff guidance (ISG) references some components that are probabilistically informed (e.g., selection of hurricane storm parameters using the Joint Probability Method) it does not describe a framework for a full probabilistic characterization of coastal hazards.

JLD-ISG-2013-01, "Guidance for Assessment of Flooding Hazards Due to Dam Failure," describes methods acceptable to NRC staff for re-evaluating flooding hazards from dam failure in response to the 50.54(f) letter. This ISG does not describe a framework for probabilistic characterization of hazards from dam failure (with the exception of using a probabilistic seismic hazard assessment for defining seismic loads on dams). The ISG stated that "[probabilistic] seismic hazard analysis is accepted current practice in both the nuclear and dam safety communities […]. Probabilistic approaches for estimating the extreme rainfall and flood events of interest in [the Dam Failure ISG] (e.g., 1×10^{-4} per year or lower annual exceedance probability) exist, but there are no industry consensus standards or Federal guidance that defines current accepted practice. NRC has established probabilistic screening criteria for man-related hazards (e.g., between 1×10^{-7} and 1×10^{-6} annual exceedance probability) that are, in theory, applicable to sunny-day dam failures. However, no widely accepted methodology exists for estimating sunny-day dam failure probabilities on the order of 1×10^{-7} [to] 1×10^{-6} annual exceedance probability."

Information Notice (IN) 2012-02, "Potentially Nonconservative Screening Value for Dam Failure Frequency in Probabilistic Risk Assessments," discusses a potentially nonconservative screening value for dam failure frequency contained in Nuclear Safety Analysis Center-60, "A Probabilistic Risk Assessment of Oconee Unit 3." IN 2012-02 states that although historical dam failure information discussed in the information notice can "provide useful qualitative insights on the general performance and failure modes for certain dam types, its applicability to site-specific dams has to be assessed to establish sufficient technical bases. This is due to the variability in site-specific characteristics (i.e., hydrologic, geologic, and operational) and the potential contributions of site-specific failure modes not covered by databases." In addition, by referring to DSO-04-08, "Hydrologic Hazard Curve Estimating Procedures," IN 2012-02 stated that frequency extrapolations of severe weather phenomena with insufficient basis may not be fully justified depending on the quality and quantity of the supporting information beyond certain values.

Various state and federal agencies currently utilize flood frequency analysis and other related methods as a means to probabilistically characterize flooding hazards for their applications (e.g., DSO-04-08 and National Oceanic and Atmospheric Administration (NOAA) Precipitation Frequency Data Server (NOAA Atlas 14)). Information on probabilistic flood hazard assessment (PFHA) in other applications is documented in NUREG/CP-0302, "Proceedings of the Workshop on Probabilistic Flood Hazard Assessment (PFHA) held at U.S. NRC Headquarters, January 29 – 31, 2013." Finally, Appendix 6C provides a summary of dam failure rates that could be useful for a risk analyst in an assessment. The Dam failure information can be supplemented by other sources of information, such as the characteristics of dam failures available at the National Inventory of Dams and the National Performance of Dams Program.

6.3 Flood Hazard Assessment

The same two-step process used for other external hazards can be used to model external floods and quantify their CDFs. These two steps are described in Section 5.2.1 and Section 5.2.2.

To evaluate the risk of external flooding events, risk analysts often need to assess the likelihood that a specified parameter or set of parameters representing flood severity (e.g., flood elevation, flood event duration, and associated effects) are exceeded at a site during a specified exposure time. This can then be characterized into hazard exceedance information for input in risk assessment as frequencies or annual exceedance probabilities.

The current state of practice in flood hazard assessment used for siting of nuclear power plants is deterministic. Regulatory Guide 1.59, "Design-Basis Floods for Nuclear Power Plants," describes the design basis floods that nuclear power plants should be designed to withstand using the concept of a probable maximum event. As discussed later, data for developing probable maximum events, such probable maximum hurricane (PMH) and probable maximum precipitation (PMP), only goes back 100–200 years. In additions, these probable maximum events are single value parameters that are deterministic and only of limited use for PRA.

6.3.1 Current State-of-Practice

Although discrete components of a PFHA are available, a comprehensive PFHA methodology has not yet been developed. As discussed in Section 6.3.2, the risk analyst should note that because of limitations on using historical data no single approach or data source is sufficient for providing estimates of flood parameters over the full range of annual exceedance frequencies required for risk assessment applications (DSO-04-08). Therefore, results from a number of approaches may need to be considered to address modeling errors considerations and to appropriately consider epistemic uncertainties and to yield a family of hazard curves. Methods developed for generating flood hazard curves include, but are not limited to, at-site frequency analysis (using site-specific data and fitting a probability distribution to data with extrapolation to ranges beyond available data), regional frequency analysis (using regional data where site-specific information is rarely available using methods such as the average parameter approach, the index flood approach, and the specific frequency approach (Hosking, 1997)), and stochastic event-based modeling. Under a stochastic event-based modeling approach, hydrologic/hydraulic model inputs (e.g., meteorological and climatological parameters as well as model inputs such as antecedent and initial conditions) are treated as random variables. Typically, Monte Carlo sampling procedures are used to simulate events by allowing the input variables to vary in accordance with their respective probability distributions, including dependencies among relevant parameters. A large number of simulations are performed to support development of magnitude-frequency curves. Example A.5 and Example A.2 in Appendix 6A are instances where regional frequency analysis and stochastic event-based modeling were used, respectively. In reviewing licensees' analyses that utilize these models and methods, the limitations of these methods as well as their differences, to the extent practical, should be recognized. Furthermore, the employed data, models, and methods should be consistent with the range of hazards of relevance to the site.

6.3.2 Credible Extrapolation Ranges

Developing hazard curves for risk assessment uses the length of record and type of data to determine the extrapolation limits for flood frequency analysis. The limits of data and flood experience for any site or region place practical limits on the range of the floods to which annual exceedance frequencies can be assigned. Flood frequency analyses are typically performed for regional applications and address return periods of less than 1,000 years. For nuclear power plant safety risk assessments, flood estimates are needed for return periods of up to 1 million years (exceedance probability of 1 in a million). Developing credible estimates at these low probabilities generally could not be achieved, even by combining data from multiple sources and a regional approach.

Table 6-1 lists the different types of data that can be used as a basis for flood frequency estimates and the typical and optimal limits of credible extrapolation for annual exceedance probabilities (DSO-04-08). In general, the scientific limit to which the flood frequency relationship can be credibly extended, based upon any characteristics of the data and the record length, will fall short of the floods that need to be evaluated in risk-informed applications.

Table 6-1. Hydrometeorological Data Types and Extrapolation Limits for Flood Frequency Analysis

Type of Data Used for Flood Frequency Analysis	Limit of Credible Extrapolation for Annual Exceedance Probability	
	Typical	Optimal
At-site stream flow data	1 in 100	1 in 200
Regional stream flow data	1 in 500	1 in 1,000
At-site stream flow and at-site paleoflood data	1 in 4,000	1 in 10,000
Regional precipitation data	1 in 2,000	1 in 10,000
Regional stream flow and regional paleoflood data	1 in 15,000	1 in 40,000

Type of Data Used for Flood Frequency Analysis	Limit of Credible Extrapolation for Annual Exceedance Probability	
	Typical	**Optimal**
Combinations of regional data sets and extrapolation	1 in 40,000	1 in 100,000

Floods can be categorized, according to (Nathan, 1997), as large, rare, and extreme. These flood categories are shown in Figure 6-1 (DSO-04-08). Large floods generally encompass events for which direct observations and measurements are available. Rare floods represent events located in the range between direct observations and the credible limit of extrapolation from the data. Extreme floods generally have very small annual exceedance probabilities, which are beyond the credible limit of extrapolation but are still needed for risk assessments. Although external flooding events with annual exceedance probabilities in the large floods range have been assessed, extrapolation beyond the data is often performed by the licensees to provide information needed for risk assessments.

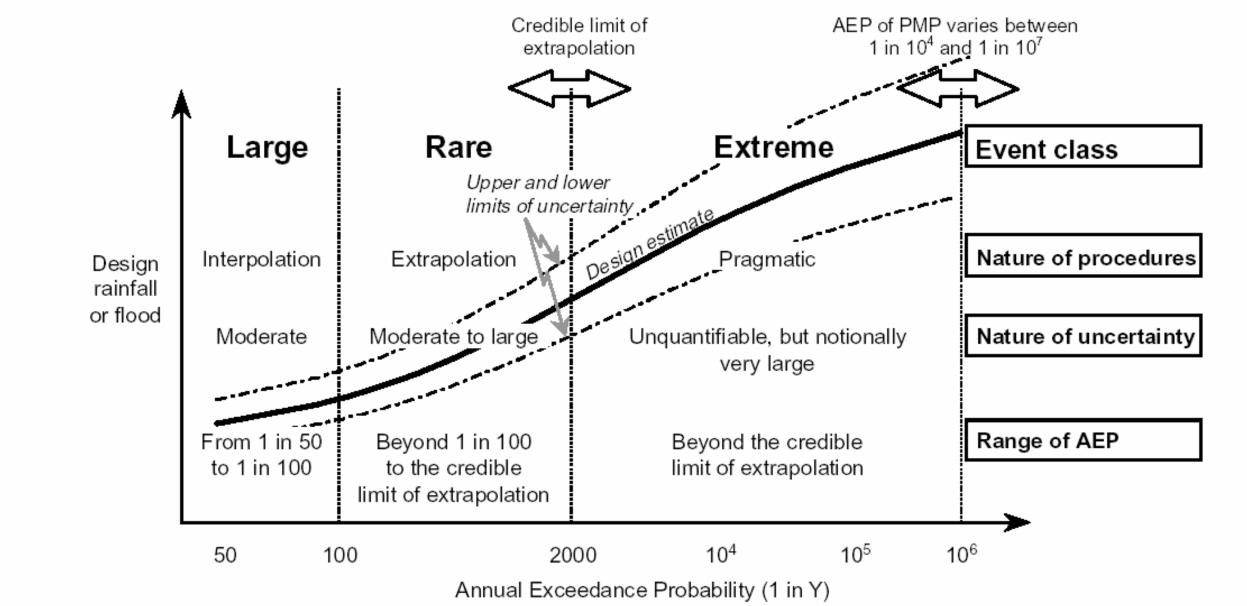

Figure 6-1. Characteristics of Notional Floods (Nathan, 1997)

Traditional sources of information used for estimating probabilities of floods (e.g., gauged stream flow records, indirect discharge measurements, tidal gauges, wind speed measures, and precipitation gauge records) have records that are less than 100 years in length with varying degrees of stationarity and homogeneity. Therefore, for large floods in Figure 6-1, the external flood hazard can be assessed with available data and records. Section 2 provides some references used to collect data. However, the limitations on length of records result in limits on the credible and technically defensible extrapolation of rare and extreme floods estimates based on conventional flood frequency analysis. U.S. Geological Survey (USGS) Bulletin 17B, "Guidelines for Determining Flood Flow Frequency," which is a consensus document among federal agencies, describes the data and procedures for computing flood flow frequency curves where systematic stream gaging records of sufficient length to warrant statistical analysis are available.[6] This report recommends use of the Pearson Type III distribution with log

[6] USGS Bulletin 17C, "Guidelines For Determining Flood Flow Frequency," which is the proposed update to USGS Bulletin 17B, was in the public review process when this guidance was under development.

transformation of data (log-Pearson Type III distribution) as a base method for flood flow frequency studies. This distribution is used to extrapolate the hazard curve in the rare floods range. Detailed process for using the log-Pearson III distribution is discussed in many documents, such as (Hamed, 1999). The limits of extrapolation for rare flooding events are determined by evaluating the lengths of records, number of stations in a hydrologically homogeneous region, degree of correlation between stations, and other data characteristics that may affect the accuracy of the data. The risk analyst may request assistance from the Office of Nuclear Reactor Regulation's Division of Risk Assessment (NRR/DRA) for subject matter expert to expedite this analysis by providing insights on limits of extrapolation for rare flooding events. In accordance with the Bulletin 17B Frequently Asked Questions, it is acknowledged that "the mathematical formula that should be used for the extrapolation [when using USGS Bulletin 17B] is not known with any confidence, and there is no agreed-upon procedure to assess or quantify the uncertainty in the extrapolation formula." Bulletin 17B Frequently Asked Questions also provides "rules" regarding extrapolation: (1) don't extrapolate unless necessary, (2) only extrapolate as far as necessary, (3) seek independent corroboration of extrapolated values and (4) "don't give too much credibility to or place too much reliance on the extrapolated values." While the appropriate limits on extrapolation for conventional flood frequency methods vary from site to site, they are generally limited to return periods ranging from 500–1000 years for typical sites and data sources. As a result, these methods alone are not appropriate for use in developing hazards curves for the entire range of return periods potentially required for external flood event assessments.

The uncertainty associated with extreme floods is very large. Oftentimes, these floods may result from unforeseen and unusual combinations of hydrologic parameters generally not represented in the flood history at a particular location (DSO-04-08). Because the extrapolation of hazard curves for the large return periods is not supported by flood frequency analyses, the risk analyst should consider the consequences of those extreme floods in making a risk-informed decision even though the hazard frequency cannot be practically characterized due to large uncertainties. Performing analyses using upper bound estimates may help determining whether further assessment of uncertainties is warranted. It is also important to note that probabilistic flooding analyses utilize the hazard curves, in combination with additional considerations and associated uncertainties, and a potential upper bound to the largest flood at a particular site or the probable maximum flood alone do not typically provide all the insights necessary for making risk informed decisions. Considering a spectrum of flooding events, including levels that are considerably lower than the PMF level, could also provide additional insights. For example, there may be a significant increase in risk when a flood elevation exceeds the switchyard elevations, which could be much lower than the PMF elevation.

6.3.3 Other Considerations

In analyzing recent event assessments for the SDP, some licensees performed detailed deterministic calculations for hazard assessment. The risk analyst may consult with experts in the fields relevant to flood-causing mechanisms considered in the analysis (e.g., hydrology, meteorology, oceanography) to develop an understanding of assumptions, determine the validity of the methods used in those deterministic assessments, and account for technically defensible interpretations of available data, models, and methods, which may vary based on the severity of floods to which frequencies must be assigned. As flooding SDP analyses are often analyzed using Inspection Manual Chapter (IMC) 0609 Appendix M, "Technical Basis for the Significance Determination Process (SDP) Using Qualitative Criteria," the deterministic assessment is one input to overall qualitative assessment of the events.

A flood of a given severity can occur from any combination of constituent contributing factors (e.g., combinations of climatological, meteorological, and antecedent conditions) or mechanisms (e.g., storm surge concurrent with a, potentially dependent, river flood at a site located near where the river enters the ocean). The risk analyst should ensure that the hazard assessment captures hazard contributions from all relevant flood-causing mechanisms and combinations of events. Examples of potentially relevant combinations of flood mechanisms may include (but are not limited to):

- River floods with concurrent site precipitation and wind-generated waves;
- Basin-wide precipitation along with snowmelt leading to river flooding;
- Seismic dam failures with concurrent river flood;
- River flooding concurrent with storm surge event;
- Storm surge events concurrent with high winds and precipitation; and
- High water level concurrent with seiche.

To address relevant mechanisms, the licensees may develop a composite flooding hazard curve combining all plausible mechanisms to obtain a single hazard curve. The analyst should note that sites that are affected by multiple flood hazards may use different strategies to protect against or mitigate the different flood hazards and, therefore, it may be inappropriate to consider a single composite hazard curve and plant fragility function.

In characterizing flood hazards, it is important to consider associated effects in addition to flood height (i.e., factors in addition to stillwater elevation such as wind waves and run-up effects; hydrodynamic loading, including debris; effects caused by sediment deposition and erosion; concurrent site conditions, including adverse weather conditions such as wind; groundwater ingress; and other pertinent factors, see JLD-ISG-2012-05, "Guidance for Performing the Integrated Assessment for External Flooding" for additional information). Flood event duration should also be considered in addition to the hazard flood height and associated effects in characterization of the hazard.

6.3.4 Sensitivity and Uncertainty

PRAs typically utilize the mean hazard for the frequency of occurrence of different external flood severities. However, significant insights can be gained as a result of understanding the uncertainties in the hazard. Consideration of these uncertainties is an important component of a risk-informed regulatory process.[7]

Appropriate treatment of aleatory variability and epistemic uncertainties allows for decision making on a range of relevant factors (e.g., insights from mean hazard curves as well as the comparison of the mean to various fractile hazard curves). The spatial, temporal and other relevant characteristics of future realizations of meteorological, climatological, hydrological, hydraulic, or other parameters typically is associated with aleatory variability and expressed by a hazard curve. There are various options for addressing epistemic uncertainty (e.g., in

[7] NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," describes treatment of uncertainties in PRAs used for risk-informed decision making. EPRI 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," provides general guidance for the treatment of uncertainties in PRAs to supplement and complement the guidance in NUREG-1855. NUREG-1855 also references EPRI 1026511, "Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," which supplements guidance in EPRI 1016737.

probabilistic seismic hazard analysis, it is common to treat epistemic uncertainty through logic trees). Epistemic uncertainty is expressed by incorporating multiple assumptions and technically defensible data, models, and methods using multiple hazard curves from which a mean, median or other fractile hazard curve can be derived. While the mean and median hazard curves convey the central tendency of the calculated exceedance frequencies, the separation among fractile curves conveys the effect of uncertainties. Examples of epistemic uncertainty include: the selection of the probability distribution that is appropriate for capturing aleatory uncertainty in parameters, selection of a technique to parse available datasets for relevance to a particular site, the appropriate hydrologic or hydraulic model to use, and the choice of various parameters needed to utilize existing models.

New information related to hazard assessment beyond the information available in sources such as FSARs or IPEEEs may become available in licensees' analyses of the flooding events. The validity of the new information should be assessed and the appropriate manner that the new information should be considered in making the risk-informed decision should be determined.

6.4 Flood Protection Measures

6.4.1 Reliability of Flood Barriers and Flood Protection Features

There are vast differences from plant to plant with regard to the flood protection features used. Examples of these features are provided in JLD-ISG-2012-05, which, in part, provides generic guidance on performing an evaluation of the capability of the site flood protection to protect systems, structures, and components (SSCs) important to safety for each set of flood scenario parameters. The generic guidance in JLD-ISG-2012-05 does not provide detailed guidance on determining different failure modes of various physical barriers or evaluating and quantifying the reliability of the degraded physical barriers.

During recent plant walkdowns and flooding events, many instances have been found where physical barriers credited to protect safety-related SSCs from inundation and static/dynamic effects of external floods were not able to reliably accommodate the flood scenario parameters. Appendix 6A provides examples of deficiencies in flood protection features analyzed through SDP. These examples include various deficiencies such as degradation of storm drain capacity and degraded conduits that lacked flooding barriers (Example A.1), degradation of penetration and conduit seals, unsealed shims, gap in the weather stripping along door, unsealed pump leakoff hub drains (Example A.2), electrical conduit penetration seals (Example A.3), conduit couplings in the air intake tunnel (Example A.4), and unsealed penetrations (Example A.5 and Example A.6). These deficiencies revealed pathways that were not effectively sealed against flooding and affected or could affect equipment in the auxiliary building (Example A.1, Example A.4, and Example A.6), reactor building, EDG building, service water building (Example A.2), and battery rooms (Example A.5). No method or guidance is available at this time for the wide range of physical flood barriers to appropriately assess the performance of the degraded physical barriers and account for their reliability at both the feature- and system-levels during a postulated design basis flooding event. In a case-by-case basis, the risk analyst must attempt to obtain information on the nature of failure modes of the flood protection system under review and consider potential ingress pathways for floodwaters (e.g., through conduits or ducts). In analyzing past events, in which the performance deficiencies were related to degraded physical flood barriers, PRA analysts have not typically given any credits to flood protection features intended to protect a specific SSC or group of SSCs and those SSCs were assumed to be failed in the analysis. As additional information is provided by a licensee, the assumption for complete failure of the barrier and, consequently, the protected SSCs may be revised and appropriate credits may be considered.

As stated in, "Recommendations for Enhancing Reactor Safety in the 21st Century, The Near Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," the flooding risks are of concern due to a "cliff-edge" effect, in that the safely consequences of a flooding event may increase sharply with a small increase in the flooding level. Therefore, the risk analyst should be aware and consider potential impact on the results of analyses due to this effect.

6.4.2 Human Reliability Considerations

Human reliability analysis (HRA) methods for evaluating flood mitigation actions, such as construction of flood protection are not well established. In the absence of specific guidance for modeling HRA in external flooding events, PRA analysts have used SPAR-H to quantify human error probabilities (HEPs) in analyzing the past performance deficiencies. Although the focus of SPAR-H is on at-power and low-power/shutdown HEP determination and there may be limitations to SPAR-H, such as when dealing with very long term actions, heuristics described in NUREG/CR-6883, "The SPAR-H Human Reliability Analysis Method," may apply to other situations such as fire, flood, seismic events to estimate the HEPs associated with the new or re-quantified flood mitigation actions. By utilizing performance shaping factors (PSFs), the SPAR-H provides a framework that account for factors, such as timeliness, procedures, training and stress, which could significantly affect the risk in external flooding events.

For evaluating the human interactions to implement the site-wide flood mitigation strategies, the analyst may need to consider new basic events representing those human actions. The analyst may also need to consider whether any human actions embedded in the SPAR model should be re-quantified to account for specific conditions resulted from the external flooding. Once the human failure events (HFEs) are identified and characterized, the analyst must identify the salient performance drivers by reviewing SPAR-H PSFs. Each PSF needs to be examined with respect to the context of the HFE. Appendix C to JLD-ISG-2012-05 adopts SPAR-H guidance and provides guidance on assigning PSF levels in the context of external flooding. Appendix 6B to this guidance provides a brief discussion of PSFs along with examples of past SDP analyses related to external flooding. The past SDP analyses show that a number of HEPs or multiple PSFs could be affected by the performance deficiencies on external flooding. For instance, in Example B.1, the HEP for plant workers failing to install levee/bin wall flood barriers was assumed 1.0 (always failed) for the deficient case because of inadequate time. The other HEPs affected by the performance deficiency in Example B.1 included the HEP for failing to protect the reactor building from flooding via alternate means (such as sandbagging) and the HEP for manual operation of reactor core isolation cooling system (RCIC) and the hard pipe vent during extended station blackout. A significant credit was not given for protecting the reactor building via alternate means because the timing, plant configuration, staffing, etc., when it is realized that the reactor building needs to be protected via this option was unclear. The risk analyst also found, for this specific example, the operation of RCIC and the hard pipe vent under an extended station blackout with significant site-wide flooding, which was expected to last for several days, to be challenging. The procedure for operating RCIC without electric power stated that reactor level and RCIC turbine speed may not be available. Radiation, temperature, lighting, etc., may also represent challenges to the operators. In Example B.3, an assessment was performed to determine the feasibility of providing inventory make-up during a flooding event. In this example, although "available time" was significantly greater than the estimated time, procedures that identify the need for make-up did not provide sufficient detail to connect to a primary system above the peak flood level, which adversely affected the PSF for procedures.

In addition, after loss of AC power and subsequent depletion of the 125 Volt DC batteries, control room instrumentation would become unavailable, leaving local instruments as the primary source for plant information, which affected the "ergonomics/human machine interface" PSF.

Because of the importance of operator actions in mitigating strategies during external flooding events, the possible need for special treatment of some ex-control room actions, and the large uncertainties that may exist in HFE estimates, the contribution of HFEs on the risk may vary substantially depending on the assumptions. The PRA analyst may request assistance form NRR/DRA for subject matter expert to expedite this analysis by providing insights on adding or re-quantifying HFEs of high significance and possibly deviating from the SPAR-H guidance as necessary. The analyst should also consider the guidance in Section 9.4 of Volume 1 of this handbook to account for dependencies between HFEs for operator actions during external flooding events.

In external flooding analyses, the "diagnosis" component of HFEs could potentially be significant for some human actions. In particular, when the analyst must qualitatively consider recovery actions that licensees want the staff to consider when those actions are not proceduralized, careful treatment of the diagnostic component of the human error is critical. During the analysis of human actions for events or conditions associated with external floods, licensees may provide information to justify crediting various operator actions that could have been implemented using resources available. In some cases, the licensee attempted to demonstrate feasibility or reliability of human actions after an event has occurred and asked the staff to provide credit for such actions, even though procedures or training may not exist for those actions. As Regulatory Issue Summary (RIS) 2008-15, "NRC Staff Position on Crediting Mitigating Strategies Implemented in Response to Security Orders in Risk-Informed Licensing Actions and in the Significance Determination Process," states, manual actions must be included in plant procedures and staff be trained to perform the actions in the context they are to be credited in order for the licensee to receive realistic quantitative credit in a risk assessment. The revision to RIS 2008-15 further states that, although quantitative credits for those actions are not warranted, the NRC will consider licensees' analyses for providing qualitative insights in cases where procedures for those human actions are under development or those human actions are included in other relevant procedures that could be reasonably identified and utilized for mitigating external flooding events. The NRC will rely more on historical evidence or supplemental inspections that demonstrate the successful feasibility and reliability of the actions that existed prior to the event, rather than information gathered after the event, and consider whether any credit may be given qualitatively in a risk-informed process using IMC 0609 Appendix M.

Appendix 6A Examples of Deficiencies in Flood Protection Features

Example A.1

On January 9, 2014, St. Lucie Unit 1 was operating at 100 percent reactor power when the site experienced a period of unusually heavy rainfall. Although this event was below the design basis flood, St. Lucie declared an unusual event because of storm drain capacity degradation. Blockage in the site's storm drain system caused water to backup within the emergency core cooling system pipe tunnel outside of the Unit 1 reactor auxiliary building. Water entered the auxiliary building through two degraded conduits that lacked internal flood barriers. An extent-of-condition review identified four additional conduits on Unit 1 that lacked the required internal flood barriers. The modification that had installed the conduits had not considered the need for internal flood barriers for conduits installed below the design-basis flood elevation. Previous walkdowns performed in 2012 using the guidance contained in Nuclear Energy Institute (NEI) 12-07, "Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features," had failed to identify the degraded conduit or the missing conduit internal flood barriers. Additionally, St. Lucie determined that previous engineering evaluations used to assess the results of the walkdowns did not account for the site flood inundation times and therefore underestimated the volume of external flood leakage through degraded flood barriers. The licensee implemented corrective actions that included installing qualified internal water seals on all of the affected conduits. Additional information regarding this event is available in licensee event report (LER) 335/2014-001 and in NRC integrated inspection report (IR) 05000335/2014009. In a letter dated November 19, 2014, (ADAMS Accession No. ML14323A786), the NRC issued the final significance determination, which characterized the finding as White (i.e., low-to-moderate safety significance).

Example A.2

On April 20, 2011, NRC inspectors at Brunswick identified that the emergency diesel generator (EDG) fuel oil tank chamber enclosure contained openings that would adversely impact the ability to mitigate external flooding of the oil tank chambers in the event of a probable maximum hurricane. The licensee subsequently performed extent-of-condition walk downs and identified numerous examples of degraded or nonconforming flood protection features, the majority of which were flood penetration seals. During walkdowns of flood protection features in accordance with NEI 12-07 in 2012, the licensee identified additional degradation in the reactor buildings and the EDG building, specifically degraded flood penetration seals, conduit seals, and a 3-inch gap in the weather stripping along the bottom of the Unit 2 reactor building railroad door. This gap would have allowed leakage into the reactor building during a PMH. The inspectors also identified an EDG rollup door that could have allowed water intrusion into the EDG building during a PMH. Additionally, the licensee identified unsealed shims under the base plates of the service water pumps, as well as leaking flood penetration seals and an unsealed conduit in the service water building that could have allowed floodwater to enter the building during a PMH. The licensee also identified a potential flood pathway from the intake canal into the service water building through unsealed pump leak off hub drains, a condition that had existed since construction of the plant. These conditions were caused by a historical lack of a flood protection program. Multiple examples were identified where credited flood mitigation equipment had no established preventative maintenance program. Corrective actions included correcting the degraded seals, developing and implementing an engineering program to mitigate consequences of external flooding, and developing topical design basis for internal and external flooding. Additional information regarding this issue is available in IR 05000325/2011014, (ADAMS Accession No. ML113610594). The NRC characterized the finding as White (i.e., low-to-moderate safety significance).

Example A.3

On December 12, 2012, the licensee at Sequoyah Nuclear Plant performed an inspection of an electrical manway and confirmed that inadequate electrical conduit penetration seals provided an in-leakage path into the essential raw cooling water (ERCW) pumping station. The licensee concluded that an external flooding event exceeding the elevation that would impact the conduits would inundate the ERCW pumping station, with impacts to both Unit 1 and Unit 2. The nonconforming seals would have allowed floodwaters to enter the pumping station at a rate greater than the capacity of the sump pump and could have resulted in the ERCW system being unavailable to perform its design function during a flood event below plant grade. Although the electrical conduit penetration seals were meant to be the flood barrier, there was no clear identification of the flood barriers and their requirements. The licensee took corrective actions that included installing qualified conduit seals and revising design-basis documents and flood barrier drawings to identify flood boundaries and to include seal details. Additional information regarding this issue is available in LER 327/2012-001 and in IR 05000327/2013011. The NRC characterized the finding as White (i.e., low-to-moderate safety significance).

Example A.4

On August 2, 2012, while observing the licensee flooding walkdowns at Three Mile Island Station, NRC inspectors noted degradation on several conduit couplings in the air intake tunnel. The air intake tunnel provides a source of air for safety-related ventilation systems and also contains both safety- and nonsafety-related electrical conduits. The couplings, which by design should have been injected with sealant to provide a barrier to design-basis flooding events, showed signs of exposure to wet environments, indicating that the sealant was missing. The licensee eventually determined that 43 conduit couplings were missing sealant. The original construction deficiency had not been identified by the licensee during a comprehensive review performed in 2010. Without adequate protection from flooding, floodwater could have bypassed all flood barriers through the conduits and impacted the operability of decay heat removal equipment. The licensee implemented prompt compensatory actions, including staging extra sandbags and earth moving equipment to restore operability of the flood barriers. The licensee implemented permanent corrective actions that included sealing the conduits by injecting watertight qualified sealant material into the associated cable conduits. Additional information regarding this issue is available in IR 05000289/2012005. In a letter dated April 30, 2013, (ADAMS Accession No. ML13120A040), the NRC issued the final significance determination, which characterized the finding as White (i.e., low-to-moderate safety significance).

Example A.5

On May 29, 2013, while performing flooding walkdowns in accordance with NEI 12-07, the licensee at R.E. Ginna Nuclear Power Plant discovered two penetrations that appeared to be unsealed leading to one of the battery rooms. Although the licensee determined that drains in the manhole would prevent the water level from reaching the unsealed penetrations, NRC inspectors raised questions about the operability of these drains, since they were not included in any maintenance or test program. In response to these questions, the licensee tested the drains and determined that they were not capable of draining enough water to prevent a design-basis flood from reaching the unsealed penetrations and flooding battery room B. Battery room A would also be flooded by a non-watertight fire door that connects it with battery room B. The potential existed to also lose offsite power leading to the loss of all alternating current power to the site and an unrecoverable station blackout. In 1983, as part of the systematic evaluation process, the licensee's design basis was changed to include additional external flooding events and the flood protection level was agreed to by the licensee at a level that was above the elevation of the manhole. The licensee did not evaluate the potential for flooding through the manhole and, therefore, did not seal the cable penetrations that were at an elevation below the new level. The licensee took corrective actions that included installing permanent hydrostatic seals in both penetrations between the manhole and the battery room. Additional information regarding this issue is available in IR 05000244/2013005. In a letter dated April 17, 2014, (ADAMS Accession No. ML14107A080), the NRC issued the final significance determination, which characterized the finding as White (i.e., low-to-moderate safety significance).

Example A.6

On March 31, 2013, following the collapse of a temporary lifting rig carrying the Arkansas Nuclear One Unit 1 main turbine generator stator, a rupture in the fire water system resulted in water leakage past floor plugs in the auxiliary building and subsequent accumulation of water inflow in the safety-related decay heat removal room B through a room drain pipe. This event overlapped the timeframe in which the licensee was assessing flood mitigation features in response to Fukushima-related orders issued by the NRC. The extent of condition reviews by the licensee related to this event and those discrepancies identified during flood mitigation response efforts found numerous other pathways that were not effectively sealed against flooding in the auxiliary building and emergency diesel fuel storage buildings. The licensee's failure to design, construct, and maintain the Unit 1 and Unit 2 auxiliary and emergency diesel fuel storage buildings so that they would protect safety-related equipment during design-basis flood events caused the overall condition. The unsealed penetrations were not identified during the walkdowns because of incomplete information on flooding barriers, some information not being kept current, and inadequate oversight of the contractor performing the flood protection walkdowns. The licensee took corrective actions that included re-performing the reviews of essential flood protection features, identifying those features that were initially not identified, completing the missed portions of the walkdowns, and submitting corrected information to the NRC. In this event, an internal flooding event resulted in the licensee discovering external flooding vulnerabilities. Additional information regarding this issue is available in IR 05000313/2014009. In a letter dated January 22, 2015, (ADAMS Accession No. ML15023A076), the NRC issued the final significance determination, which characterized the finding as Yellow (i.e., substantial safety significance).

Example A.7

NRC inspectors at Brunswick identified that the licensee failed to identify and correct conditions adverse to quality involving degraded and nonconforming flood penetration seals and openings in multiple safety-related buildings. In August and September of 2012, the licensee performed walkdowns of flood protection features in accordance with NEI 12-07 and identified degraded and/or nonconforming flood protection features, the majority of which were attributed to degraded or nonconforming flood penetration seals. Based on these findings, the inspectors determined the licensee had not fully identified all of the degraded flood protection features during walkdowns in 2011, as a result of the EDG fuel oil tank chamber flooding issue. The licensee also identified flood protection feature degradation in the service water building (SWB). The licensee identified a potential flood pathway from the service water pump (SWP) intake to the 20-foot elevation of the SWB through unsealed SWP leakoff hub drains. A combination of the deficiencies in the SWB flood protection, and an additional ground caused through other building inleakage, could result in failure of the service water pumps during flooding events. Licensee representatives provided an assessment of the significance of the findings, the root cause evaluation, corrective actions taken and planned, and the methodology used to evaluate storm surge and flooding. The discussion included information which addressed the sources of uncertainty identified in the preliminary significance calculation performed by the NRC. Descriptions of the testing performed to determine the flow characteristics of the penetrations used in the licensee's calculations of the inleakage rates were also presented. The results of flooding calculations for both the SWB and the high-pressure coolant injection room in the reactor building, which demonstrated increased margin to immersion of critical equipment were also discussed. In addition, the duration of an assumed maximum storm surge flood was presented using the results of state of the art methodologies. The NRC concluded that the Unit 1 preliminary Green finding was appropriately characterized, and the Unit 2 preliminary White finding should be re-characterized as a Green finding, an issue of very low safety significance. Additional information regarding this issue is available in IR 05000324/2014011.

Appendix 6B Examples of Factors Affecting Operator Actions to Flood Events

This appendix provides a list of PSFs identified in Appendix C to JLD-ISG-2012-05 adapted from SPAR-H methodology and a generic description of circumstances related to external flooding events that could necessitate considering any of these factors as a performance drive or re-quantifying the HFEs. JLD-ISG-2012-05 provides a more detailed discussion of these PSFs in the context of flooding.

B.1 Available Time

Reviewing recent findings identified a number of instances in which available time to implement mitigating strategies was not sufficient because of variety of issues such as not accounting for particular steps in planning, not considering the sequential manner that some activities in the implementing procedures should be directed, underestimating the time to perform some of the more complex and coordinated work activities, etc. Following is an example for which the available time was either inadequate or barely adequate (Example B.2 and Example B.4 are also related). For those actions with inadequate available time, the probability of failure is one.

Example B.1

During an inspection from September 12, 2012, to May 15, 2013, NRC inspectors identified that the Monticello Nuclear Generating Plant site failed to maintain a flood mitigation procedure such that it could support the implementation of flood protection activities within the 12-day timeframe credited in the USAR to protect against a probable maximum flood (PMF) event. The licensee believed that flood mitigation actions for the protected area could be taken within the 12 days specified in the USAR by citing an independent engineering assessment performed in 2001. However, the licensee did not perform a verification walkthrough of the activities in the procedure and, therefore, did not identify vulnerabilities in its flood plan. The licensee took corrective actions, which included revising its procedure to add more detail, as well as pre-staging materials necessary to complete the bin wall in the timeframe cited in the updated safety analysis report (USAR). Additional information regarding this issue is available in IR 05000263/2013008. In a letter dated August 28, 2013 (ADAMS Accession No. ML13240A435), the NRC issued the final significance determination, which characterized the finding as Yellow (i.e., substantial safety significance). Although not a typical application of the available time PSF, this example demonstrates how it can be utilized broadly.

B.2 Accessibility

Actions that must be performed in inundated areas or requiring operators and/or equipment to travel through inundated areas, should be considered infeasible unless it can be shown that elevated pathways or other means are available to enable movement through the inundated areas and significant hazards to operators (e.g., electrical hazards due to presence of water, low temperatures, etc.) are not present. Other accessibility issues include obstructions (e.g., charge fire hoses) and locked doors.

B.3 Environmental/Stress Factors

Stress refers to the level of undesirable conditions and circumstances that impede the operator from easily completing a task. Stress can include mental stress, excessive workload, or physical stress (such as that imposed by difficult environmental factors). During an external flooding event, environmental conditions could affect and impede completion of actions. The environmental conditions associated with flood events that could increase stress include: adverse weather (e.g., lightning, hail, wind, and precipitation), temperatures (e.g., air and water temperatures, particularly if operators must enter water), conditions hazardous to the health and safety of operators (e.g., electrical hazards, hazards beneath the water surface, drowning), lighting, humidity, radiation, and noise.

B.4 Diagnostic Complexity, Indications and Cues

In the context of flooding, indications should be available to provide notification that a flood event is imminent if operator actions are required to provide protection against the flood event. Examples of indications include river forecasts, dam condition reports, and river gauges. Appendix C to JLD-ISG-2012-05 states that any operator manual action initiated by the indication should be considered infeasible, if durable agreements are not in place to ensure communication from offsite entities and the plant does not have an independent capability to obtain the same information onsite. Consideration should be given to the quality of the agreements in place between offsite entities and operators at the nuclear power plant site as well as the potential for the communication mechanisms to fail.

In the context of mitigation actions, indications should be available to alert operators to the failure of flood protection features and presence of water in locations that are intended to be kept dry or otherwise protected from flood effects. For cases in which indications are not available, the evaluation can consider compensatory measures (e.g., local operator observations). If cues or indications are not available to operators, the mitigation actions should be considered infeasible.

B.5 Experience and Training

This factor refers to the experience and training of the operator(s) involved in the task. Included in this consideration are years of experience of the individual or crew, and whether or not the operator/crew has been trained on the type of accident, the amount of time passed since training, and the systems involved in the task and scenario. As some licensees have not fully developed flood mitigation procedures or they have not been adequately trained on implementing those procedures, this factor could become a performance driver. The following are examples related to experience and training.

Example B.2

In 2013, the licensee at Watts Bar identified that it could not demonstrate the capability to implement site external flood mitigation procedures in the time assumed between the notification of an imminent design-basis flood event and flood waters reaching the Watts Bar site. The design-basis flood event for Watts Bar would result in flooding above plant grade. Accordingly, the licensee relied on procedures used to reconfigure plant systems in preparation for site inundation to ensure the ability to safely shut down the reactor and remove decay heat. Examples of issues that challenged the assurance that the flood mitigation procedures could be implemented within the available time included: work activities in the implementing procedures were directed in a sequential manner, which added to the overall time required; piping interferences and the lack of suitable rigging locations for inter-system spool pieces; mislabeled or missing equipment was used in the implementing procedures; the time to perform some of the more complex and coordinated work activities was underestimated. Other PSFs that were considered during the assessment were the available time, which was determined to be approximately the time required; stress, which was determined to be high; and procedure guidance, which was determined to be incomplete. The licensee took corrective actions that included revising the flood mitigation procedures to add more detail, increasing the frequency of the training for the procedures, and staging equipment and developing preventive maintenance activities to periodically validate that the equipment is in place. Additional information regarding this issue is available in IR 05000390/2013009. The NRC characterized the finding as White (i.e., low-to-moderate safety significance).

B.6 Procedures

This factor refers to the existence and use of formal operating procedures for the tasks needed for mitigating an external flooding event. In evaluating the feasibility of an operator manual action, the quality of procedures should be assessed based on its ability to assist operators in correctly diagnosing an impending flood event (i.e., flood height and associated effects) or the compromise of a flood protection feature, to identify the appropriate preventative (or mitigation) actions and to account for prevailing current conditions, if applicable (e.g., high wind or lightning that makes it difficult for operators to work outdoors). The following examples illustrate cases where procedures were not available or incomplete.

Example B.3

In August 2012, while observing licensee simulations at Dresden Nuclear Power Station, Units 2 and 3 for executing flood protection procedures as part of the NEI 12-07 walkdowns, NRC inspectors noted that the procedures did not account for reactor coolant system (RCS) inventory losses. The procedures assumed a flood duration of 4 days, during which time systems that provide normal and makeup capacity to the RCS would be flooded and unavailable. The licensee calculations accounted for the 5-gallon per minute (gpm) maximum technical specification allowance for unidentified RCS leakage, but it did not account for inventory losses from identified leakage, which could be as high as an additional 20 gpm. The licensee strategy did not originally provide for a method to maintain RCS inventory above the top of active fuel for RCS leakage rates that were allowable under technical specifications. The licensee took corrective actions, including modifying procedures to provide makeup capacity and to isolate the reactor recirculation loops during flood conditions when reactor vessel makeup capabilities are limited so that sources of identified leakage would no longer impact the reactor vessel level. Additional information regarding this issue is available in IR 05000237/2013002. In a letter dated July 31, 2013 (ADAMS Accession No. ML13213A073), the NRC issued the final significance determination, which characterized the finding as White (i.e., low-to-moderate safety significance).

Example B.4

In March 2013, inspectors found that the Point Beach Nuclear Plant licensee failed to establish procedural requirements to implement external wave run-up protection design features as described in the FSAR. Flood protection procedures directed installation of concrete jersey barriers to protect the turbine building and pumphouse from flooding. While performing the flooding walkdowns, the licensee discovered, among other issues, that when the barriers were installed, gaps were created and there were no provisions in the procedure for using sandbags to protect the openings in the jersey barriers or the gaps between the barriers and the ground. The licensee also had failed to consider the time that would be required to erect the barriers. The licensee took corrective actions, including modifying existing jersey barriers to eliminate openings, revising the procedure to direct the installation of jersey barriers in conjunction with sandbags, and pre-staging additional sandbags and jersey barriers. Additional information regarding this issue is available in IR 05000266/2013002. In a letter dated August 9, 2013 (ADAMS Accession No. ML13221A187), the NRC issued the final significance determination, which characterized the finding as White (i.e., low-to-moderate safety significance).

Example B.5

In September 2009, during a component design basis inspection at Fort Calhoun Station, NRC inspectors identified that the licensee failed to maintain adequate procedures to protect the intake structure and auxiliary building during external flooding events. These procedures described stacking and draping sandbags on top of installed floodgates to protect the plant up to the flood elevation described in the USAR. When inspectors asked plant staff to demonstrate this procedure, they were unable to complete the procedure as written because the cross section on the top of the floodgates was too small to accommodate enough sandbags to retain a 5-foot static head of water. The licensee took corrective actions that included revising the procedures. Additional information regarding this issue is available in IR 05000285/2010007. In a letter dated October 6, 2010 (ADAMS Accession No. ML102800342), the NRC issued the final significance determination, which characterized the finding as Yellow (i.e., substantial safety significance).

B.7 Staffing

In assessing the feasibility and reliability of an operator manual action, the persons involved in performing the operator manual action should be qualified. The feasibility assessment should consider the availability of a sufficient number of trained operators without collateral duties during a flood event such that the required operator action can be completed as needed. In evaluating the reliability of an operator manual action, uncertainties in the number of operators onsite (or that can be brought in from offsite) should be considered.

B.8 Communication

Equipment may be required to support communication between operators to ensure the proper performance of manual actions (e.g., to support the performance of sequential actions and to verify procedural steps). Also because of the long durations of many flooding scenarios and because of the possible need of offsite support, communication with corporate and governmental organizations is important. Therefore, consideration of the causes of the floods impact on offsite communications must be considered. Consideration should be given to whether operators are trained to ensure effective communication and coordination during a flood event.

B.9 Human Factors Engineering

Human factors engineering refers to the equipment, displays and controls, layout, quality, and quantity of information available from instrumentation, and the interaction of the operator/crew with the equipment to carry out tasks. Many of the human actions anticipated for dealing with floods will be external to the main control room. As such, it is not the layout and design of the controls and annunciators in the control room that are of primary concern but instead those external to the control room. In Example B.2, one of the challenges in implementing flood mitigation procedures was the use of mislabeled or missing equipment in the implementing procedures.

Appendix 6C Dam Failure Rates for External Flooding

Dam failure is well documented and can be characterized by type of dam. Table 6-2 is a summary of point estimate failure rates for dams that are broken down by large dams (greater than 50 feet) and all sized dams. Characteristics of U.S. dams and dam failures are available at the National Inventory of Dams and the National Performance of Dams Program.

Of the 79,777 dams in the United States, 72 percent are embankment type and 28 percent are concrete. Nineteenth century dams would fail at 5 percent in the first 5 years after construction but would settle out to a 1–4 percent additional failure by 20 years of life. This was reduced to 2 percent in the first 5 years for dams built after 1930. By 1960, dam failure rates were less than 0.01 percent due to better engineering. Whatever the era, half of all dams that ever fail, do so in the first 5 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 1×10^{-4} and 4×10^{-4}, even for blue-sky events. Determining flood levels, however, is a complex matter. The U.S. Army Corps of Engineers has software named Hydraulic Engineering Center (HEC) that when combined with Geographic Information System (GIS) data will model river flow and flooding in great detail.

Table 6-2. 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

Notes:
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

Notes:
Prior distribution obtained using empirical Bayes method in SAS.
Dams constructed with mixed materials are not counted; dams with no construction dates available are not counted.

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

1. Objective

This Appendix 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.

2. Input

The inputs to these calculations are:

- Seismic initiating event frequencies (seismic hazard distribution) as a function of seismic g-level obtained from licensees' submittals as part of the effort to address NRC Near-Term Task Force (NTTF) Recommendation 2.1 in 2014 and 2015); and
- Structures, systems and components (SSCs) (e.g., ceramic insulator) fragilities as a function of g level (NUREG-6544, "A Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences").

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 A-1:

- Seismic initiating event mean frequency of a 0.05g or higher earthquake per year,
- Given an earthquake occurs, the conditional LOOP probability caused by the earthquake (based on failure of ceramic insulators), and
- Frequency of seismically induced LOOP event (per year).

Table A-2 and Table A-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

- 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, AC 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.

- 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.

- For the sites to the east of the Rocky Mountains, an EXCEL workbook is used to calculate the seismically-induced LOOP frequencies for 61 sites. The EXCEL file can be found with ADAMS Accession No. ML11220A195 as well as in the RASP Tool Box Web site (internal use only). The same generic ceramic insulator seismic fragility distribution is used for these calculations.

- For the three U.S. nuclear power plants west of the Rocky Mountains, plant-specific seismic event frequency distributions (seismic hazard curves) are obtained from licensees' submittals as part of the effort to address NRC NTTF Recommendation 2.1 in 2015. The seismic fragility distributions for LOOP are obtained from the Individual Plant Examination of External Events (IPEEE) submittals whenever available. If not available, generic ceramic insulator fragilities are used

- The calculations can be readily customized for plant-specific SSC fragilities (e.g., ceramic insulators) and/or hazard curves. The EXCEL workbook (ADAMS Accession No. ML11220A195) is available for this purpose.

Table A-0-1. Frequencies of Seismically-Induced LOOP Events (Based on hazard vectors in NTTF 2.1 submittals)

	Plant	Seismic IE_{freq}	Conditional LOOP Probability	Seismically-Induced LOOP Frequency	Plant Type	# of Units
		A	B	A*B		
1–2	ANO 1 & 2	1.22E-03	5.22E-02	6.36E-05	B&W/CE	2
3–4	Beaver Valley 1 & 2	7.68E-04	5.02E-02	3.85E-05	W	2
5–6	Braidwood 1 & 2	6.31E-04	5.90E-02	3.72E-05	W	2
7–89	Browns Ferry 1, 2 & 3	1.46E-03	5.79E-02	8.45E-05	BWR	3
10–11	Brunswick 1 & 2	8.00E-04	5.26E-02	4.21E-05	BWR	2
12–13	Byron 1 & 2	6.41E-04	8.23E-02	5.28E-05	W	2
14	Callaway	4.54E-03	8.17E-02	3.71E-04	W	1
15–16	Calvert Cliffs 1 & 2	2.81E-04	3.96E-02	1.11E-05	CE	2
17–18	Catawba 1 & 2	1.22E-03	7.11E-02	8.68E-05	W	2
19	Clinton	2.20E-03	4.09E-02	9.01E-05	BWR	1
20	Columbia	3.94E-03	1.04E-01	4.09E-04	BWR	1
21–22	Comanche Peak 1 & 2	7.27E-05	3.56E-02	2.59E-06	W	2
23–24	Cook 1 & 2	9.06E-04	6.06E-02	5.49E-05	W	2
25	Cooper	2.94E-04	5.43E-02	1.60E-05	BWR	1
26	Davis-Besse	4.14E-04	5.94E-02	2.46E-05	B&W	1
27–28	Diablo Canyon 1 & 2	2.30E-02	1.08E-03	2.48E-05	W	2
29–30	Dresden 2 & 3	7.21E-04	6.90E-02	4.97E-05	BWR	2
31	Duane Arnold	1.12E-04	5.99E-02	6.71E-06	BWR	1
32–33	Farley 1 & 2	7.25E-05	7.43E-02	5.39E-06	W	2
34	Fermi 2	3.91E-04	6.78E-02	2.65E-05	BWR	1
35	Fitzpatrick	2.91E-04	4.39E-02	1.28E-05	BWR	1
36	Fort Calhoun	6.95E-04	5.56E-02	3.87E-05	CE	1

	Plant	Seismic IE_{freq}	Conditional LOOP Probability	Seismically-Induced LOOP Frequency	Plant Type	# of Units
		A	**B**	**A*B**		
37	Ginna	2.07E-04	5.78E-02	1.20E-05	W	1
38	Grand Gulf	2.31E-04	2.99E-02	6.91E-06	BWR	1
39–40	Hatch 1 & 2	6.87E-04	3.55E-02	2.44E-05	BWR	2
41	Hope Creek	4.46E-04	6.05E-02	2.70E-05	BWR	1
42–43	Indian Point 2 & 3	7.04E-04	1.31E-01	9.21E-05	W	2
44–45	LaSalle 1 & 2	1.98E-03	5.61E-02	1.11E-04	BWR	2
46–47	Limerick 1 & 2	4.47E-04	6.95E-02	3.11E-05	BWR	2
48–49	McGuire 1 & 2	9.66E-04	7.10E-02	6.86E-05	W	2
50–51	Millstone 2 & 3	4.13E-04	6.87E-02	2.84E-05	CE/W	2
52	Monticello	2.44E-04	7.56E-02	1.84E-05	BWR	1
53–54	Nine Mile Point 1 & 2	3.00E-04	4.42E-02	1.33E-05	BWR	2
55–56	North Anna 1 & 2	1.07E-03	1.67E-01	1.78E-04	W	2
57–59	Oconee 1, 2, & 3	1.31E-03	8.26E-02	1.08E-04	B&W	3
60	Oyster Creek	1.01E-03	3.64E-02	3.68E-05	BWR	1
61	Palisades	1.29E-03	5.90E-02	7.61E-05	CE	1
62–64	Palo Verde 1, 2, & 3	3.97E-04	5.32E-02	2.11E-05	CE	3
65–66	Peach Bottom 2 & 3	6.70E-04	1.27E-01	8.53E-05	BWR	2
67	Perry	3.86E-04	7.70E-02	2.97E-05	BWR	1
68	Pilgrim	1.39E-03	1.15E-01	1.60E-04	BWR	1
69–70	Point Beach 1 & 2	3.69E-04	4.71E-02	1.74E-05	W	2
71–72	Prairie Island 1 & 2	5.48E-05	5.93E-02	3.25E-06	W	2
73–74	Quad Cities 1 & 2	3.21E-04	6.40E-02	2.05E-05	BWR	2
75	River Bend	1.80E-04	5.20E-02	9.36E-06	BWR	1
76	Robinson 2	3.72E-03	9.75E-02	3.63E-04	W	1
77–78	Saint Lucie 1 & 2	5.21E-05	5.04E-02	2.63E-06	CE	2
79–80	Salem 1 & 2	3.67E-04	5.72E-02	2.10E-05	W	2
81	Seabrook	1.11E-03	1.29E-01	1.44E-04	W	1
82–83	Sequoyah 1 & 2	2.39E-04	2.42E-02	5.79E-06	W	2
84	Shearon Harris	3.31E-04	3.38E-02	1.12E-05	W	1
85–86	South Texas 1 & 2	4.36E-04	2.67E-02	1.16E-05	W	2
87–88	Surry 1 & 2	4.36E-04	2.67E-02	1.16E-05	W	2
89–90	Susquehanna 1 & 2	2.17E-04	6.34E-02	1.38E-05	BWR	2
91	TMI-1	4.60E-04	8.52E-02	3.92E-05	B&W	1
92–93	Turkey Point 3 & 4	2.65E-05	5.50E-02	1.46E-06	W	2
94	V.C. Summer	1.44E-03	7.80E-02	1.12E-04	W	1
95–96	Vogtle 1 & 2	4.04E-03	1.07E-01	4.34E-04	W	2
97	Waterford	2.47E-04	4.29E-02	1.06E-05	CE	1
98–99	Watts Bar 1 & 2	1.56E-03	7.57E-02	1.18E-04	W	2
100	Wolf Creek	1.18E-03	6.09E-02	7.18E-05	W	1
			Average =	**6.72E-05**	**Sum =**	**100**

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

Table A-0-2. LOOP Frequency Comparisons (Power Operation)

		Mean Frequency	95%	Mean Duration (hours)	95% Duration
1	Plant centered	2.23E-03	4.49E-03	1.6	6.2
2	Switchyard centered	1.41E-02	3.40E-02	3.2	12.3
3	Grid related	1.17E-02	5.39E-02	4.1	14.4
4	Severe weather related	5.08E-03	2.61E-02	31.9	111.9
5	Seismically induced	6.72E-05			

Table A-0-3. LOOP Frequency Comparisons (Shutdown Operation)

		Mean Frequency	95%	Mean Duration (hours)	95% Duration
1	Plant centered	4.88E-02	1.47E-01	1.6	6.2
2	Switchyard centered	7.02E-02	2.70E-01	3.2	12.3
3	Grid related	1.17E-02	2.09E-02	4.1	14.4
4	Severe weather related	3.76E-02	1.92E-01	31.9	111.9
5	Seismically induced	6.72E-05			

Note:
Source = INL/EXT-15-34443, February 2015

Attachment A – Calculations

This attachment documents the calculation details of the frequencies of seismically-induced loss of offsite power (LOOP) events given in the main body of Appendix 1.

A-1 Input-1, Seismic Event Frequencies

The seismic event frequencies for all 61 U.S. nuclear power plants are obtained from licensees' submittals as part of the effort to address Near-Term Task Force (NTTF) Recommendation 2.1 in 2014 and 2015. The submittals are available at the following NRC SharePoint site.

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

Generally, the ceramic insulators with the lowest fragilities among the structure, system, component (SSCs) modeled in the probabilistic risk assessments (PRAs) govern the occurrence of a LOOP following a seismic event. The generic fragility data for ceramic insulators is taken from NUREG-6544, "A Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences," as shown in Table A-4. The mean failure probabilities at different g level earthquakes are calculated by using the equation:

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

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, and
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 the plant-specific Individual Plant Examination of External Events (IPEEE).

Table A-4 contains the four types of SSC seismic fragilities used for calculating the conditional LOOP probabilities, given the occurrence of a seismic event at a certain g level. An example of these probabilities for a plant is given in the column named LOOP Probability in Table A-5.

Table A-0-4. Fragilities of SSCs Causing Seismically Induced LOOP

SSC	Median Capacity	β_r	β_u	HCLPF	Notes
Generic Ceramic Insulators	0.30	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 IPEEE
Offsite Power	1.40	0.22	0.20	0.7	Diablo Canyon IPEEE
Ceramic Insulators	0.30	0.3	0.45	0.1	Palo Verde

Table A-0-5. Clinton SI LOOP Calculation Using NTTF 2.1 Data

	PGA[g]	Mean Frequency per year	LOOP Probability at g	SE g Interval (begin)	SE g Interval (end)	Interval IE_{freq}	Interval Conditional LOOP Probability	Weighted Average
	a	**H(a)**						
1	0.0005	9.59E-02	1.40E-32					
2	0.001	8.14E-02	2.64E-26					
3	0.005	3.38E-02	1.86E-14					
4	0.01	1.87E-02	1.60E-10					
5	0.015	1.24E-02	1.52E-08					
6	0.03	5.15E-03	1.03E-05					
7	0.05	2.20E-03	4.62E-04	0.050	0.075	1.21E-03	1.55E-03	1.87E-06
8	0.075	9.94E-04	5.18E-03	0.075	0.10	4.54E-04	1.05E-02	4.75E-06
9	0.1	5.40E-04	2.11E-02	0.100	0.15	3.24E-04	4.59E-02	1.49E-05
10	0.15	2.16E-04	1.00E-01	0.150	0.30	1.75E-04	2.24E-01	3.92E-05
11	0.3	4.09E-05	5.00E-01	0.300	0.50	2.97E-05	6.43E-01	1.91E-05
12	0.5	1.12E-05	8.28E-01	0.500	0.75	7.49E-06	8.89E-01	6.66E-06
13	0.75	3.71E-06	9.55E-01	0.750	1.00	2.10E-06	9.71E-01	2.04E-06
14	1	1.61E-06	9.87E-01	1.000	1.50	1.15E-06	9.93E-01	1.14E-06
15	1.5	4.61E-07	9.99E-01	1.500	3.00	4.18E-07	9.99E-01	4.18E-07
16	3	4.30E-08	1.00E+00	3.000	5.00	3.71E-08	1.00E+00	3.71E-08
17	5	5.92E-09	1.00E+00	5.000	7.50	4.90E-09	1.00E+00	4.90E-09
18	7.5	1.02E-09	1.00E+00	7.500	10.00	7.55E-10	1.00E+00	7.55E-10
19	10	2.65E-10	1.00E+00	> 10		2.65E-10	1.00E+00	2.65E-10
				Sum =		**2.20E-03**		**9.01E-05**

Summary of Results ***Overall Seismic LOOP frequency for events with PGA >0.05g***	
Seismic Event Frequency =	2.20E-03
Seismically induced LOOP probability =	4.09E-02
Seismically induced LOOP frequency =	9.01E-05

A-3 Calculation of LOOP Frequency

Once the initiating event frequencies at different g levels and their corresponding conditional LOOP probabilities are known, the frequency of seismically-induced LOOP event can be calculated as a weighted average of frequencies at different g intervals. A sample calculation is shown in Table A-5.

A-4 Summary of Results

Table A-1 of Appendix 1 provides the summary of these information for all 61 U.S. nuclear power plants:

– Seismic initiating event frequencies

– Conditional probability of LOOP given seismic event

– Frequency of seismically-induced LOOP event

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

The seismically-induced LOOP frequency calculations for all 61 U.S. nuclear power plants are performed in a MS EXCEL workbook, which can be found with ADAMS Accession No. ML11220A195 as well as in the RASP Tool Box Web site (internal use only).