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Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants

Draft Report for Comment

P. G. Prassinos, M. K. Ravindra, and J. B. Savy

Prepared for the U.S. Nuclear Regulatory Commission under the guidance and with the concurrence of the Expert Panel on the Quantification of Seismic Margins: R. J. Budnitz (Chairman), P. J. Amico, C. A. Cornell, W. J. Hall, R. P. Kennedy, J. W. Reed, and M. Shinozuka.

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ABSTRACT

This report is the third report of the Expert Panel on the Quantification of Seismic Margins. The objective of this report is to present detailed guidelines for the performance of seismic margin reviews of nuclear power plants. The guidelines presented in this report are based on the Panel's second report entitled "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants."

It is intended that these guidelines be used in at least one trial plant review to demonstrate whether the approach and the quantification techniques are adequate. Based on lessons learned from these trial reviews, the Panel can then be more prescriptive about defining guidelines for general use.

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PREFACE

The Nuclear Regulatory Commission has requested that a technical basis be developed to resolve regulatory needs relating to Seismic Margins. This work is being conducted under the guidance and with the concurrence of the Expert Panel on the Quantification of Seismic Margins. The Panel is supported by Technical Support Personnel and a Working Group provides NRC liaison. This report is a collective effort and presents the ideas and views of the Expert Panel and the authors.

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EXECUTIVE SUMMARY

Recent studies by the nuclear industry and the NRC indicate that nuclear power plants are capable of withstanding earthquake motion substantially greater than the safe shutdown earthquake (SSE) acceleration. However, there has not been any systematic program performed by the industry or the NRC to quantify the actual seismic capacity of the plants. For several reasons, especially in light of the changing perception of the seismic hazard, there is a continuing need to determine this capacity for licensing purposes. The NRC and the industry are developing soundly based, efficient, and effective methods for identifying how much margin actually exists in important safety-related components, structures, systems, and the plant as a whole. These methods are designed to be applied when questions arise about the seismic capacity of a plant.

This report is intended to give the details of performing a seismic margin review of a nuclear power plant. The level of effort and cost associated with performing a seismic margin review are dependent on its scope and end use. The objective is to determine whether a plant can resist with high confidence a specified earthquake level greater than the SSE. To accomplish this objective, analyses are performed on components, systems, and the plant as a whole to determine whether there is a high confidence of a low probability of failure (HCLPF).

The margin review process involves both the screening of components based on their importance and seismic capacity, and the quantification of HCLPF values for components, systems, accident sequences, and the plant. Systems analysis is used to determine those plant systems and components that are important contributors to plant seismic safety and thus allow focusing of effort on components requiring a margin review. By studying previous PRA studies, the Panel found that, for PWRs, there are primarily two plant-safety functions that were identified as being the major contributors to plant seismic safety: reactor subcriticality and early emergency core-coolant injection. For PWRs, the initial candidates for margin review will include those components that make up the systems that perform and support these two functions. At the present time, for BWRs, the margin review will include all components and structures relevant to the plant safety functions.

Concurrently, knowledge of the inherent capacity of components (structures and equipment) obtained from previous Probabilistic Risk Assessments, and presented in NUREG/CR-4334 (Ref. 1), is used to screen out high capacity items. However, all important components are initially reviewed and inspected to assure that their seismic capacity can be represented by the generic information used for this screening before they are dropped from further consideration. In addition, any potential systems interactions and important plant-unique global features that are discovered are added to the list of review items.

The components remaining after the systems and fragility screenings plus the system interactions and plant-unique global features are then subjected to a margins quantification. Prior to this quantification, each remaining component is thoroughly inspected and studied, and system models are developed

to describe the possible seismic-initiated accident behavior of the plant. The quantification is then accomplished by calculating the HCLPF values for each of these components using structural/mechanical analyses based on the results of the detailed studies and inspections, and then analyzing the minimal cut sets derived from the systems analysis. The results of the quantification are the HCLPF values for each important low capacity component, important systems, accident sequences, and the plant as a whole, which provide information that can be used to make decisions about the seismic capacity of the plant in relation to the selected earthquake review level.

While these trial seismic margin review guidelines represent the best attempt at this time, the Panel feels they should be applied to trial plant reviews before being finalized. These trial plant reviews are intended to demonstrate whether this approach and the quantification techniques presented herein are adequate. Based on lessons learned, the Panel can then be more prescriptive about defining guidelines for general use.

CHAPTER 1.

INTRODUCTION

In May of 1984, The U.S. Nuclear Regulatory Commission (NRC) formed an "Expert Panel on the Quantification of Seismic Margins" to provide technical guidance and advice on the subject of seismic margins of nuclear power plants. The members of the Panel are on contract to the Lawrence Livermore National Laboratory and supported by technical personnel. The Panel is charged to work closely with the NRC staff's "Working Group on Seismic Margins" to address regulatory needs in the area of seismic margins. The Panel's function has been primarily to develop an approach for the quantification of seismic margins in nuclear power plants.

This report is the third document produced by the Panel. The first report, entitled "NRC Seismic Design Margins Program Plan" (Ref. 2) outlines a proposed NRC program in the area of seismic margins and plans how the Panel's work is to be accomplished, including schedules, tasks, and milestones.

The second Panel report, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants" (Ref. 1), presents a comprehensive review and analysis of the various methods, data, and results of available seismic risk assessments of nuclear power plants and other seismic studies. Its aim is to establish seismic margin review guidelines. This second Panel report includes insights about seismic margins, an assessment of the importance of plant functions and systems to seismic safety, an assessment of available component, structure, and equipment fragility information and its categorization, and outlines an approach to seismic margins quantification. The information given in this second Panel report provides the technical basis for the development of the Seismic Margin Review Guidelines contained in this report.

During the seismic margin study, the Panel adopted the following general definition of "seismic margin":

> A general definition of seismic margin is expressed in terms of the earthquake motion level that compromises plant safety, specifically leading to melting of the reactor core. In this context, margin needs to be defined for the whole plant. The margin concept also can be extended to any particular structure, function, system, equipment item, or component, for which "compromising safety" means sufficient loss of safety function to contribute to core melting if combined with other failures.

Also, during the course of its work, the Panel recognized that this approach to the quantification of seismic margins has the following limitations:

1. The "systems screening" part of the approach presently applies only to Pressurized Water Reactors (PWRs).

2. Design and construction errors are not covered.

3. Possible vulnerabilities in very small-bore hydraulic systems associated with sensors and pneumatic systems are not fully covered.

4. Electrical and control systems are not completely covered because unrecoverable relay chatter and breaker trip cannot be adequately treated from a systems standpoint at this time.

5. Evaluation of the effects of wear and aging on equipment functioning is not fully covered.

6. Possible adverse human responses caused by earthquake-induced stress are not covered.

With the seismic margins quantification approach and its limitations given in the second Panel report, the objective of this report is to give guidance on the implementation of seismic margin reviews. In the seismic margin context, the term "review" pertains to the entire process of screening, inspection, and analysis detailed in this report. The details of the review process include:

1. The organization and information needed to perform the review,

2. A review of the systems and fragility screening methodology and seismic margins quantification methodology,

3. The performance of plant walkdowns for screening and data gathering,

4. Guidance on performing component screening and inspections,

5. Systems analysis techniques including the development of event trees and fault trees,

6. A discussion of the uses and documentation of the results of the reviews, and

7. A discussion of the limitations of this review methodology and the relationship of these reviews to other seismic studies.

While these trial seismic margin review guidelines represent the best attempt at this time, the Panel feels they should be applied to trial plant reviews and then finalized. These trial plant reviews are intended to demonstrate whether this approach and the quantification techniques presented herein are adequate. Based on lessons learned, the Panel can then be more prescriptive about defining guidelines for general use.

Chapter 2 gives an outline of the margin review process and discusses the review ideas presented in the previous Panel documents. This is followed by a discussion, in Chapter 3, of the relationship between these seismic margin reviews and other seismic studies, in particular, seismic PRAs and the limitations in the current methodology. The main body of the report is given in Chapter 4, which contains a step-by-step procedure and explanation for performing the margin review along with the qualification requirements of the review team. Chapter 5 describes the resources needed to perform a Seismic Margin Review and Chapter 6 provides a format for documenting the results.

CHAPTER 2.

OUTLINE OF THE REVIEW PROCESS

2.1 Concept of Screening

The screening process developed in this report includes the use of a series of filters that enable the reviewer to sort the relevant elements (components and functions) into several classes.

Some of these filters were developed by a Panel of experts, as described in (Ref.1) They were designed to be conservative with regard to assessing the seismic margin of a plant. The purpose of these filters is to eliminate from the analysis the components that have seismic capacity conservatively estimated as being above the earthquake review level, and those components that are judged not to be important contributors to seismic-induced core melt (or both).

One filter sorts the components into two classes having HCLPF either higher or lower than the earthquake review level. This sorting is performed with the guidance of Chapter 5 of Ref.1, which is partially summarized in Table 2-1 of this report. Fragility values in Table 2-1 are consistent with spectra specified in Ref. 3.

A second filter, also conservatively developed, sorts the components into two different classes depending on the nature of the plant functions they serve. One class includes components associated with the functions of Group A as described in Ref. 1 (summarized in Table 2-2) and the other class contains the remaining elements. This second filter presented in the form of a table, Table 2-3, appears in Ref. 1 as Table 4-13. It is only applicable to PWRs. For BWRs, all plant safety functions are treated as belonging to Group A. It should be noted that there are many plant functions that are not important to plant safety. These functions are not considered as belonging to Group A.

These two filters are not sufficient, however, to segregate all relevant components, especially when dealing with plant-unique features or borderline components.

Thus, we have developed a multi-step screening procedure using a combination of plant walkdowns and engineering analyses to isolate all the functions and components of importance to the seismic margins review. The material in Chapters 4 and 5 of Ref. 1, (summarized here in Tables 2-2 and 2-3 is used as guidance to identify the components that fall clearly into one of the classes. A component that is screened out is a component that has been identified in the sorting either as not being associated with any Group A safety function, or has HCLPF value higher than the earthquake review level, or both. This component is eliminated from the review process and is not considered further in the analysis. The remaining elements may require a more detailed structural and mechanical analysis and/or a more detailed walkdown to finalize the screening. In the end, the screened-in components (i.e. those not eliminated in the screening process) are considered in the final systems

analysis where it is determined whether there is a high confidence that the plant has a capacity greater than the earthquake review level.

Section 2.2 describes the sorting process in a diagrammatic fashion. This description does not necessarily follow the actual chronological order in which the review will be performed.

Section 2.3 describes briefly the mechanics of the review organized chronologically into 8 steps including 2 walkdowns and the final step consisting of the quantification of the HCLPF values for the screened-in components, accident sequences, and the plant.

2.2 Screening of Elements

This section discusses the classification of components of a nuclear plant according to their relationships with Group A functions, their commonality with components at other plants, their seismic capacity, and their uniqueness.

The Venn diagram in Fig. 2-1 represents the actual state-of-nature where the components of a nuclear plant are segregated according to their true relationship to the functions of Group A and their true seismic capacity. Figs. 2-2 to 2-5 are Venn diagrams that idealize the classes obtained by using the different tools of the screening process. The process of screening, described by the series of Figs. 2-2 to 2-5, converges towards the final classification which, in theory, is the state-of-nature, as symbolized by making the final Fig. 2-5 identical to Fig. 2-1. The steps of the review process described in Figs. 2-2 to 2-5 do not necessarily proceed in the chronological order in which the actual review of the plant will be performed. They are only a means of explaining the sorting (or segregation) of the components of the plant into different classes.

It is noteworthy that a component that has been screened-out after the first walkdown of the review process will not be revisited later. However, a component screened-in at one step may or may not be screened-out in a subsequent step.

The term "component" includes pumps, pipes, tanks, and electrical gear found in a nuclear power plant. The term also includes the civil structures as well as all of the equipment of the plant. The term "generic component" is used for components that are commonly found in nuclear plants. They are the usual or customary types of components that have been investigated (see Ref.1 and Table 2-1 of this report) and for which a lower bound estimate of the HCLPF is likely to be available.

By default, a component for which little or no fragility information is available, even if it is a commonly used component, is a "non-generic" component. It will require some detailed structural and mechanical evaluation or use of test and experience data to estimate its HCLPF. These latter types of components are not to be confused with "plant-unique features" that are unusual features of the plant that may lead to common mode failures. They can be, for example, an adjacent dam upstream of the plant, an adjacent industrial facility, or an adjacent stack above the plant level.

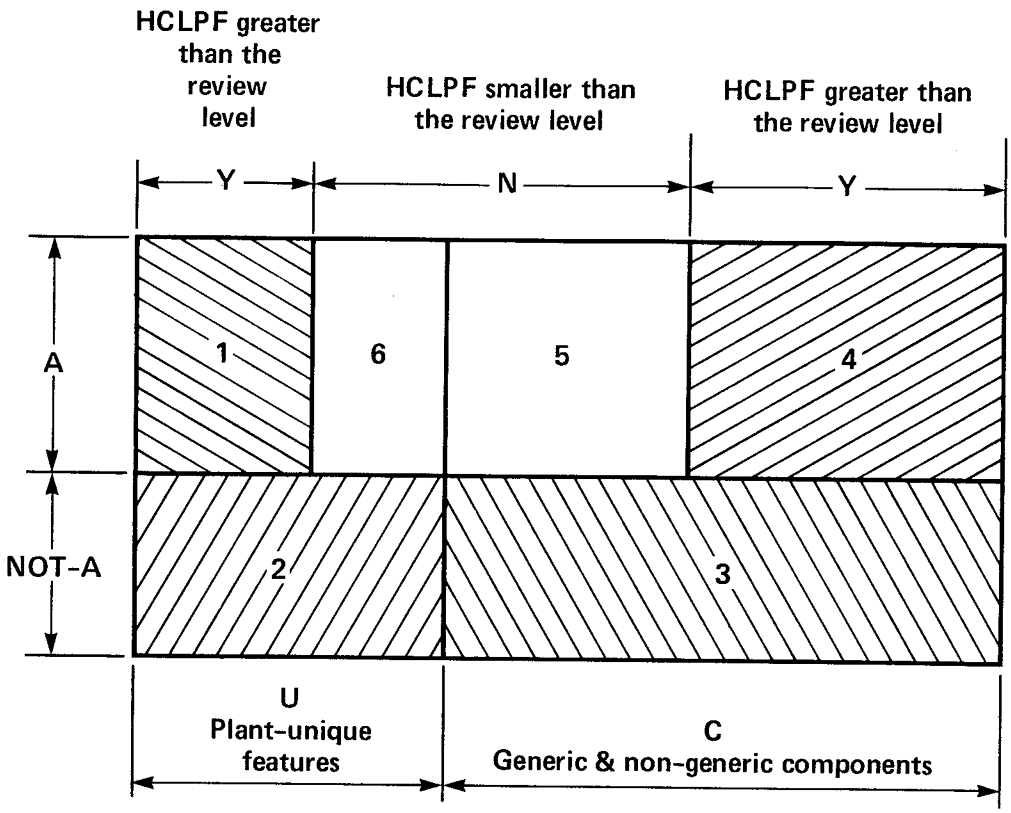


Fig. 2-1 Venn diagram representing the true State-of-Nature. The shaded areas of this Venn diagram identify the components that will be eliminated by the screening. The components of the subset 5 and 6 will be retained for the final systems analysis.

The following describes the sorting of the components, from the bulk of all the components and unique features of a plant, to the final partition into those components screened out and the components retained for the final systems analysis.

- The state-of-nature is idealized in the Venn diagram of Fig. 2-1, as described above, where the various subsets are:

  U - plant unique features
  C - generic and non-generic components
  A - components serving functions of Group A
  NOT-A - components not serving functions of Group A
  Y - components and plant unique features with HCLPF above the earthquake review level
  N - components and plant unique features with HCLPF below the earthquake review level

Thus, only subsets 5 and 6 (nonshaded area in Fig. 2-1) would be retained for the final systems analysis. All components and plant unique features not

serving the functions of have been screened out (subsets 2 and 3) and among those remaining, all those with a HCLPF above the earthquake review level (subsets 1 and 4) have been screened out (shaded area in Fig. 2-1).

In practice, the process of screening in (and out) is performed by a series of operations. These successive operations consist of applying a series of filters to the set of components. The chronological order of applying these may differ from the order shown in Figs. 2-1 to 2-5.

- The first filter represented in Fig. 2-2 partitions the complete set of components and plant-unique features into those which serve the functions of Group A (subset A) and those which do not (subset NOT-A). The components in subsets 2 and 3 are immediately screened out.

- The second filter partitions the remaining subsets according to the estimate of their minimum seismic capacities by using the guidance provided by Table 2-1 and the material in Chapter 5 of Ref. 1. This is a conservative filtering that requires verification by a walkdown. This filter also includes performing a review of the plant unique features associated with the Group A functions. Some of these unique features may be identified by the analyst from a review of the drawings and all general plant information available, but others may also be discovered at the time of the first walkdown, which is part of the next filtering operation. The outcome of this filtering is shown in Fig. 2-3, in which the Subsets 7 and 10 represent the elements screened out and Subsets 8 and 9 represent those remaining.*

- The third filtering includes a first walkdown that is designed to confirm the pre-screening performed by the second filter. In this walkdown, some components that were thought to have sufficient seismic capacities will actually appear to have a lesser capacity for a number of reasons (e.g, anchorage, or some systems interactions will be discovered, which were not identified earlier). Thus, the boundaries of the Subsets 8 and 9 will move to their new location shown in Fig. 2-4 to become Subsets 13 and 14.*

- The fourth and final filtering, which includes a second plant walkdown concentrating on components for which a further detailed structural and/or mechanical (fragility) analysis is required, finalizes the sorting by screening out the components and plant-unique features for which the HCLPF has been estimated to be above the earthquake review level. Thus, the boundaries of the subsets 13 and 14 move to become the boundaries of subsets 16 and 17 which approximate subsets 6 and 5 of Fig. 2-1.*

At this stage, the final screening has been performed and only the components

*The prime (') and double prime (") used on the Venn diagram indicate that the subsets N' and Y', N" and Y" are not necessarily the final <u>subsets</u> N and Y. As the successive filters are applied, the subsets (N',Y'), (N",Y") converge toward (N,Y).

of subsets 5 and 6 are considered in the systems analysis. An alternative graphic representation of the screening operations is given in Fig. 2-6 where each component is described by its association with the functional group and the value of its estimated HCLPF. This figure represents the classes of components screened out and the components remaining for the final systems analysis.

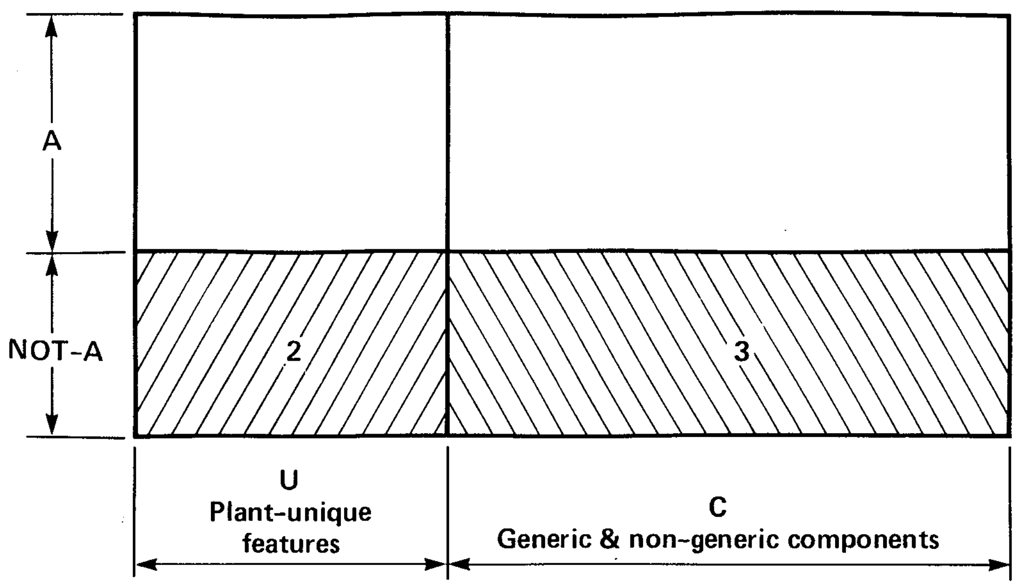


Fig. 2-2. Venn Diagram Representing the First Filtering System. Components (U & C) are separated into those serving functions of Group A (Subset A) and those not serving functions of Group A (Subset NOT-A). The Subsets 2 and 3 are screened out.

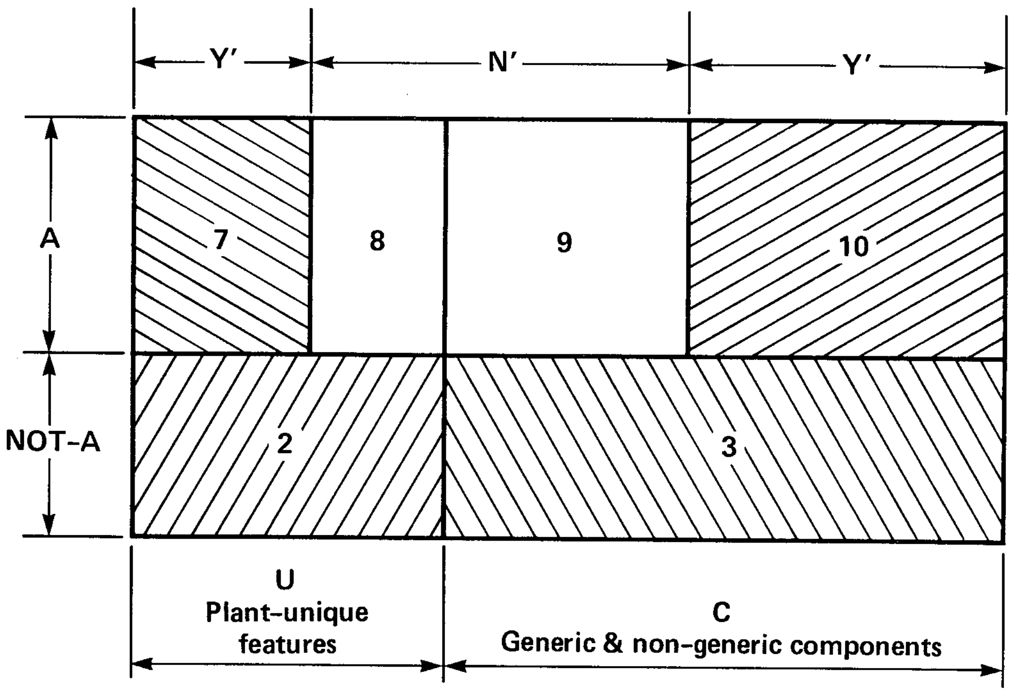


Fig. 2-3. Venn Diagram Representing the Second Filtering-Seismic Capacities. Components remaining after first filtering are partitioned according to the estimate of their minimum seismic capacities by using the guidance provided in Chapter 5 of Ref. 1 as partially summarized in Table 2-1. Some plant-unique features are identified while reviewing drawings and all the general information on the plant. The component and plant-unique features with estimated HCLPF values greater than the earthquake review level (Subsets 7 and 10) are identified by the symbols Y' and A. The components with estimated HCLPF values potentially lower than the earthquake review level (Subsets 8 and 9) are identified by the symbols N' and A).

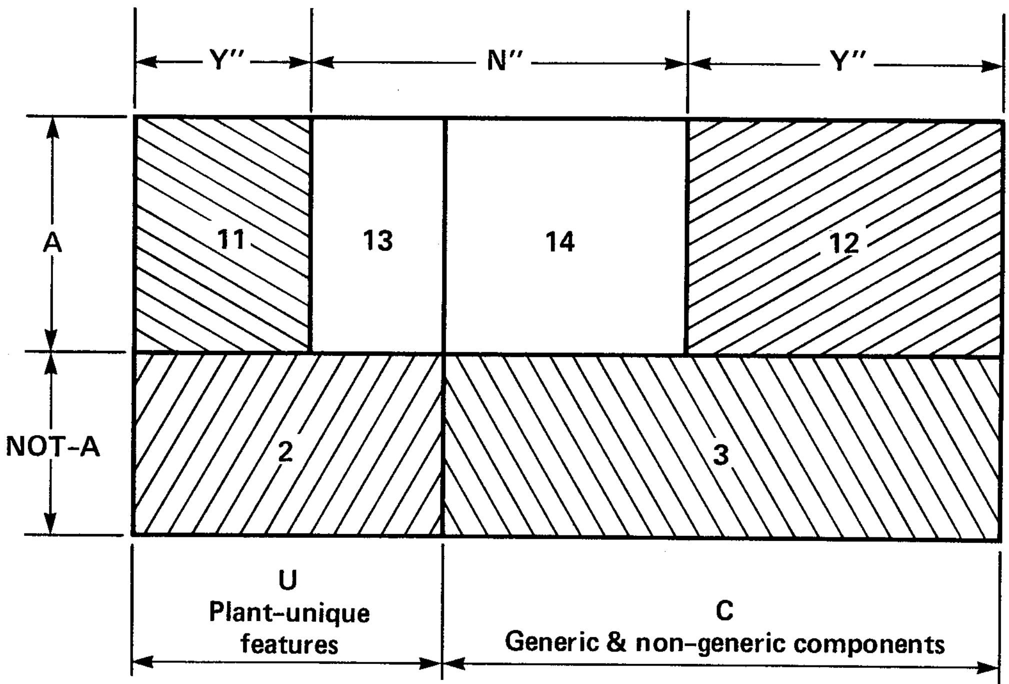


Fig. 2-4. Venn Diagram Representing the Third Filtering-First Walkdown. A first walkdown leads to adjustments in the results of the pre-screening performed by filters one and two. Some components thought to have sufficient capacities will be found to have a lower capacity, as a result of their anchorage for example, or some system interactions will be discovered which were not identified earlier. On the other hand, some components will be found to have a capacity higher than previously estimated. Thus, the partition of Subset A (which included the Subsets 7, 8, 9 and 10 after the second filtering) will change its partition into the Subsets 11, 12, 13 and 14. A, N" for components with the estimate of their HCLPF values potentially lower than the earthquake review level and A, Y" for estimated HCLPF greater than the earthquake review level.

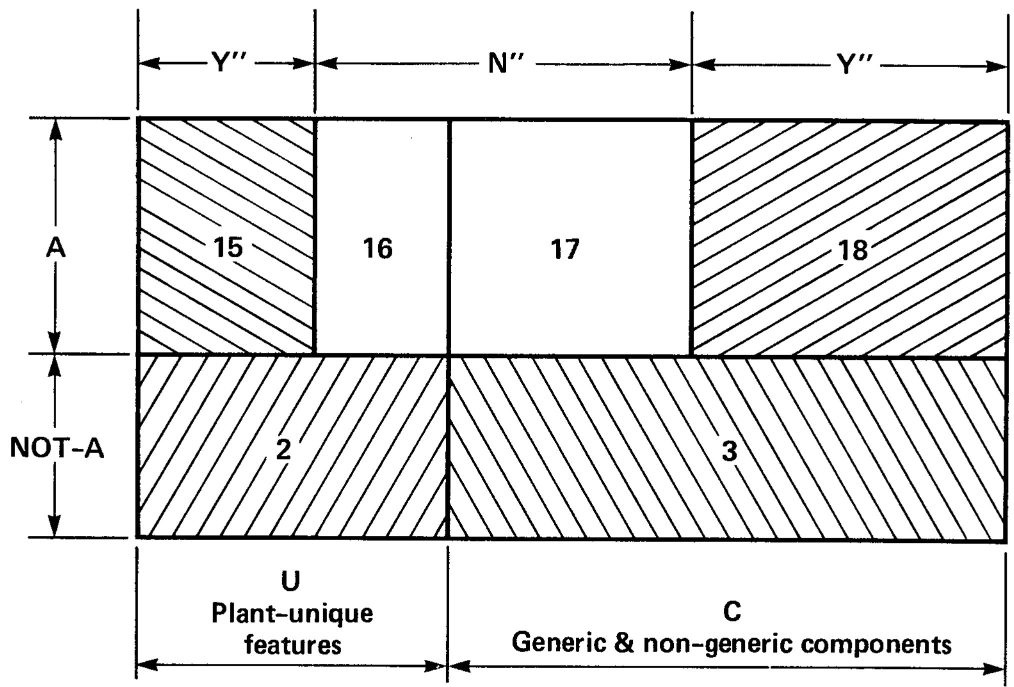


Fig. 2-5. Venn Diagram representing the Fourth Filtering-Second Walkdown. A detailed analysis, following data collection during second walkdown, will lead to further adjustments in the partition. The Subsets 11, 12, 13 and 14 become 15, 16, 17 and 18 which, if the screening process is correct, are equal to the Subsets 1, 6, 5 and 4, respectively, of Fig. 2-1 (which represents the true state-of-nature). Subsets 15 and 18 are screened-out. The final systems analysis is performed with the remaining components of Group A, which have an estimated HCLPF values lower than the review level represented by the intersection of A and N" (Subsets 16 and 17). In the event of having empty Subsets 16 and 17 (Group A components having an estimate of their HCLPF greater than the earthquake review level), no numerical capacity value would be obtained in the margin assessment except that it is greater than the review level.

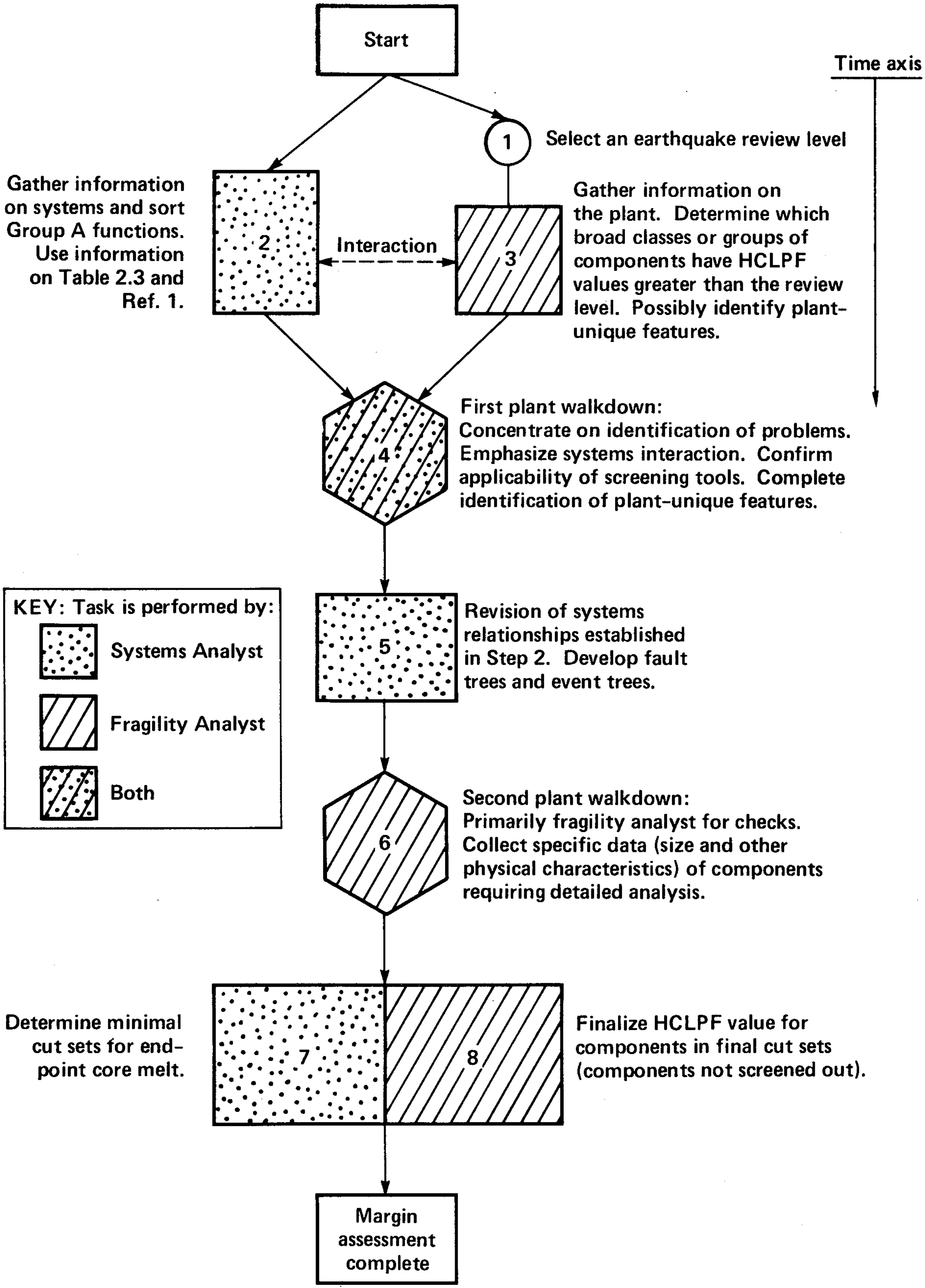


Fig. 2-6. Graphic Representation of the Screening Operations.

2.3 Schematic Representation of the Mechanics of the Review

This section provides a flow chart overview of the review scheme. Eight steps are identified that are likely to be performed in chronological order. A detailed description of these steps is given in Chapter 4 of this report.

Figure 2.7 describes the flow of tasks to be performed in eight steps. The purpose of steps 1 through 6 is two-fold. On one hand, the process screens "out" the components and systems that do not contribute to core melt at the review level (either because they do not affect the functions of Group A or because the estimate of their HCLPF is higher than the review level). On the other hand, the process provides the information necessary to build the final fault trees for systems and event trees to develop the cut-sets for the end point "core melt".

The steps appear to have different emphases. Steps 3, 6 and 8 are mostly concerned with capacity assessment and are performed by a team of fragility analysts. Step 3 relies more on judgment and experience than on engineering analysis while steps 4 and 6 rely on analysis, possibly a detailed engineering analysis. Steps 2, 5, and 7 are concerned with functions and are performed by a team of systems analysts.

2.4 Description of the Eight Steps of the Review:

Step 1 - Selection of Earthquake Review Level

It is assumed that some organization (perhaps the NRC staff, or the utility) has designated the earthquake level for which a margin review is to be performed. The review process may, however, use a different (higher) earthquake level referred to as the earthquake review level, as described in more detail in Section 4.1. The selection of the earthquake review level is the first step in the review procedure. When this "review level" has been established, the margin review will require seven subsequent steps, generally to be taken in chronological order. Note that Steps 2 and 3 must be performed before Step 4, and that Steps 5 and 6 must be performed before Steps 7 and 8, but after Step 4. Steps 2 and 3 could be performed in parallel, but most likely Step 2 would be started before Step 3. Steps 7 and 8 could be performed in either order (see Fig. 2-7).

Step 2 - Initial Systems Review

Beginning with information in Table 2-3 of this report, the systems analyst should determine which functions (Group A functions) need be considered further. These functions are screened in and the systems analyst should determine which systems are used in the plant to carry out the functions, and which components support these systems. These Group A components are carried forward to Step 4.

Step 3 - Initial Component HCLPF Categorization

The fragility analyst should gather the information about the structural, equipment, and related features of the plant needed to carry out the fragilities part of the margin review. A key part of this step is the

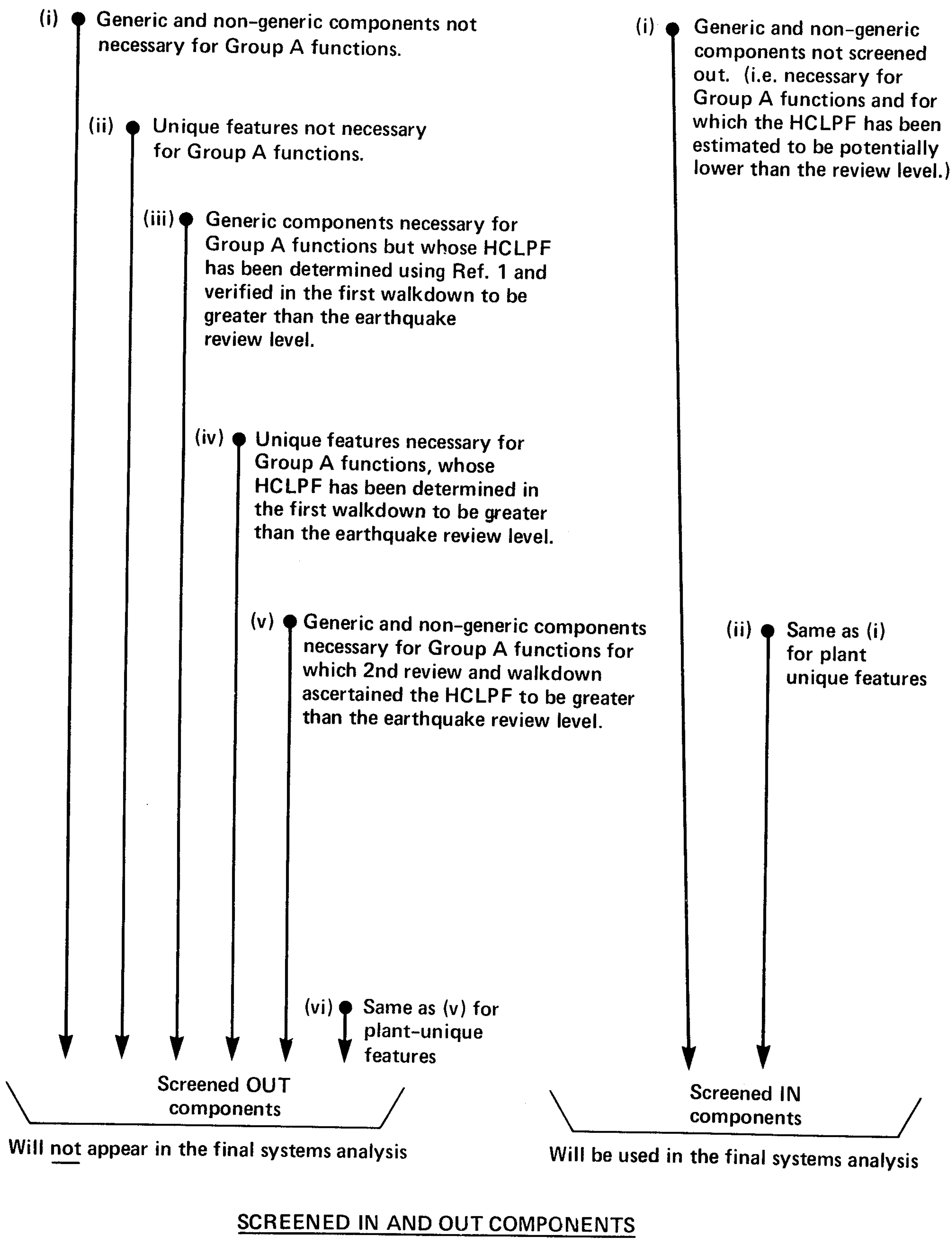


Fig. 2-7 Review Scheme.

information contained in Chapter 5, Ref. 1, and partially summarized in Table 2-1 of this Report. Using the earthquake review level the analyst determines which broad classes or groups of components have estimated HCLPF values greater than the "review level," and which have estimated HCLPF values potentially lower than the review level. These latter components, if necessary for the Group A functions will require a more detailed fragility analysis. This information is carried forward to Step 4.

Step 4 - First Plant Walkdown

The first plant walkdown is performed in Step 4. This walkdown consists of a review of the plant structures documentation (e.g., drawings, specifications, FSAR, and design analysis reports), walkdown of the plant and discussions with plant personnel. The objectives of this walkdown are the following:

1. To gain a general understanding of the plant layout and relationships of the components.

2. To verify the validity of the decisions made in Step 3 to screen out components with generically high HCLPFs.

3. To identify plant-unique features which must be considered in the margins review.

4. To gather information needed to perform more detailed reviews of components that potentially have HCLPFs less than the earthquake review level.

5. To identify systems interaction and other types of dependencies not identified during Step 2.

Step 5 - System Modeling

Taking the results from Step 4, the systems analysts should revise the systems relationships established in their earlier work in Step 2. Fault tree development for systems, event tree development to establish relationships, and other similar systems analysis work should be accomplished. The objective of this step is to provide a nearly complete set of fault and event trees that incorporate all of the Group A components and plant-unique features whose fragility values have been preliminarily assessed in Steps 3 and 4 as being of continuing concern.

Step 6 - Second Plant Walkdown

This task is the "second plant walkdown", and is primarily carried out by the fragilities analysis team, taking the results of the first walkdown and the Step 5 systems analysis into account. Prior to this walkdown some fragility analysis should have been done. It is intended that this walkdown emphasizes actual physical study of those plant components requiring detailed fragility analysis. Systems analysis input will be needed but in a supporting capacity. Although most systems interactions and similar dependencies will have been discovered in Step 4, it is likely that a review of systems interaction in this Step will be necessary to assure that other such

dependencies do not remain undiscovered. The second walkdown will primarily be a gathering of information to develop the HCLPF value of the components needing a detailed analysis (Step 8). The result of this Step will be a categorization of the remaining components by seismic capacity, that is, into those whose HCLPF value is judged to be high enough, and those for which a detailed HCLPF determination will be necessary in Step 8 below.

Step 7 - System Modeling Analysis

For those components passed through from Steps 5 and 6, the systems analyst should determine what accident sequence groups they participate in. From such determinations, detailed cut sets (Boolean expressions) should be developed for the end-point of core melt.

Step 8 - Margin Evaluation of Components and Plant

For all the components contained in the Boolean expressions developed in Step 7, the HCLPF values are developed. Special attention should be given to any singles or doubles, plus any other cut sets that appear to have HCLPF values at the lower end of the range under consideration. The HCLPF values for the combinations of components (all but "singles") may be determined using one of several methods as discussed in Chapter 4.

In practice, Steps 7 and 8 may be performed in parallel with extensive interaction between the fragility and systems analysts. These two steps will build on one another thereby involving an interaction process between the fragility and system analysis teams.

The result emerging from Step 8 is a HCLPF value for each Boolean expression leading to the end-point of a core-melt, or at least for each Boolean expression that is judged to be among the "important" ones by having HCLPF values potentially lower than the review level. If the plant's HCLPF is a greater value than the earthquake review level, the quantitative estimate of the plant HCLPF is not obtained. Rather, a statement can be made that the HCLPF value is at least as high as the earthquake review level.

The eight steps just outlined comprise the framework for the seismic margins review methodology. As can be observed, the steps are intended to screen in (or out) each component, either by itself or ultimately in combination with related components that support a system or carry out a safety function.

TABLE 2-1

SUMMARY OF CAPACITY ASSESSMENT OF NUCLEAR POWER PLANT COMPONENTS

(Taken from Ref. 1, Table 5.1)

Component	Peak Ground Acceleration Range		
	<0.3g	0.3-0.5g	>0.5g
1. Containments			
a. Prestressed and reinforced concrete	C	(1)	(2)
b. Steel	(3)	(3)	(3)
2. NSSS Supports	C	(4)	X
3. Reactor Internals	(3)	X	X
4. Control Rod Drive Mechanisms	C	X	X

(1) The capacities of major penetrations should be evaluated. HCLPF value exists for containment structure.

(2) The concrete containment structure should be evaluated for pga greater than 0.8g. Minor penetrations should be evaluated for potential leakage for pga greater than 0.5g. Capacities of major penetrations also should be evaluated.

(3) Sufficient data are not available for recommendations to be made by the Panel.

(4) Evaluation should be conducted only for BWR recirculation pumps.

C HCLPF value is adequate for this acceleration range.

X Margin evaluation for all potential failure modes is required to determine the HCLPF value. In addition, margin evaluation (or exceptions) is required as discussed in (1) through (4) above.

TABLE 2-1

SUMMARY OF CAPACITY ASSESSMENT OF NUCLEAR POWER PLANT COMPONENTS
(Continued)

Component	Peak Ground Acceleration Range		
	<0.3g	0.3-0.5g	>0.5g
5. Structure Failures			
a. Shear Walls, diaphragms and footings	C	(5)	X
b. Special non-ductile details	C	(6)	X
c. Impact between buildings	C	(7)	X
6. Block Walls	X	X	X
7. Piping	(8)	(8)	(9)X

(5) Evaluation is required only for structures that do not comply with the requirements of either the ACI 318-71 or ACI 349-76 concrete building codes or have not been designed for a SSE of 0.1g or greater.
(6) Review of construction drawings, design criteria, and simplified analyses should be conducted to determine if a margin evaluation is required.
(7) Potential for relay chatter caused by impact should be evaluated.
(8) Walkdown of example piping-system runs should be conducted. Also, piping between buildings should be inspected and margins evaluated if problems are found.
(9) Margin evaluation should include detailed walkdown of all critical piping systems.

C HCLPF value is adequate for this acceleration range.

X Margin evaluation for all potential failure modes is required to determine the HCLPF value. In addition, margin evaluation (or exceptions) is required as discussed in (5) through (9) above.

TABLE 2-1

SUMMARY OF CAPACITY ASSESSMENT OF NUCLEAR POWER PLANT COMPONENTS
(Continued)

Component	Peak Ground Acceleration Range <0.3g	0.3-0.5g	>0.5g
8. Valves	C	(10)	X
9. Heat exchangers	(11)X	(11)X	X
10. Tanks	(12)X	(12)X	X
11. Batteries and Racks	(13)	(13)	X
12. Active Electrical Equipment (capacities given for structural failure mode. Relay chatter and breaker trip failure modes must be evaluated for all earthquakes selected for screening exceeding the design basis.)	(14)	(15)X	X
13. Diesel Generators	C	(16)X	X
14 Pumps	C	(17)X	X

(10) Evaluation should be conducted only for motor-operated valves on lines 2 in. in diameter or less.
(11) Support and anchorage failure modes should be evaluated.
(12) Except for connecting piping, buried tanks do not have to be evaluated.
(13) Inspection during walkdown may indicate that battery support, racks, and anchorage are sufficiently rugged and margin evaluation is not required.
(14) Walkdown should be conducted to verify that the cabinets are securely anchored to the floor or walls, and the instruments are rigidly attached to the cabinets.
(15) Margin evaluation should focus on cabinet anchorage and attachment of instruments to cabinet.
(16) Margin evaluation should focus on anchorage and support of peripheral equipment.
(17) Margin evaluation is required only for vertical pumps with shafts unsupported at their lower ends or casings with lateral supports greater than 20 ft. apart. An exception is in cases where the shaft and casing are less than 20 ft. long. For this case, margin evaluation is not required for pga less than 0.5g.

C HCLPF value is adequate for this acceleration range.

X Margin evaluation for all potential failure modes is required to determine the HCLPF value. In addition, margin evaluation (or exceptions) is required as discussed in (10) through (17) above.

TABLE 2-1

SUMMARY OF CAPACITY ASSESSMENT OF NUCLEAR POWER PLANT COMPONENTS
(Continued)

Component	Peak Ground Acceleration Range		
	<0.3g	0.3-0.5g	>0.5g
15. Soil liquefaction	(18)X	(18)X	(18)X
16. HVAC Systems			
a. Fans & Cooler Units	(19)	(20)X	X
b. Ducting	C	(21)	(20,21)X
17. Cable Trays & Cabling	C	(22)X	X
18. Control Room Ceilings	(23)	X	X
19. Dams, Levees, and Dikes	X	X	X

(18) Margin evaluation is required only for plants judged to be susceptible to potential liquefaction.

(19) Review is required for units supported on vibration isolators to establish lateral stability.

(20) Margin evaluation should focus on anchorage systems.

(21) For ducting that spans between buildings, potential failure caused by large relative displacements should be evaluated.

(22) Margin evaluation should focus on anchor plate weld connections, taut cables, and sharp edges at ends of cable trays.

(23) Inspect for adequate bracing.

C HCLPF value is adequate for this acceleration range.

X Margin evaluation for all potential failure modes is required to determine the HCLPF value. In addition, margin evaluation (or exceptions) is required as discussed in (18) through (23) above.

TABLE 2-2

DEFINITION OF PLANT SAFETY FUNCTIONS

IDENTIFICATION OF SAFETY FUNCTIONS

1. Reactor Subcriticality - shutting down the nuclear reaction such that the only heat being generated is decay heat.
2. Normal Cooldown - providing cooling to the reactor core through the use of the normal power conversion system, normally defined as the main steam, turbine bypass, condenser, condensate, and main feedwater subsystems.
3. Emergency Core Cooling (Early) - providing cooling to the reactor core in the early (transient) phase of an event sequence by the use of one or more emergency systems designed for this purpose. The exact timing of "early" is somewhat plant specific and sequence dependent. However, for our purposes it can be deemed to be the time period during which these systems are initially called upon to operate.
4. Emergency Core Cooling (Late) - providing cooling to the reactor core in the late (stabilized) phase of an event sequence by the use of one or more emergency systems designed for this purpose. In context with the above definition of "early", for our purposes "late" can be deemed to begin with the switchover to recirculation (for LOCAs) or with the achievement of residual heat removal conditions (for transients).
5. Containment Heat Removal - removing heat from the containment to the ultimate heat sink during the late (stabilized) phase of an event sequence by the use of one or more safety systems designed for this purpose.
6. Containment Overpressure Protection (Early) - controlling the buildup of pressure in the containment caused by the evolution of steam by condensing this steam during the early phase of an event sequence by using one or more safety systems designed for this purpose. "Early" in the context of containment functions is not the same as "early" for core cooling. In this case "early" is deemed to be the time period commencing when this function is required after the beginning of core melt when these systems are operating in the injection mode.
7. Containment Overpressure Protection (Late) - controlling the buildup of pressure in the containment caused by the evolution of steam by condensing this steam during the late phase of an event sequence using one or more safety systems designed for this purpose. In the context of the previous definition, "late" in this case is deemed to start when these systems are operating in the recirculation mode.

IDENTIFICATION OF THE FUNCTIONAL GROUPS FOR PWRS AND BWRS.

PWR

Group A:	Functions 1,2,3
Group NOT-A:	Functions 4,5,6,7 + All plant functions not related to Safety

BWR

Group A:	Functions 1,2,3,4,5,6, 7
Group NOT-A:	All plant functions not related to Safety

TABLE 2-3a

FUNCTIONS REQUIRING DETAILED EVALUATIONS IN SEISMIC MARGIN REVIEWS FOR PWRS
(Taken from Ref. 1, Table 4.13)

Function	Screening Requirement	Remarks
Initiators:		F= Assume Failure
Offsite Power	F	
RCS Integrity (LOCA)	X	
Containment Integrity	X	
Functional Group A:		
Reactor Subcriticality	X	X = Margin evaluation for all potential failure modes is required.
Normal Shutdown	F	
Emerg. Core Cooling (Early)	X	
Functional Group NOT-A:		
Emerg. Core Cooling (Late)	A	A= Assume failure if core melt occurs resulting from failure of Functional Group A, assume successful if functional Group A is successful in preventing core melt. Conditional on plant walkdown not finding any extremely gross plant-specific differences.
Containment Heal Removal	A	
Cont. Overpress. Prot. (Early)	A	
Cont. Overpress. Prot. (Late)	A	

TABLE 2-3b

FUNCTIONS REQUIRING DETAILED EVALUATIONS IN SEISMIC MARGIN REVIEWS FOR BWRS

Function	Screening Requirement	Remarks
Initiators:		F= Assume Failure
Offsite Power	F	
RCS Integrity (LOCA)	X	
Containment Integrity	X	
Functional Group A:		
Reactor Subcriticality	X	X = Margin evaluation for all potential failure modes is required.
Normal Shutdown	F	
Emerg. Core Cooling (Early)	X	
Emerg. Core Cooling (Late)	X	
Containment Heal Removal	X	
Cont. Overpress. Prot. (Early)	X	
Cont. Overpress. Prot. (Late)	X	

CHAPTER 3.

RELATIONSHIP OF SEISMIC MARGIN REVIEW TO OTHER SEISMIC STUDIES

Seismic margin review methodology described in this report has similarities with and differences from other seismic studies such as seismic PRAs, deterministic and probabilistic seismic margin studies, and the Systematic Evaluation Program. A seismic margin review is also fundamentally different from the Safe Shutdown Earthquake (SSE) design or reevaluation for the SSE. In this chapter we discuss these similarities and differences with the objective of highlighting the merits and limitations of the recommended seismic margin review methodology. Finally, we describe some limitations of the seismic margin review methodology.

3.1 Comparison with Seismic PRAs

In recent years, seismic PRAs for eight nuclear power plants (Zion, Indian Point 2 & 3, Oconee, Limerick, Midland 2, Seabrook, and Millstone 3) have been published. Many others are underway. In addition, the NRC funded the Seismic Safety Margins Research Program (SSMRP, Ref. 4) at Lawrence Livermore National Laboratory (LLNL) to develop seismic risk analysis methods. The SSMRP methodology was first applied to the estimation of seismic risk of the Zion Nuclear Generating Station. Presently, LLNL is estimating the seismic risk of LaSalle County Station as part of an overall risk assessment being performed by Sandia National Laboratories under the Risk Methods Integration and Evaluation Program (RMIEP).

The key elements of a seismic PRA are:

- o Seismic hazard analysis
- o Seismic fragility evaluation
- o Plant system and accident analysis
- o Evaluation of consequences

The outputs of a seismic PRA may include:

- o Frequencies of occurrence of core melt and consequences to the public (e.g., early fatalities, potential adverse health effects, and property damage).
- o Identification of dominant seismic risk contributors - if the plant seismic risk is not acceptable, these elements may be upgraded to reduce the risk.
- o Identification of the range of earthquake peak ground accelerations that contribute significantly to plant risk. In the seismic PRAs completed to date, the uncertainties in the risk estimates have been quite large. These arise from the large uncertainties in the seismic

hazard analysis results and from the fact that the fragility evaluation relies heavily on analysts' judgment in the absence of actual fragility test data. Although the total uncertainty in the seismic risk estimates is considered by some to be larger than the uncertainties of risks from internal accident initiators, one possible explanation is that seismic PRA explicitly includes modeling uncertainties (Ref. 5).

In theory, a seismic PRA can provide complete answers regarding the seismic safety of the plant. However, a seismic PRA done for risk estimation purposes is a controversial and cumbersome way to do a seismic margins evaluation. It is recognized by the Panel that precise risk estimates are not needed to resolve most seismic margin issues. The concept of HCLPF value avoids some of the problems associated with the use of seismic PRA. The problem is narrowed down to the following: given an earthquake review level, the seismic margin review methodology assesses whether the plant can survive that earthquake with high confidence. Seismic margin reviews will establish an estimate of the minimum seismic capacity of the plant. In contrast, seismic PRAs give additional information on seismic capacity in a probabilistic format.

Both seismic PRAs and seismic margin reviews can identify dominant contributors to seismic induced risks. In the case of seismic PRA, the need to upgrade is decided on the basis of significance of contributions to the overall seismic risk. Because of the relatively shallow slope of the seismic hazard curves in the region of interest, seismic PRAs have generally indicated that extremely large increases in seismic capacities are needed to significantly reduce the seismic risk.

A fundamental difference between the seismic PRA and the seismic margin review is that the seismic margin review requires an earthquake review level to be specified. In the seismic PRAs, the random unavailability of components is treated directly. In a seismic margin review, random unavailability must be considered as described in Chapter 4. A seismic margin review gives less information on the seismic safety of a plant than a seismic PRA, a seismic margin review does provide a high confidence statement of the seismic capacity of the plant.

Table 3-1 gives a comparison of the seismic margin review methodology described in this report with the seismic PRA methodology as currently practiced.

3.2 Deterministic Code Margin Studies

Some nuclear power plants have been reanalyzed for seismic margins when the definition of seismic hazard, ground motion, or other characteristic was revised. For example, Midland Station was reevaluated for existing margins when the ground response spectra and peak ground accelerations were revised from 0.12g to 0.14g (Ref. 6).

In code margin studies, code (elastic-computed) seismic margins are computed using possibly less stringent response parameters (e.g., site-specific spectra and higher damping than RG 1.61 limits) and less stringent load combination (normal plus seismic) but assuming essentially elastic behavior and capacities

defined by governing codes such as AISC, ACI, and ASME. Code margin evaluations are typically performed for structures, systems, and components selected based on system consideration. Code margin studies are more conservative than the seismic margin review methods discussed in this report. For further details, see Ref. 7.

3.3 Probabilistic Seismic Margin Studies

In some instances the results of seismic PRAs have been studied and interpreted with respect to the capability of nuclear power plants to withstand earthquakes larger than the original design basis. The seismic margin study (Ref. 8) conducted using the results of the probabilistic safety study for Millstone Unit 3 is an example of this approach as described below:

1. Dominant contributors to seismic risk (i.e., accident sequences and critical structures and equipment) were identified in the seismic PRA.

2. For each critical structure and piece of equipment, the HCLPF value was calculated for the structure/equipment at the earthquake level (i.e., peak ground acceleration) for which there is 95 percent confidence that the probability of failure is less than 5 percent, using a lognormal fragility model. The seismic fragility information was used in quantifying the margin. These HCLPF values ranged from 0.30g to 0.88g for critical structures and equipment at Millstone Unit 3.

3. Although adequate margin for individual structures and equipment was shown to exist, it was necessary to investigate the seismic levels at which the plant safety systems or the entire plant would be severely damaged. These seismic levels may be less than the HCLPF for critical structures or equipment because the system or plant may be damaged if one or more critical structures or equipment fail, and the plant may have several components in series. The HCLPF values for dominant plant damage states were estimated using the Boolean expressions and the seismic fragilities of structures and equipment appearing in those Boolean expressions. The HCLPF for dominant plant damage states ranged from 0.26g to 0.60g. These capacities were calculated using the lognormal fragility model and assuming that the component failures are statistically independent. The component fragility curves were not truncated (i.e., "cut off") at any lower bound capacities. When the component failures were assumed to be perfectly dependent, the capacities of dominant plant damage states ranged from 0.30g to 0.62g. When the fragility curves were cut off at some selected peak acceleration values (i.e., at probabilities of 2.5% to 0.1%), the HCLPF of plant damage states did not change significantly from the previous cases studied.

4. The seismic margins of individual critical components and plant damage states have been stated in Steps 2 and 3 without reference to the seismic hazard at the site. One could say with high confidence that the plant has a low probability of sustaining damage for earthquakes up to 0.26g. The frequencies of occurrence of dominant plant damage states were calculated by convolving the plant-damage state

fragilities with the site seismic hazard. The contribution to these frequencies from earthquakes below 0.3g was estimated to be very small regardless of which family of seismic hazard curves was used.

3.4 Current SSE Design Practice

Safety-related structures, equipment, and piping and other systems in a nuclear power plant are designed to withstand the effects of a Safe Shutdown Earthquake (SSE) and an Operating Basis Earthquake (OBE). The ground response spectra for the SSE and OBE of recently designed plants are specified in Regulatory Guide 1.60. Regulatory Guide 1.61 specifies the damping ratios for different structures, piping and equipment. The NRC Standard Review Plan (Ref. 9) describes the procedures for soil-structure interaction analysis, and structural and piping response analysis. It also describes the seismic qualification methods for equipment. The load combinations and stress allowables are specified in the applicable AISC, ACI, and ASME codes and in the NRC Standard Review Plan. This design practice is intended to assure that structures and equipment respond essentially within the elastic range and that the plant can be safely shut down following the SSE.

The earthquake review level used in this report is a performance-check earthquake, it is not intended to be related to the SSE. The plant has already been designed for the SSE using conservative methods. The purpose of the seismic margin review is to estimate an earthquake level for which we have a high confidence of a low probability of failure of the plant. The HCLPF point for any component is the earthquake level at which failure is extremely unlikely. This capacity is expected to be higher than the design capacity corresponding to the SSE because of conservatisms built into the nuclear power plant design process. Conservative specification of the earthquake input (i.e., response spectra), conservative building and component response calculations, and conservative allowable stress limits all contribute to the margins that exist. As explained in Section 4.8, more realistic criteria should be used when evaluating the performance of components for the earthquake review level than were used in the (SSE) design. In design practice, all safety-related structures, systems, and components are analyzed and designed to meet the design criteria. In a seismic margin review, a screening procedure is adopted to select a subset of these components for margin evaluation. This screening is based on the importance of the components to the Group A safety functions and the seismic capacities of generic categories of components.

3.5 Systematic Evaluation Program

During the 1977-1982 period, some older operating nuclear power plants were reviewed under the NRC's Systematic Evaluation Program (SEP). These plants were designed and constructed during the period when seismic design procedures were in a state of rapid change. In many cases, these plants were designed to criteria that are less rigorous than the criteria specified today in NRC regulatory guides and the Standard Review Plan. Hence, the SEP focused on a review of these plants in light of current criteria with the objective of seismic upgrading. The approach in SEP consisted of a detailed inspection of the plant, review of existing documentation, reports, plans and calculations, and a reanalysis of critical structures and equipment using less conservative

criteria than the Standard Review Plan for ground response spectra, damping, ductility, etc. (Ref. 3).

A seismic margin review, as discussed in this report, is an extension of the Systematic Evaluation Program. A seismic margin review is expected to serve the intent of SEP updated by about 10 years of additional industry experience. Also, during these years the industry has collected data on equipment performance in actual earthquakes and qualification tests. The seismic PRAs conducted so far have also identified the important risk contributors (i.e., structures or equipment) and low capacity components so that seismic margin reviews can concentrate on them. At the time of SEP, the selection of systems and components to review was based on the NRC's safety classification. The deterministic criteria for the HCLPF calculations in seismic margin reviews are similar to those adopted in the SEP.

An important feature of a seismic margin review is the recommendation of two plant walkdowns. In the SEP, the walkdown was not as detailed as recommended in seismic margin reviews. Also, the earthquake review level contemplated in seismic margin review is higher than the levels used in the SEP.

3.6 Addressing the Limitations of Seismic Margin Methodology

For the purpose of determining seismic margins, the seismic margin methodology described in this report is suggested as an alternative to the performance of a seismic PRA. It also circumvents the difficulties of describing the seismic hazard at the site. The methodology cannot provide risk estimates, although with further development it can give risk insight. As such, the definition of acceptable seismic margin (or the earthquake review level) is subjective. Other limitations of the methodology are discussed below.

3.6.1 Ways to Verify Seismic Margin Review Methodology

a. By performing the trial reviews on one or two selected plants followed by seismic PRAs, the validity of the proposed margin evaluation methodology can be checked. In order to accomplish this validation, the fragility analysis of critical ("screened in") structures and equipment must be more detailed than currently done in seismic PRAs. The comparison would help to ascertain that the screening procedures (both components and systems) are adequate and appropriately conservative and that HCLPF values calculated for components and the plant are valid.

b. The methodology could also be verified by having the trial reviews performed on the same plant by two independent groups. Do the two groups screen out or screen in the same components and systems? Do they obtain the same HCLPF values?

3.6.2 Comparative Studies on HCLPF Evaluation Methods

There have not been enough studies done to compare the HCLPF estimated using the CDFM (Conservative Deterministic Fauilures Margin) method (Ref. 7) and the fragility-analysis method (Ref. 10) for different components. The objective of comparison studies would be to identify

situations where both methods would yield comparable results or where the results would differ widely. A review of such results may also lead to a "calibration" of the parameter values of either or both methods so that they would give essentially identical capacity estimates. This should also lead to improved HCLPF determinations.

3.6.3 Further Research Needs

a. The Panel could not make specific recommendations on BWR system screening criteria. When more seismic PRAs on BWR plants are published, the Panel recommendations will be revised to include them.

b. The effects of potential design and construction errors, relay chatter, and operator errors are not well understood at this time. Further research is needed in these areas to assess their impact on the proposed seismic margin review methodology. (See Chapter 7 of Ref.1).

TABLE 3-1

COMPARISON OF SEISMIC MARGIN REVIEWS WITH SEISMIC PRAS

AREA	SEISMIC MARGIN REVIEW	SEISMIC PRA
Purpose	Estimate Seismic Capacity Beyond SSE	Estimate seismic risk
Seismic Hazard	Excluded	Important part of PRA
Seismic Hazard Uncertainty	Excluded from Consideration	Has a major effect on results
Output	o Plant seismic capacity in terms of HCLPF - not related to specific probability distribution	o Gives seimic risk, frequency of core melt
	o Identifies lowest HCLPF elements	o Identifies lowest capacity elements
Plant Walkdown	o Two walkdowns - level of detail depends on the end use	o As currently practiced, requires a less-detailed walkdown than a margin review
Use of generic information	For screening out components	For "high" capacity components
Plant-specific Evaluation	HCLPF calculations for components; doubles, triples	Fragility curves for "low" capacity components
Safety measure decisions	Decided on HCLPF value of components and plant	Decided on frequency of core melt and/or consequences
	Estimation of HCLPF value is based on analysis, testing, experience, and judgement	Estimation of component fragility parameters (A_m β_R β_U) is more judgemental than HCLPF estimation. Large uncertainties in seismic risk estimates make decisions difficult
Earthquake input	Needs the earthquake review level	Needs seismic hazard curves
Use of random	Care should be taken to identify this in estimating the HCLPF for cut sets	The PRA methodology includes consideration of this issue

CHAPTER 4

DETAILED DESCRIPTION OF THE SEISMIC MARGIN REVIEW PROCESS

The seismic margin review process is divided into eight steps as outlined in (Ref. 1) and briefly discussed in Chapter 2. This chapter gives the details of the performance of each of these steps. The results of this review process are the HCLPF values for components, systems, accident sequences, and the overall plant that can be used to make decisions about the seismic capacity of the plant for the earthquake review level.

The next eight subsections give the details of the eight steps. Several of these steps can be performed concurrently. Section 4.9 gives the qualification requirements of the review team.

4.1 Step 1 - Selection of the Earthquake Review Level

The choice of the review level is a critical one since it is used as a basis for screening out components. The review level should be specified in terms of pga and enough spectral information to assure the applicability of the material of Chapter 5 in Ref. 1 which is partially given in Table 2-1 of this report. If the review earthquake spectral content is not consistent with the assumptions made in Ref 1, then this difference needs to be taken into account. Table 2-1 was constructed to cover most spectra generated by magnitude 6.5 earthquakes or less. In addition, spectral information will be needed to calculate HCLPF values.

If the level for which the existence of a seismic margin must be demonstrated is substantially lower than 0.3g (0.2g, for example) there is no need to perform the review for a level higher than 0.3g. Yet because the guidance given in Table 2-1 is based on only two acceleration values (0.3g and 0.5g), the analyst might choose a review level of 0.25 to 0.3g. This choice would still enable him to generically screen a maximum number of components out of the analysis.

If, on the other hand, the level for which margin must be demonstrated is about 0.3g, the analyst is presented with two alternatives:

1. The first alternative is to perform the review for 0.3g (using the first column of Table 2-1) recognizing that a substantial amount of conservatism was considered in the development of the screening tools (Tables 2-1 and 2-3). However, this alternative will lead to screening out a large number of components without requiring special detailed analysis and may lead quickly to a conclusion that there is sufficient margin with only minor additional required quantification.

2. The second alternative is to choose a review level larger than 0.3g (use the second column of Table 2-1). The drawback of this alternative will be that fewer generic components will be screened out in Steps 3 and 4. Consequently, more components will need a detailed engineering analysis to determine their actual HCLPF value. On the other hand, the analyst will probably have the advantage of being able to quantify the overall HCLPF of the plant, thereby going one step

further than merely confirming qualitatively whether there is sufficient seismic margin.

It is, therefore, clear that there exists a trade-off associated with the choice of a earthquake review level.

For the case when the level for which margin has to be demonstrated is higher than 0.3g but still below 0.5g, (Note: the screening tools presented here do not apply above 0.5g), the analyst might choose a value of 0.4 to 0.5g. Choosing a review level above 0.5g will lead to a large analysis effort by the fragility analyst since few, if any, components would be screened out and many would require detailed engineering analysis.

4.2 Step 2 - Initial Systems Review

The objective of this step is to identify those systems and components that are dominant contributors to plant seismic safety. The background information and basis for this step are given in Chapter 4 of (Ref. 1).

The methodology described in this report is intended to be applied to all nuclear power plants although the material in Ref. 1 was developed specifically for PWRs.

The general procedure for this step is as follows:

- Define the initiating events for the Group A functions found to be most important to seismic induced core melt, i.e., reactor subcriticality and early emergency core cooling.

- Review the configuration and operation of the plant. Identify the specific systems (front-line systems) that perform the Group A functions and identify the components within these systems.

- Develop systemic event trees for the defined initiating events utilizing the identified front-line systems.

- Identify support systems and components for Group A front-line systems and components.

- Identify the systems and components supporting the operation of the support systems.

- Document the systems review results.

The results of this step are a set of plant specific components that are needed to perform the important Group A functions and systemic event trees for each defined initiating event. The identified components are merged with the results of Step 3 and carried forward for further review in Step 4. Also as a result of this step, the systems analyst should gain a good understanding of the configuration and operation of the plant under review.

The front-line systems that support the functions, and their configuration, are plant specific as are the support systems and their associated

components. Guidance for determining these front-line systems, support systems and the components along with developing system models can be obtained in the IREP Procedures Guide (Ref. 11), the PRA Procedures Guide (Ref. 12), and the methods being developed in the Accident Sequence Evaluation Program (ASEP) (Ref. 13). An example of the front-line and support systems that perform the safety functions for the Millstone 3 Plant is given in Ref. 1.

The techniques and results developed by ASEP are particularly applicable to the performance of the systems analysis steps (3, 5, and 7) presented in this chapter. ASEP has categorized the safety systems configurations of the various nuclear plants and developed, for each specific configuration, simplified schematic diagrams and fault trees for both the front-line and support systems. In addition, ASEP has developed event trees for both PWRs and BWRs.

Because the ASEP information and plant models have been reviewed and are well documented, using this information as a basis for the systems analysis portion of the seismic margin review is recommended and documenting the development of the systems models can be made by reference. However, if ASEP does not provide sufficient information or the analysts prefer to develop the systems models and schematics independently, documentation of their justification, development, and analysis must be provided.

To perform this step, plant-specific information and data must be readily available for the plant under review. This information includes, for example, the Safety Analysis Report (SAR), system descriptions, plant configuration diagrams, piping and instrument diagrams (P&IDs), operating procedures, and any other supporting documentation that gives details of the configuration, actuation, operation and control of the safety systems, and their supporting systems at the plant.

4.2.1 Define Initiating Events

Starting with the information given in Table 2-3, the initiating events for the analysis must be defined in light of the two Group A plant functions that were found to be most important to seismic induced core melt. There are several initiating events that should be considered.

The transient events initiated by the loss of offsite power should be considered. These initiating events will test the reactor subcriticality function along with the ability of the plant to depressurize and provide cooling to the reactor vessel. The loss of offsite power initiating event will be representative of several transient initiators since many plant systems rely on offsite power for their operation. Other transient initiators that are not represented by the loss of an offsite power event should be considered with respect to a seismic event and the performance of the Group A functions at the plant under review. References 11 and 12 provide lists of transient initiating events that should be reviewed for their applicability.

The initiating events that result in the loss of reactor coolant system (RCS) integrity must be considered. These initiating events are the small, medium and large loss-of-coolant accidents (LOCA). The small LOCA will require the reactor subcriticality function and portions of the early ECC function. In

particular, plant cooldown, depressurization, and high pressure injection will be included. The medium LOCA may not need to be considered because its consideration is dependent on the configuration and operation of the plant safety systems. The large LOCA initiating event will not require the reactor subcriticality function but will require the early ECC function. This is because the fission process will be shut down upon core voiding and reactor depressurization will not require further action. In addition, small ECC flow may not be sufficient to provide adequate core cooling. However, this initiating event will test those systems that provide large amounts of coolant to the reactor vessel following a large LOCA. Some of the transient event sequences will result in loss of coolant from the plant caused by safety and relief valves or primary coolant pump seal failure.

The remaining initiating events that should be considered are those that result in loss of containment integrity. The loss of containment integrity of interest here is the initial loss of the containment isolation function caused by the seismic event. These initiating events should be considered for their effect on the performance of the Group A functions. Dependent on the plant under review, a containment isolation failure may render a system that performs a Group A function inoperable.

The initiating events should be grouped by their effect on the Group A function or by system response. This grouping reduces the number of event trees that represent the initiating events. Each initiating event in a group should require the same set of system actions. The groups of events can be refined further by examining the specific system responses and associated temporal considerations.

4.2.2 Identify Front-Line Systems

For PWRs the two Group A functions found to be most important to plant seismic safety as indicated in Table 2-3 are reactor subcriticality and early emergency core cooling. The plant-specific information is reviewed to determine those front-line systems that perform these two functions. For each of these systems, the analyst should determine how the system operates and interfaces with other systems, the instrumentation and control for the system, and how it is tested and maintained.

The reactor subcriticality function is performed by those systems that shut down the nuclear chain reaction within the core of the reactor. This function requires the detection and actuation of systems that provide negative reactivity to the core such as the insertion of control rods or the injection of soluable neutron poisons into the primary coolant system.

The early ECC function is intended to provide sufficient coolant to the reactor coolant system so that the core is continuously covered or to immediately recover the core if uncovery should occur as a result of a postulated accident. There are several systems that perform this early ECC function. Examples are high pressure coolant injection, low pressure coolant injection, accumulator tank injection, RCS depressurization, and auxiliary feedwater cooling.

ASEP has tabulated the front-line safety systems for the various nuclear power

plants in the United States. This tabulation can be used to indicate those Group A front-line systems that will need to be studied for any PWR under consideration.

A table of the two Group A functions and the identified front-line systems needed to perform these functions should be developed. An example of the format of this table is given in Table 4-1.

4.2.3 Event Tree Development

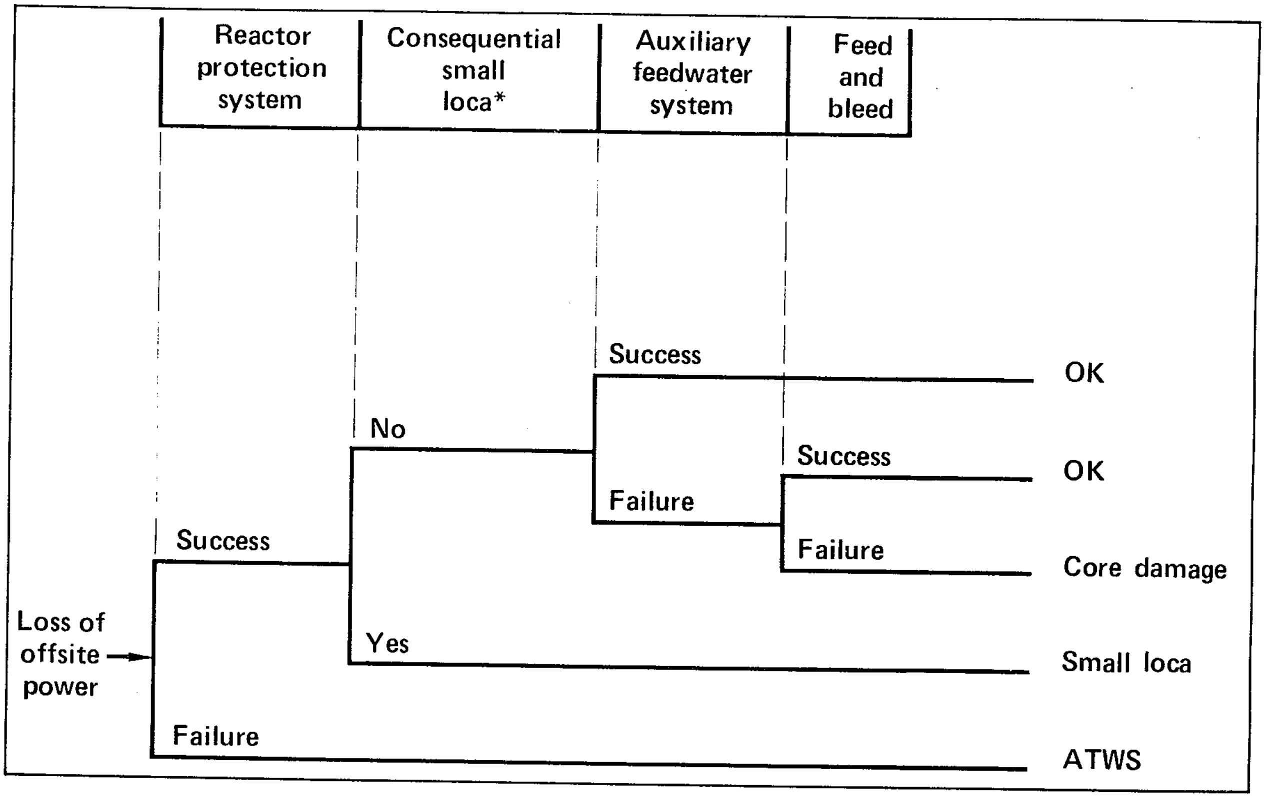
From an understanding of the plant configuration and the operation of the front-line systems gained during the performance of the earlier portions of this step, the analyst should develop systemic event trees leading to core melt for each of the defined initiating events. Each event tree should include only those identified Group A front-line systems that can provide accident mitigation for the plant conditions resulting from the particular initiating event group. In addition, for each initiating event group, the front-line systems should be approximately ordered according to when they will be called upon to respond. The dependencies among the systems should be determined. The analyst should also identify where transient-induced LOCAs transfer into the LOCA systemic event trees. An example of a systemic event tree for loss of offsite power is given in Fig. 4-1. Excellent discussions of event tree development are presented in the IREP Procedures Guide (Ref. 11) (Part II - Sections 2.2 and 2.3 and Part III - Section 2.2) and in the PRA Procedures Guide (Ref. 12) Sections 3.4.3 and 3.4.4.

ASEP has provided systemic event trees for a large range of initiating events in both PWRs and BWRs. The BWR trees can be used as presented in ASEP, since the BWR Group A functions consist of all plant safety functions. For PWRs, revisions of the ASEP event trees can be used in the analysis by eliminating those systems that do not pertain to the PWR Group A functions and truncating the trees at early core melt.

4.2.4 Identify Front-Line Components

Once the front-line systems that perform the needed safety functions are determined, the components within these systems must be identified and recorded. It would also be useful to identify the general location of these components within the plant using the plant configuration diagrams. These components will be mostly valves, pumps, tanks, piping, and electrical equipment located in various buildings at certain floor elevations in rooms, vaults or piping tunnels. Simplified schematics of the front-line systems and components should be developed. These schematics will be used in Step 4 during the first plant walkdown.

It is highly recommended that the methods discussed in Part II, Section 3.2.1 and Part III, Sections 3.1 and 3.2 of the IREP Procedures Guide (Ref. 11) be used for this task. Briefly summarized, the guide states that simplification of the plant drawings should include the omission of instrumentation, omission



*PORV/SR valve opens and fails to reclose or RCP seals fail

Fig. 4-1. Example of a Systematic Event Tree for Loss of Offsite Power.

of pipe segments that do not have a significant impact on the system performance (e.g., piping less than one-third the diameter of the main system piping), and omission of supply lines for which credit is not taken in the analysis. In addition, lines containing normally closed manual valves that could only improve system performance if opened may be conservatively omitted unless procedures specify their opening in response to accidents.

The simplified schematics should contain all piping segments and components included in the analysis. They should show the state of the components just prior to system actuation and identify components corresponding to the plant equipment labels for such components. The plant system descriptions should address these components and specify which components change state upon system actuation. The systems may also be decomposed into piping or wiring segments to facilitate the analysis. An example of a simplified drawing for a High Pressure Injection System is shown in Fig. 4-2.

ASEP provides simplified schematics of the front-line systems tabulated for each plant. These schematics were produced using the guidance just discussed and can be used directly in Step 4 without further modification or justification.

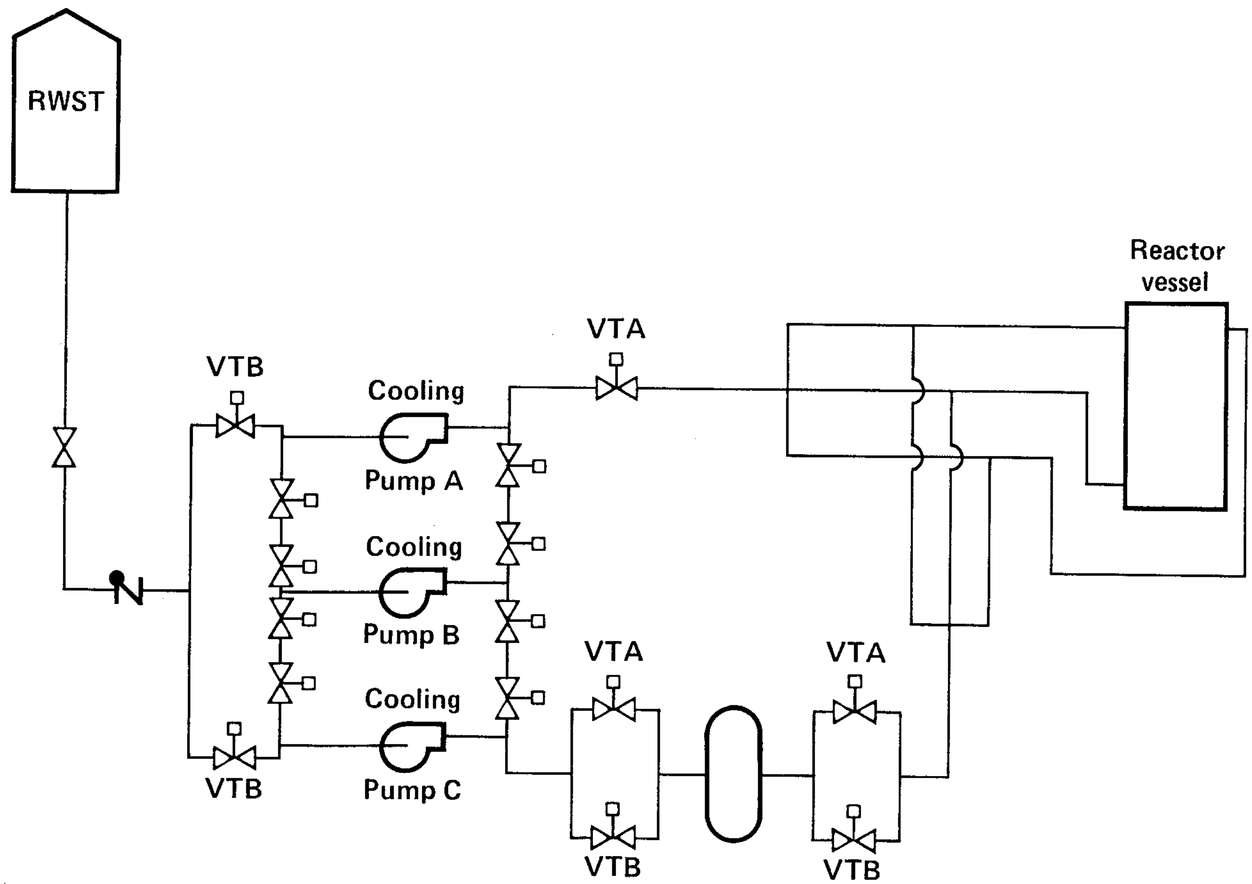


Fig. 4-2. Simplified Schematic of a High Pressure Injection System.

4.2.5 Determine Support Systems and Components

For each front-line component identified in the previous step, the systems and components that provide support for their operation must be identified. The systems that provide this support are called support systems. These support systems provide cooling, lubrication, power, fuel, actuation, and control. Pumps require cooling and lubrication for their bearings and are driven by AC or DC motors or with turbines that require pressurized steam. Valves are pneumatically, hydraulically, or electrically driven. Diesel generators require cooling, lubrication, fuel oil, intake air, and exhaust. Most components are actuated and controlled by AC or DC circuits. These circuits are made up of relays, circuit breakers, and electrical cabling. Some components are controlled by pneumatic or hydraulic actuators. While these are some of the more common supporting functions, the systems that provide these functions and their interface must be determined for each of the front-line components.

Once the necessary support systems have been determined, they must be reviewed to identify those portions of the systems (and each component) that support the front-line components. A Failure Modes and Effects Analysis (FMEA) similar to IREP (Ref. 11) (Part II - Section 1.3.1 and 3.2.1) can be performed

to determine the support-system success criteria needed for each front-line system. The support-system success criteria are expressed in terms of those portions of the systems and their operating mode needed to support a front-line system (or group of front-line systems).

There may be large portions of a support system that do not provide any needed supporting function. These portions can be eliminated from further consideration. However, care must be taken not to eliminate any portion of a system or a component that may render the entire support system incapable of performing a needed supporting function.

All support system components should be recorded and a simplified schematic of each system developed in a manner consistent with the front-line system schematics previously discussed. In addition, a matrix of the front-line systems and needed support systems should be developed. The general locations of these components also needs to be recorded.

Many support-system components also require support. Any additional support systems and components also need to be identified, recorded, and a simplified schematic developed in addition to a general description of their location in the plant. A matrix of the support systems and their supporting systems should also be developed.

Support systems and their supporting systems are identified by ASEP, which also provides simplified schematics. This information, if available, can be used in lieu of performing the above analysis to indicate which systems should be considered for a particular plant under review. In this case, no further analysis or justification would be required.

4.2.6 Documentation of Initial Systems Review

The results of Step 2 are a list of components needed to actuate, perform, control, and support the Group A functions of:

- Reactor Subcriticality and Early Emergency Core Cooling,

- Several simplified schematics of the front-line and support systems that fullfill these functions, and

- Systemic event trees.

The format for presentation of the results of this step is given in Chapter 6.

The component list should indicate each important component, its designation, a brief description of its use and operating mode, and a description of its plant location. For example, a particular pump might be described as Pump SI-2a, Safety Injection Pump for delivery of coolant to the primary system with a shut-off head of 1500 psi and a maximum flow rate of 200 gpm, located on the 219 ft. elevation of the Auxiliary Building between column lines A-B and 1-2.

During the analysis performed in this step, it will become apparent that additional information and details will be needed on particular systems and components. This information could include more details about the support

systems of a component or the operating mode of a particular safety system. All needed additional information should be recorded and, if possible, the source for this information should be identified.

The results of this step will be combined with the results of Step 3 to determine those systems and components that will be studied during Step 4, the first plant walkdown. This process is best accomplished by a joint meeting between the fragility analysts who have performed Step 3 and the systems analysts who have performed Step 2. The objective of this meeting will be to plan the first plant visit.

4.3 Step 3 - Initial Component HCLPF Categorization

The tasks in Step 3 are the following:

- Gather information on the plant and its systems and components.
- Identify the target areas and strategy for the first visit to the plant.
- Start screening out broad classes or groups of components.

This step needs to be performed by a team of analysts who have experience in the field of structural and mechanical analysis, and determination of seismic capacities. It is seen as a pre-screening step where some components will be screened out prior to the first visit to the plant, by using all the available information on the plant. The plant visit described in Step 4 will verify the status of the components identified as targets in Step 3.

The next subsection gives some guidance on the type and quantity of information needed to carry out Step 3 (subsection 4.3.1). Subsection 4.3.2 provides some guidance on how to identify the target areas for the first plant walkdown.

4.3.1 Gathering of Information

The fragility analyst needs to have the necessary information that describes the structural features of the plant including buildings, tanks, pipes etc., and all the related features which could have an impact on the review. Given the large amount of documentation available to the owners of any given plant, the analysts will have to be selective in their requests for information. The analysts can proceed in two steps to gather the necessary information.

<u>Step A:</u>

The analysts will collect all the basic information of a generic or global nature necessary to understand the overall functioning of the plant. This is described by, but is not limited to the following list of items.

List of Initial Information Request:

- Sections of the FSAR or other documents (e.g., PSAR, utility reports) that describe the seismic design criteria and the site soil conditions.

- Plant general arrangement drawings including floor elevations and equipment locations, to become familiar with the plant.

- List of drawings.

- Information on available reports.

The analysts will review the initial information to evaluate whether a reanalysis is necessary based on the design analysis and the earthquake review level. The general arrangement drawings will be reviewed for structural items that may require examination (e.g., large cutouts). If necessary, the reanalysis of some selected structures will be started at this point. Then, the team of fragility analysts will meet with the team of systems analysts to narrow down their scope of analysis and to formulate their additional needs for information as described in Step B.

Step B

In Step B, a second, more specific, information request will be made to carry out the reanalysis needs identified in Step A. This specific information is listed below but is not limited to the following items:

Second Request for Specific Information Needs:

- Selected structure and foundation drawings.
- Tank drawings and tie-down provisions.
- Soil property information.
- Seismic Qualification Review Team (SQRT) forms (if available).
- Selected equipment anchorage details.
- Selected calculations and design reports

4.3.2 Determination of Target Areas for First Plant Visit

The fragility analyst will prepare a list of components of the Group A functions, prior to the first plant visit: then develop a list of items to be reviewed carefully at the time of the visit (first walkdown). This list will include a precise description and location of the items in the field. However, this list is not exhaustive and is intended to be a minimum of what must be inspected. The analyst will use his judgment in the field in selecting other items that deserve attention. An example of a partial list is given in Tables 4-2 and 4-3.

The first visit to the plant (first walkdown) is described in Step 4. It will confirm the pre-screening performed in Steps 2 and 3.

4.4 Step 4 - First Plant Walkdown

The objectives of the first plant walkdown are:

1. To confirm that no weaknesses exist in the plant structures and equipment that would make their HCLPF lower than the generic values shown in Table 2-1, and to look for signs of abnormal aging or poor maintenance that would invalidate the use of Table 2-1.

2. To confirm the accuracy of system descriptions found in plant design documents (e.g., FSAR, general arrangement drawings, piping and instrumentation diagrams, and line diagrams for electrical equipment).

3. To identify any system interactions, system dependencies and plant unique features not already identified in Steps 2 and 3 (Section 4.4.4).

4. To gather information on certain potentially weak components for further HCLPF calculations (i.e., Group A components that do not pass initial screen - Table 2-1).

In defining these items, the system analyst should consider those components that comprise systems supporting Group A functions. He must also consider potential systems interactions, i.e., component and system failures of non-Group A functions that can lead to failure of systems performing Group A functions. He should mark these items on general arrangement drawings, piping and instrument diagrams and line diagrams, for electrical equipment. By reviewing such marked up drawings, the fragility analyst is able to identify the plant areas that require inspection.

4.4.1 Plant Walkdown

In Section 4.3.2 we have described the review to be performed by the systems and fragility analysts to become familiar with the plant systems, structures, and components and to identify areas of the plant requiring physical walkdown. The objectives of the first physical walkdown conducted by a team of systems and fragility analysts are stated above. It is expected that the systems analyst will review all areas of the plant and provide advice to the fragility analyst on the function of any component in the safety systems and the consequences of its failure.

The steps in performing the first physical walkdown are described below:

1. A pre-plant visit meeting between system and fragility analysts to plan the plant walkdown and discuss areas to concentrate on should take place.

2. Necessary arrangements must be made with the plant management regarding radiation protection, badging, and scheduling the walkdown activities to create minimal conflict with normal plant operations. The walkdown team should either include or have access to the following:

- o A reactor operator or utility engineer familiar with the plant systems,

- o An electrical technician capable of deenergizing and opening electrical cabinets for anchorage inspection.

3. An orientation meeting among the systems analyst, the fragility analyst, and the plant operating personnel should take place for exchange of general and plant-specific information.

4. The walkdown by the systems analyst is expected to:

   - o Verify P&ID correctness as to system configuration

   - o Look for potential systems interactions

   - o Verify locations of each piece of equipment

   - o Identify unusual features in the plant

   - o Advise on the significance of different failure modes of components identified by the fragility analyst.

   For example, the following failure may or may not be significant to core melt:

   - o Objects (e.g., roof slabs, adjacent unreinforced masonry walls) falling on the component or on the electrical cabinets supplying power and control to the component.

   - o Tank failures flooding the vicinity of the component.

   - o Direct seismic failure (e.g., anchorage failure).

   Certain valve failure modes may or may not be important. For example, valve stem sticking may or may not be important depending on whether the valve has to move or not during an earthquake. Identification of potential failure modes of a given component will assist in decisions on the significance of different systems interactions.

   During the plant walkdown, the systems analysts should record the above information directly on the simplified schematics that were developed in Step 2. The arrangement of the equipment being reviewed within rooms or vaults should be recorded along with the location of any other items whose failure may possibly affect the equipment or components. These items can be drawn directly on the schematic. The room or vault identification and elevation should be recorded. The location of electrical equipment, and if appropriate, its attachment location should be shown. The location of piping penetrations through walls should also be indicated. Any overhanging piping, ducting, and/or equipment, such as HVAC, should be indicated along with the existence of tanks and floor drains within the equipment rooms.

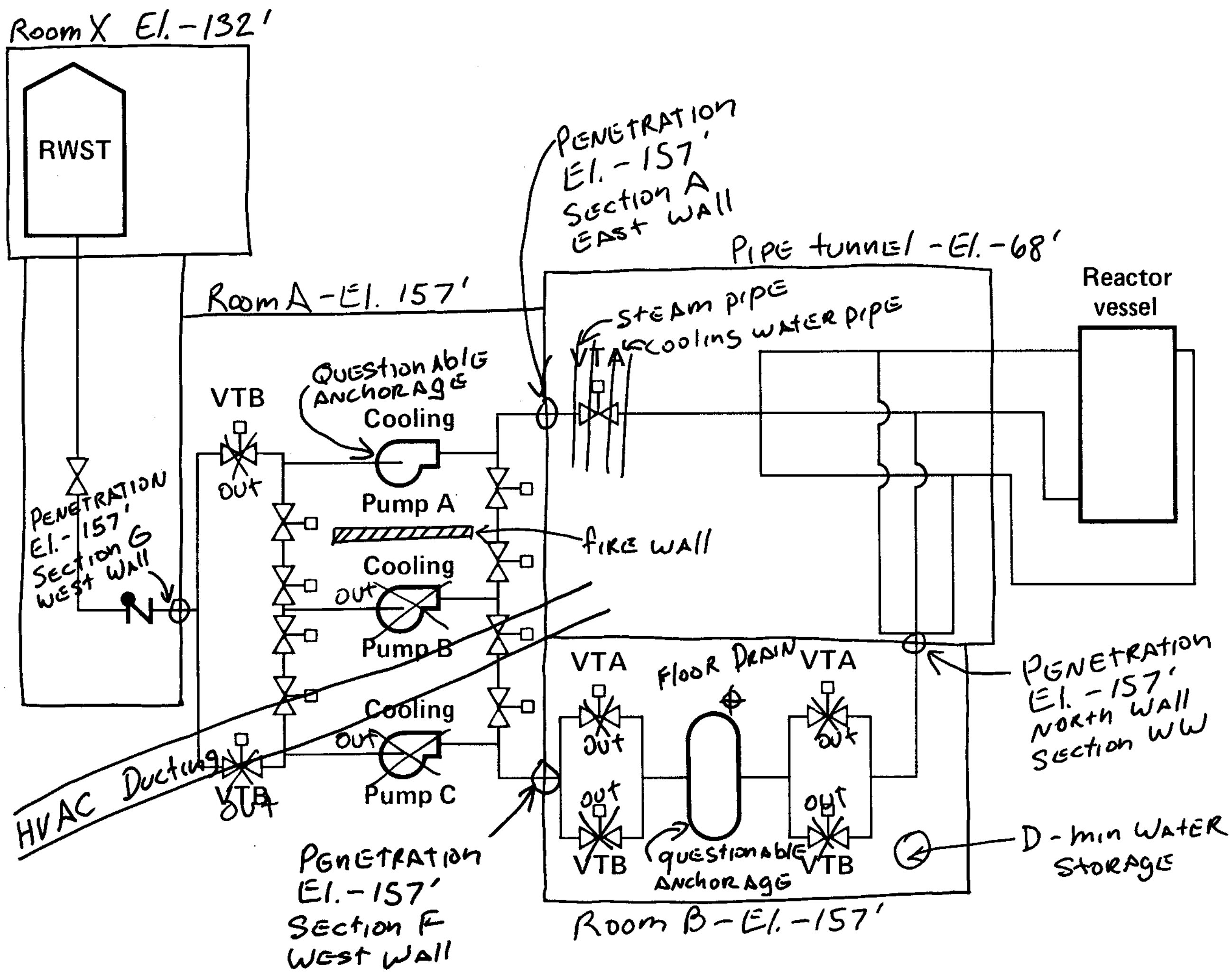


Fig. 4-3. An example of an Annotated Simplified schematic. A fragilities analyst would verify his findings with a hand-drawn schematic like this one.

While inspecting a particular piece of equipment, if the fragility analyst verifies that no weaknesses exist, this result should be noted and can also be indicated on the simplified schematic. An example of an annotated schematic resulting from the first plant walkdown is shown in Fig. 4-3.

5. One objective of the walkdown by the fragility analyst is to confirm that no weaknesses exist in the components judged to have generically high capacities and to uncover any systems interactions. In this first walkdown, the fragility analyst will concentrate on components supporting Group A functions and systems interaction that may affect them. The output of this effort will be to assign components into two categories: one category whose HCLPF is clearly higher than the earthquake review level and the other requiring further consideration.

   By a review of design documents described in Section 4.3.1, the fragility analyst will have by this stage identified components to focus on during plant walkdown. He should have become familiar with the plant arrangement, and details of equipment supports and anchorage. The items to be inspected in this first walkdown are a function of the earthquake review level. The number of items increases with the earthquake review level. Table 4-2 and 4-3 list the components to be inspected and the items to be focused on in the plant walkdown for the earthquake review levels less than 0.3g pga and between 0.3g pga and 0.5g pga, respectively. Note that the components requiring review for earthquake levels less than 0.3g pga also appear in Table 4-3 for earthquake review levels of 0.3g to 0.5g pga, but the detail of review may vary. If the earthquake review level is higher than 0.5g pga, all Group A components need to be examined in greater detail. The detail and extent of such examination is not described here.

   The fragility analyst should record these findings in a data book; the data should be supplemented by photographs where appropriate.

   In general, the survey of screened in equipment may include:

   - A photograph of the equipment to show an overall view and details of auxiliary attachments to the equipment (conduit, piping).

   - If drawings are not readily available, then general dimensions will be taken of the equipment, including the size of attachments such as piping or conduit.

   The anchorage inspection of the screened in equipment may include:

   - Notation of the type and size of the anchorage, and its arrangement around the base of the equipment.

   - Location and description of supports other than floor anchorage, such as bracing to the nearest wall.

   - If the equipment of Group A functions is anchored by bolts, the type of bolts will be determined if possible (expansion anchors or embedments). This may need to be determined through information on design documents and verified later by field inspection, if this information is significant to the results of the margin analysis.

- A demonstration that anchor bolts are properly installed. (Not needed for plants that have met the IE Bulletin 79-02 or have satisfactorily addressed the Task Action Plan A-46 issues).

- Photographs will be made of selected anchorages where possible.

6. At the conclusion of the first plant walkdown, the following objectives will have been achieved:

   - The plant system configuration is verified in order to proceed with event tree and fault tree analyses.

   - Systems interactions, other types of dependencies or plant unique features are identified.

   - Certain Group A components that are judged to generically possess high capacities (i.e., larger than the earthquake review level) have been verified to contain no weaknesses.

   - Further analyses needed to establish the capacities of remaining Group A components are identified and necessary field data are obtained.

   - Information on components is obtained to assist in HCLPF evaluation and peer review of the seismic margin study.

4.4.2 Simplified Analysis

Following the first plant walkdown, the fragility analyst may screen out certain potentially low capacity Group A components using the data collected in the design review and walkdown. This is accomplished by performing some simplified analysis.

The first step in this analysis is to estimate the response that the component experiences at the earthquake review level. As explained later (Sec. 4.8) two candidate approaches--conservative deterministic failure margin method and fragility-analysis method--are proposed to estimate the HCLPF values of components. In both methods, some reanalysis of structures and equipment may be needed to calculate the seismic responses. Such reanalysis (of structures at least) should be done before the second plant walkdown is conducted. Details of structural analyses are given in Sec. 4.8. If the structural responses (e.g., floor spectra) calculated for the plant SSE can be scaled to the responses at the earthquake review level, the effort of seismic margin studies would be considerably reduced. For this purpose, the fragility analyst should review the structural models to confirm the adequacy of the models and the appropriateness of scaling the responses. Next, he will make analyses to judge if the components' seismic capacity exceeds the earthquake review level.

The screening can be aided by using tables such as those for anchorages being developed by URS/J. A. Blume for EPRI (Ref. 14). The fragility analyst is expected to develop such rules of thumb to assist in screening of

components. Such screening tables can also be used in the plant walkdown to minimize inspection time by focusing the data collection (e.g., dimensions, number and size of bolts, etc.) efforts. A peer review team (see Sec. 5.2) is expected to critically review these screening tables and approve their use.

4.4.3 Walkdown Documentation

The systems analysts should document the following:

- o The accuracy of the P&ID's and the systems descriptions for the various Group A front-line systems and their support systems.

- o Any systems interactions or unique-plant global features uncovered during the walkdown.

In addition, the systems analyst should have a complete set of annotated schematics of the Group A, front-line, and support systems along with any additional documents, analyses, and results of discussions with plant personnel on the configuration and operation of the plant under review.

The fragility analyst should document the following:

- o For each class of components judged to possess high HCLPF values (larger than the earthquake review level),

  - state that anchorage and supports are adequate and similar to the generic component described in Chapter 5 of (Ref. 1).

  - provide example photographs showing the overall view and details of support and anchorage systems of components screened out.

  - state that no weak spots were observed in the local (supporting) structures that may make the component vulnerable to earthquakes

  - state that there are no seismically weak objects (e.g., block walls) near the equipment, and that there is no potential for flooding caused by tank failure or fire caused by obvious electrical problems.

- o Any observed systems interactions and their possible effects on the functioning of Group A components.

- o Information needs for components requiring detailed review.

4.4.4 Plant-Unique Features

Plant-unique features are systems and components of a specific plant that are not typically found in other nuclear power plants of the same reactor vendor type and design vintage. The Panel's recommendations (Ref.1) suggest that the plant review should identify plant-unique features that may potentially reduce the seismic margin of the plant. Examples of plant-unique features observed in past seismic PRA studies are the failure of Jocassee Dam at Oconee, impact between buildings at Indian Point 2, and a potential stack failure at Turkey

Point. In the plant review and walkdown, the analysts should consider such plant-unique features.

4.5 Step 5 - Systems Modeling

The objective of this step is to develop the systems models for the plant. The tasks of Step 5 are the following:

- o Review the event trees developed in Step 2 and revise, if necessary.
- o Develop fault trees for the front-line systems used in the event trees and for their support systems.
- o Document the systems modeling results.

The event trees developed in Step 2 are reviewed and revised, if needed, based on the increased understanding of the operation of the plant gained in the performance of Step 4, the first plant walkdown. The fault trees are developed for the front-line systems that appear in event trees incorporating those Group A components whose fragility values have been assessed in Step 4 as being of continuing concern and including any discovered system interactions and plant-unique global features. Fault trees for the support systems are developed along with those for the front-line systems.

4.5.1 Finalize Event Trees

Using the increased understanding of the plant configuration and operation and any additional information gained from the performance of Step 4, the event tree models developed in Step 2 should be verified and finalized. This requires the examination of each event tree to verify that at each branch point the operational characteristics of the associated system are consistent with the initiating event and the accident behavior of the plant. For example, it may have been discovered during Step 4 that a certain plant system is not called upon to respond to a particular initiating event.

The initial ordering of the systems in each event tree should be reviewed. There is often a given ordering of systems that will minimize the number of potential outcomes. This is particularly true if the same system is involved with differing success criteria in combination with other systems to perform a given function. For example, it may have been discovered that the operational characteristics of a certain system will render it incapable of performing any mitigative function given the plant conditions expected to occur as a result of the preceding sequence of system success/failure choices.

Once the event trees are finalized based on their review and any revision, the system failure definitions and modeling conditions particular to each initiating event must be developed. Each system fault tree developed in the next task must be constructed with the assumptions and criteria used to develop the event tree in which that system is used.

4.5.2 Fault Tree Development

A fault tree model should be developed for each of the front-line systems used in the event trees and their support systems The front-line and support systems were identified in Step 2 and their applicability to the event trees was verified in the first task of this step. A detailed technical discussion of fault tree modeling can be obtained from the "Fault Tree Handbook" (Ref. 15).

The development of fault trees will be facilitated by the use of piping and wiring segments. The simplified schematics developed in Step 2 should be decomposed into piping or wiring segments by placing a node on the drawings at each point where two or more wires or pipes intersect. Each portion of the system between nodes is a segment.

The development of fault trees using the system segment approach is presented in "Modular Fault Tree Analysis Procedures Guide" (Ref.16). Individuals only slightly familiar with fault tree construction will find this document to be an exceptionally useful aid to the development of these trees.

The front-line fault trees are developed with consideration of the branch point conditions in the event tree which they represent. The top-level logic of the fault trees should be constructed in terms of the piping or wiring segments. The basic events of the front-line system fault trees should include the failure modes of each front-line segment that is of continuing concern and the segments that are intersected with these segments. The failure modes for components in these remaining segments were determined in Step 4.

When determining the failure modes for each segment, the following failure causes should be considered:

- o Direct seismic failure of the segment,
- o Failure of the segment's support system(s),
- o Test and maintenance unavailability,
- o Human errors,
- o Segment failure as a result of a system interaction such as the shorting of station batteries due to the seismic failure of ventilation ducting located directly above the battery racks,
- o Segment failure as a result of the seismic failure of a unique global feature such as flooding due to the seismic failure of a nearby dam or levee,
- o Random failures.

The completed front-line system fault trees should be reviewed to ensure that all support system interfaces are included in the tree. The support system interfaces will be used to define the top events for the support system fault trees in the context of the front-line system fault tree. The individual

components remaining in each segment and the applicable failure modes should be separately documented on tables designed to accompany the fault trees. Care must be taken to ensure that systems interactions failure modes which affect more than one system segment (in one or more systems) be consistently identified as the same basic event on each fault tree for which it appears.

Once the support-system top events are defined, the support-system fault trees are developed in a manner analogous to the development of the front-line-system fault trees previously discussed. However, the support-system fault trees are developed to reflect only those portions of the system needed to support the front-line components. These fault trees should include all supporting systems and components and their failure modes.

ASEP provides fault trees for the front-line and support systems. These fault trees must be reviewed and revised based on the components that are remaining in the analysis and the inclusion of identified system interactions and plant-unique global features. However, by using this ASEP information, the fault-tree modeling effort will be greatly reduced. Even if the ASEP trees are not used directly (or are not available for the plant under review) it is strongly recommended that the analyst review the ASEP fault-tree development techniques and emulate it in the development of the fault trees for the plant under review. The simplified yet comprehensive nature of these trees is particularly suited to the evaluation of seismic margins within the format and scope of this review process.

Each fault tree should be developed considering those components remaining after the completion of Step 4 and those components that are directly intersected with them, the system interactions, and the plant-unique features. The component list and simplified schematic diagrams developed in Step 2 and completed in Step 4 should be used to develop the fault trees.

4.5.3 Documenting System Modeling Results

The results of this step are completed event trees for each of the initiating events defined in Step 2 and associated fault trees and component failure tables for each of the front-line and support systems that are needed to perform the Group A functions. A brief description of each of the front-line and support systems along with a discussion of the modeling assumptions and criteria should accompany these models.

The systems models will be verified for their correctness and applicability to the plant configuration and operation during Step 6, the second plant visit. During Step 6, additional information about component testing, maintenance, and human factors will be obtained to complete the fault trees. The complete set of system models will be analyzed for accident sequences and system failure in Step 7.

4.6 Step 6 - Second Plant Walkdown

This step is difficult to separate from the activities of Step 4. After the first plant walkdown, structural analysis is performed either by doing a full reanalysis or by scaling of design calculated responses (forces, moments, spectra, etc.). Some equipment response analyses might also have been

conducted before this plant visit. Certain Group A components suspected to have generically low capacities might have been screened out using simplified analyses.

The second plant walkdown is primarily carried out by the fragility analyst, taking into account the results of the first walkdown, preliminary analysis and the systems analysis results obtained so far. This walkdown will emphasize actual physical study of plant components requiring detailed fragility analysis. Systems analysis input will be needed but in a supporting capacity.

The objectives of this walkdown are:

- To obtain additional specific information (i.e., dimensions, number and size of anchor bolts, support details, estimate of weight, etc.) for evaluating the HCLPF values of screened in components, and

- To verify the systems models and collect any additional needed information.

4.6.1 Plant Walkdown Procedures

By this time, the fragility analyst has a good idea of the components that require margin evaluation. This second walkdown is used to determine whether additional components may be screened out using plant-specific data. This is possible because the Panel's recommendations (Table 2-1) are conservative.

Before conducting the second physical walkdown of the plant, the fragility analyst should study the design details of each screened in component (e.g., design criteria used, details of supports and anchorage systems, qualification method) using stress reports, equipment qualification reports, and other sources. He should examine the photographs of the component taken during the first walkdown to assess the as-built conditions. He should also identify the potential failure modes of the component so that he can concentrate on those elements (e.g., anchorage) during the walkdown.

Since the event- and fault-trees are not finalized at this stage, the systems analyst can make preliminary recommendations that certain components be more thoroughly studied during the walkdown and that others may not require extensive analysis because of their function in the plant systems.

During the second plant walkdown, the fragility analyst will determine details of the components; note the type and size of the anchorage and its arrangement around the base of the equipment; determine the size of piping and overhang of the motor actuator, and estimate the actuator weight for motor-operated valves on small pipes. He will make simple analyses to judge, for example, valve stem binding or the adequacy of anchorage for the earthquake review level. He may use the screening tables developed earlier (Sec. 4.4.2) to screen out additional Group A components.

For the remaining components, detailed measurements (additional to those shown on design drawings) are taken such that their HCLPF can be estimated. In practice, this is the last chance for the analysts to inspect and obtain field information on the screened in components.

4.7 Step 7 - System Model Analysis

The objective of this step is to analyze the event trees and fault trees to determine the Boolean expression for front-line system failures, accident sequences, and the aggregation of accident sequences. Step 7 is performed concurrently with Step 8 and requires significant interaction between the systems analysts and fragility analysts. The general tasks to be performed are the following:

- Analyze the event trees to determine the accident sequences that lead to seismic induced core melt.
- Analyze the fault trees to determine the Boolean expression for each front-line system failure.
- Determine the Boolean expression for the seismic induced core melt accident sequences.
- Document the results of the system model analysis.

The system failure, accident sequence, and overall plant Boolean expression will describe the single basic events or combinations of basic events that must occur for the occurrence of the top event, system failure, or core melt. The system models analyzed in this step were developed in Step 5 and reviewed and finalized as a result of Step 6.

There are many computer codes available for the analysis of event trees and fault trees. The computer code SETS (Ref. 17) is widely used for the analysis of both event trees and fault trees. Section 6.5 of the PRA Procedures Guide (Ref. 12) describes a number of computer codes currently available for the qualitative and quantitative evaluation of system and plant logic models.

4.7.1 Accident Sequence Selection

The event trees developed in Step 2 and completed in Step 5 are analyzed to determine those accident sequences that lead to seismic induced core melt. An accident sequence is a sequence of events, all of which must occur for the occurrence of the end state of the sequence. Each event in an accident sequence is either the success or failure of the front-line system associated with the branch points in the event tree or the initiating event. It is important to include both system success and failure in the accident sequences. The occurrence of each of the system events in an accident sequence is determined by analyzing the fault tree for that system.

The accident sequences identified in the event trees must be assessed to determine those sequences that lead to seismic induced core melt. For PWRs, these sequences will involve the failure to remove core heat, whether it is decay heat or heat produced during power operation, early in the accident (during the injection phase). For BWRs, the sequences will involve a number of functional failures, both early and late in the accident progression.
Any sequence that does not result in core melt should be eliminated from further analysis. The core melt accident sequences and the system failure

equations determined in the next task will be combined to give the Boolean expression for the accident sequences that lead to seismic induced core melt.

4.7.2 System Failure Analysis

The fault trees for each of the front-line systems developed in Step 5 are finalized as a result of Step 6 and analyzed to determine the Boolean expression that describe their failure. These Boolean expression are the single basic events or combination of basic events that must occur for the occurrence of the top event, system failure. A basic event refers to the failure of components and wiring or piping segments, human actions, etc., that were used to develop the fault trees.

The system fault trees developed in Step 5 should be reviewed and revised, if necessary, based on any additional information gained during the second plant walkdown. Additional information concerning plant-unique features and systems interactions should be added to the trees. Component test and maintenance, and operator error information should be used to complete the failure causes for each of the basic events. In addition, each fault tree should be reviewed to ensure that common equipment and common faults among different systems have been given the same name. This review is of particular concern between front-line and support-system trees.

The finalized fault trees can then be "pruned". This pruning will eliminate (set equal to a zero chance of failure) "AND" gates where one of the input basic events is a high capacity component with a low random failure probability. These high capacity/low random failure probability components are those that have been reviewed and determined to have a HCLPF value above the review level and a random failure probability less than about 1E-2. This pruning process can be accomplished in two ways.

The first pruning method is to locate on the fault trees each occurrence of an "AND" gate that has a high capacity/low random failure component contributing to it. The analyst can physically eliminate the above event and its following subtree from further consideration. This method will reduce the size of the trees that will need to be analyzed.

The second pruning method is performed after the fault trees have been analyzed to give the minimal cut sets for the system failure. Any cut set that contains a high capacity/low random failure component can then be eliminated from further analysis. This elimination will include those low-capacity components contained in these cut sets that are not also contained in any other cut set that has all low-capacity components.

Whether the pruning process is performed before or after the fault trees are analyzed, the front-line and associated support-system fault trees must be combined into a single tree that models the failure of each front-line system. This combination process is accomplished by attaching the support-system fault trees to the front-line-system fault trees at the associated interface nodes. The result is a set of complete front-line fault trees, one for each event-tree heading. A drawing of each of these fault trees should be made and thoroughly checked for consistency of event names, compatibility with failure definitions, and the absence of logic loops and dangling gates.

Excellent discussions of the merging of front-line and support-system fault trees are given in Refs. 1, 11, and 17.

The completed fault tree for each front-line system is then analyzed to develop the Boolean expression for the system failure. The Fault Tree Handbook (Ref. 15) discusses fault tree analysis and the associated Boolean algebra. Section 6.5 of Ref. 12 gives a brief discussion of some of the computer codes that can perform this analysis.

The basic events that will appear in the system Boolean expression will be the failure modes of those important low-capacity components/segments that have not been screened out in the previous steps as well as retained random failures and human errors. For each front-line system, the Boolean expression can be arranged as an ordered list of cut sets or as a reduced, factored equation. A complete description of each of the basic event should accompany this list of cut sets.

The cut sets for each of the front-line systems will be quantified in Step 8 to give the HCLPF values for each of these system failures.

4.7.3 Accident Sequence and Plant Analysis

Each accident sequence will contain an initiating event and one or more system events. Each system event is the success or failure of the associated front-line system. The Boolean equation for each accident sequence is determined by the logical combination of the Boolean equations for each system success or failure that is contained in that accident sequence. The Boolean expression for each accident sequence should be divided into a list of minimal cut sets or left in reduced factored form. This will be used in Step 8 to quantify the HCLPF value for each of these sequences.

The system successes in the accident sequences are included in the analysis to eliminate minimal cut sets that are precluded by the logic associated with the system successes. Any minimal cut set of an accident sequence fault tree that causes the failure of a system defined to be in a success state should be eliminated. A system success that is independent of the system failures need not be considered. The complimentary events analysis usually requires the use of a computer code and is briefly discussed in Ref. 11.

The Boolean expression for the plant will be quantified in Step 8 to give a HCLPF value. Examples of Boolean equations for accident sequences and the plant are given in Appendix B of Ref. 1.

4.7.4 Documenting Systems Analysis

The results of the systems analysis are Boolean equations and/or cut sets for each front-line system failure and accident sequence that leads to seismic induced core melt. These will be used in Step 8 to indicate those components and failure modes for which a HCLPF value will need to be determined. The Boolean expression will be used to determine a HCLPF value for each systems failure, accident sequence, and for the plant.

The systems analyses performed to develop the Boolean expression should be fully documented. This documentation should include the analysis techniques and tools used, the method used to "prune" the fault trees and the justification, and a description of the basic events (their failure modes) that make up the Boolean expression

4.8 Step 8 - Margin Evaluation of Components and Plant

Steps 7 and 8 are performed concurrently and with close interaction between the system and fragility analysts.

The components that require margin evaluation, called the "screened in" components, have been identified during the plant review and the two plant walkdowns. Design details and actual existing conditions have been recorded (as far as practical).
The objectives of the analysis in Step 8 are:

- o To estimate the HCLPF of these components
- o To estimate the HCLPF of the plant.

For each HCLPF evaluation, two alternative approaches are presented.

4.8.1 Capacity of Components

The concept of HCLPF is similar to the traditional notion of using code-minimum strengths and code-maximum loads in structural design codes. The specification of these minimums and maximums was done by code committees using past performance data, results of analysis and research, and collective expert judgments. They implicitly or explicitly recognize the uncertainties in loads and strengths. The capacity of a component calculated using these specifications was considered to be conservatively low. The HCLPF value calculated using the procedures described in this report has similar attributes: it is conservative, and it recognizes the uncertainties based on the Panel's judgment.

There are two candidate approaches for calculating the HCLPF value of components: the Conservative Deterministic Failure Margin (CDFM) method proposed by Kennedy (Ref.7) and the fragility-analysis method. In the CDFM method, a set of deterministic rules (e.g., ground response spectra, damping, material strength, and ductility) is prescribed; the capacity of the component determined using these rules gives a HCLPF value that may be more conservative than necessary. In the fragility-analysis method, the median ground acceleration capacity A_m and the logarithmic standard deviations β_R and β_U for which there is less than a five percent probability of failure with 95 percent confidence. The randomness and uncertainty in the median capacity are assumed to be lognormally distributed. In the trial plant reviews, both these methods may require that seismic response analyses separate from the design analyses be performed. The fragility analyst must review the structural models used in the plant design to confirm the adequacy of these models and the appropriateness of scaling the responses. If scaling is not appropriate, the response analysis becomes a major effort in seismic margin reviews. In the CDFM method, values for a number of parameters (e.g., system ductility,

damping, and response spectra) need to be selected. In the fragility-analysis method median values β_R and β_U need to ber estimated by the fragility analyst. There have not been enough studies done to compare the HCLPF estimated using these two candidate methods for different components. Additional comparison studies should be conducted to identify situations where both methods would yield comparable results and those where the results would widely differ. A review of such results would also lead to a "calibration" of the parameter values of either or both methods so that the two methods give essentially identical capacity estimates. The final goal of such studies would be to provide a set of deterministic rules in the CDFM method for calculating the HCLPF of screened in components. Until such research is done, it is recommended that both the candidate methods be used to calculate the HCLPF of components in trial plant reviews. The trial plant reviews should be viewed as providing further basis and guidance on research towards calibration of the two candidate methods.

4.8.1.1 Conservative Deterministic Failure Margin (CDFM) Method

In this method a failure margin is computed using conservative material and response parameters but taking credit for conservatively defined failure capacity and inelastic energy absorption capability of structures and components. The following parameter values have been proposed (Ref. 7) and might be more conservative than necessary:

Load Combination:	Normal + Earthquake Review Level
Ground Response Spectrum:	84% Non-Exceedence Probability Site-Specific Spectrum
Damping:	Depending on the earthquake review level, the following are the conservative estimates of the median values: Structure: 7% Piping: 5% Cable trays: 15%
Structural Model:	Best-estimate - median
Soil-Structure Interaction:	Envelope expected parameter variation
Material Strength:	95% exceedance actual strength
Static Capacity Equations:	84% exceedance by test data or code equation
System Ductility: (Inelastic Energy Absorption)	Conservatively selected to be between 1.0 and 1.5. For shear wall structures, should not be less than 1.3.
Floor Spectra Generation:	Median damping value for equipment Frequency shifting of floor spectra rather than peak broadening.

For structure/equipment qualified by analysis, the response of the equipment is calculated using the above structural and equipment response parameters. Potential failure modes of the equipment are identified. The static inelastic capacities of the structure/equipment are estimated. If the capacity of the structure/equipment exceeds the calculated response for the load combination (Normal + Earthquake Review Level), it is assumed that the component has a HCLPF value exceeding the earthquake review level peak ground acceleration. For equipment qualified by test, the floor spectrum for median equipment damping is generated using the above conservative structural and/or equipment response parameters. If the floor spectral values throughout the equipment frequency range of interest are less than generic equipment ruggedness spectrum (GERS) for the equipment (Ref. 18), it is assumed that the equipment has a HCLPF exceeding the earthquake review level PGA. So far, GERS has been developed for seven classes of equipment (i.e., motor-operated valves, motor control centers, switchgear, batteries and battery racks, inverters, battery chargers, and relays). For other equipment, one should use the highest spectral value for which similar equipment has been qualified as the capacity.

The GERS will be lower than the lowest observed failure level for the equipment (i.e., the GERS is the highest level for which the equipment did not fail). For equipment mounted on floors at higher elevations in the structure, the conservatisms in floor spectra generation and the conservatisms in structural parameters (i.e., damping and system ductility) yield HCLPF values that are considerably less than the median capacities. However, for equipment on grade that do not include significant response conservatism, use of GERS or experience data may not guarantee that there is no "cliff" in the capacity beyond the value of HCLPF (i.e., the component may fail suddenly when the peak ground acceleration exceeds the HCLPF value, instead of a gradual increase in the probability of failure increases). To avoid this problem, it is recommended that the capacity determined by experience data for grade level equipment be reduced by a factor. This factor may be determined during the trial plant reviews.

By a judicious selection of the values of different parameters, the CDFM method aims to produce a conservative estimate of the component's HCLPF. However, the CDFM method is less conservative than the procedures given in the Standard Review Plan (Ref. 9). The load combination specified is more liberal compared to the SRP requirements, i.e., no OBE load combination and no LOCA + review earthquake load combination in the CDFM method. The ground response spectrum is a 84% nonexceedence probability site-specific spectrum and is expected to be less conservative than the R.G. 1.60 spectrum. Similarly, the damping values proposed for the seismic margin review are more liberal than those specified in the Standard Review Plan.

The basis for the selection of values of different parameters in the CDFM methods and how they contribute to the high confidence in the capacity that assures a low probability of failure should be studied. For example, the use of 84% nonexceedence-probability site-specific spectrum and conservative estimates of the median damping are expected to result in a computed capacity indicating a low probability of failure. The use of material strength at 95% exceedance value and 84% exceedance value for static capacity prediction equations is expected to contribute to the high confidence statement about the capacity. However, this approach cannot be used to determine the

contributions of different parameters because the seismic capacity of a component is a nonlinear function of these parameters; the impact on capacity of any value of a single parameter depends not only on the significance of the parameter on the median capacity but also on the relative variabilities (i.e., randomness and uncertainties) of all the parameters. The CDFM method discussed here may be even more conservative than necessary. Until further research on calibration is performed (discussed earlier), the degree conservatism cannot be quantified.

4.8.1.2 Fragility-Analysis Method

One method of describing the fragility of a component is to express it in terms of three parameters (Ref. 19): median capacity A_m, logarithmic standard deviations β_R, and β_U representing, respectively, randomness in the capacity and uncertainty in the median value. (Fragility Handbook, Ref. 19) Rather than estimating the median capacity as a product of an overall median safety factor times the SSE pga for the plant (where the overall safety factor is a product of a number of factors representing the conservatisms at different stages of analysis and design), the median capacity is evaluated using median structural and equipment response parameters, median material properties, and ductility factors, median static capacity predictions, and realistic structural modeling and method of analysis. If the fragility analyst is convinced that the scaling of response is appropriate, the median seismic capacity may be estimated as the product of the overall median safety factor and the SSE pga.

The median response of the structure/equipment for the earthquake review level (REL) is calculated. The median capacity of the structure/equipment is estimated as the median static capacity multiplied by the median inelastic energy absorption capacity factor. The median ground acceleration capacity of the structure/element is approximately estimated as:

$$A_m \simeq (\text{REL}) \; \frac{\text{Median Capacity} - \text{Normal Design Load Response}}{\text{Median Response caused by REL}}$$

This is valid because the normal loads have low variability and the normal design loads are conservatively selected.

In lieu of explicitly determined β_R and β_U, the HCLPF value for the structure/equipment may be conservatively estimated by assuming $\beta_R + \beta_U = 0.08$ and the lognormal model: (Ref. 10 and 12)

$$\text{HCLPF} \approx 0.25\, A_m$$

If the HCLPF value calculated as above does not exceed the earthquake review level, the analyst may revise the capacity by estimating β_R and β_U using plant-specific data and PRA methods (i.e., seismic fragilities). Another option, if this proves to be too conservative, is to revise the median-capacity estimate by performing further studies such as nonlinear, inelastic static, or time history dynamic analyses.

4.8.2 HCLPF Value of Systems and Plant

As in the case of components, there are two approaches to determine HCLPF of the plant.

4.8.2.1 Deterministic Approach

This approach is based on the assumption that the HCLPF estimated in Section 4.8.1 are lower bound values.

The systems analysis as described in Section 4.7 is expected to provide a list of dominant cut sets for core melt. These cut sets can be grouped into "singles," "doubles," "triples," etc. In general, the singles and doubles make the most significant contributions to the core melt frequency.

The HCLPF of each component is calculated as described in Section 4.8.1.1. In each "doubles" cut set, the HCLPF of the cut set is estimated as the higher of the two component HCLPF values. The HCLPF of a "triples" cut set is estimated as the highest of the three component HCLPFs. In the event that one of the components in these cut sets is suspected to be unavailable (because of random failure, testing or maintenance), it requires further evaluation. If the random failure probability of the component is less than 0.01, then the component is not considered in the HCLPF evaluation. Since the emphasis in a margin review is on estimating the seismic capacity, not seismic risk, a detailed assessment of component unavailabilities is not necessary and generic data from NURER/CR-2815 (Ref. 20) should be used.

The HCLPF of the plant against core melt is evaluated as follows:

1. HCLPF of all "singles" are calculated.

2. HCLPF of all "doubles" and "triples" cut sets are calculated using the above procedure.

3. HCLPF of the plant is the lowest of the HCLPF values of components in Steps 1 and 2.

Note that if all components have been screened out, then a numerical value of the plant HCLPF is not obtainable by this procedure.

The HCLPF of the plant may also be obtained directly by studying the Boolean expressions. For example, the plant level (i.e., core melt, (CM)) Boolean expression may be

$$CM = 1 + 2 + 3 * 4 + 5 * 6 * 7$$

where the numbers represent component failures, "+" indicates "OR" gate (union) and "*" indicates "AND" gate (intersection). Assume the HCLPF of the components are 0.3g, 0.35g, 0.35g, 0.40g, 0.25g, 0.30g, and 0.4g, respectively. Then the HCLPF of the plant (i.e., for core melt) is calculated as follows:

HCLPF of CM = MIN 0.30, 0.35, max (0.35, 0.40), max (0.25, 0.20 and 0.40)
= 0.3g

If random (non-seismic) failures remain in the equation or cut sets, they should be reduced as much as possible to a HCLPF value (for the seismic terms) times the random terms. For example, the cut set A*B*RC, where RC is a random failure and A and B have HCLPF values of 0.2 and 0.25, respectively, should be shown as 0.25g*RC.

4.8.2.2 Probabilistic Approach

In this approach, a Boolean expression for core melt is developed using the event- and fault-trees described in Section 4.7. The fault trees would include both the generically high capacity (screened out) components and the screened in components. Fragility parameters for the "high" capacity components are taken from the fragility data base. Fragility parameters for the screened in components, A_m, β_R, β_U, are estimated as described in Section 4.8.1.2.

For example, the Boolean expression for core melt could be:

$$CM = 1 + 2 + 3 + 4 + 5 * 6$$

where the numbers represent failure of the corresponding component whose fragility parameter values are given in Table 4-4. The notation "+" indicates probabilistic addition (union) and "*" indicates probabilistic multiplication (intersection). Following the rules of Boolean algebra, the individual component fragilities are combined, using a numerical procedure (e.g., the Discrete Probability Distribution approach (Ref. 21), and/or Monte Carlo simulation). Figure 4-4 shows a plot of the plant level (core melt) fragility curves in which the family of curves is reduced to the 5%, 50%, and 95% confidence fragility curves. From this plot, the HCLPF (defined as peak ground acceleration corresponding to 5% probability of failure at 95% confidence) for core melt is obtained as 0.26g. In the above analysis, the component failure events are assumed to be statistically independent (both in randomness and uncertainty). This may be a realistic assumption because the components involved in the plant level Boolean expression are generally dissimilar items of structures and equipment (e.g., buildings, tanks, electrical equipment, control rod drive system, RCS piping, and service water pumps), their locations in the plant and within the structures are found to be different, and their dynamic characteristics are also judged to be different. Hence, correlation in the seismic responses and in the seismic capacities of these components may be assumed to be minimal.

In some instances, correlation in the component failures may be expected. Assumption of perfect dependence in both uncertainty and randomness is an extreme case. Assumption of perfect dependence in the uncertainties of different component fragilities means that the median ground acceleration capacities of all components are known if the median ground acceleration capacity of one component is given. Since the uncertainty arises from insufficient understanding of structural material properties, approximate modeling of the structure, inaccuracies in the representation of mass and

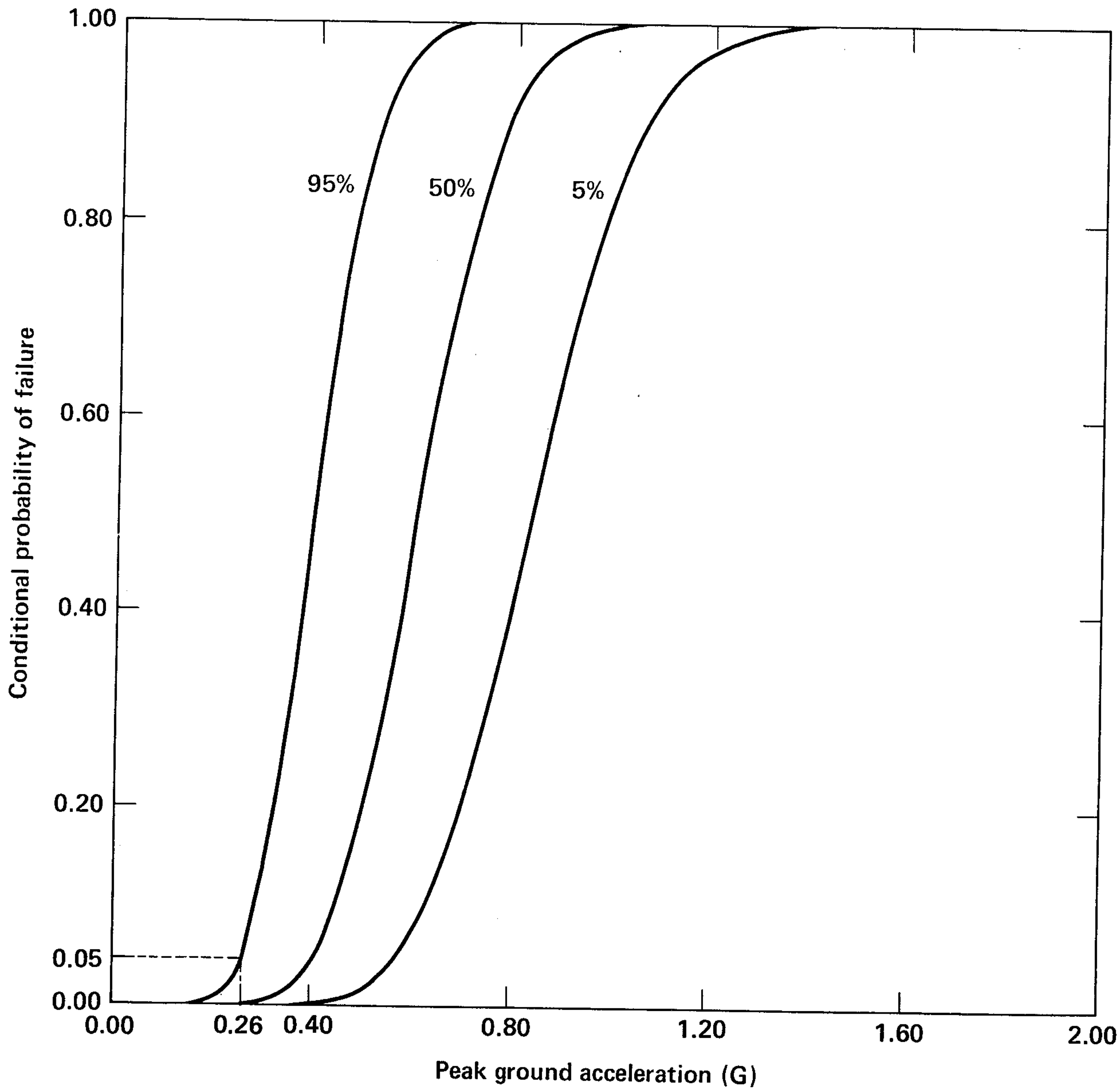


Fig. 4-4 Plant Level Fragility Curves.

stiffness, and the use of engineering judgment in lieu of plant specific data, it is expected that all the components will be affected to some degree by these uncertainties. Therefore, some probabilistic dependence between component median capacities may be expected. Perfect dependence in uncertainty is an extreme case.

Dependence in the randomness arises from a common earthquake generating the responses in different components and common structural/material properties. Assumption of dependence in the randomness means that if the fragility (probability of failure) of a component for a given peak ground acceleration is known, the probability of failure of the other components is somewhat modified by that knowledge.

The plant-level fragility and, therefore, the HCLPF, depend on the degree of dependence in randomness and uncertainty between the component fragilities. For example, when the component failures are assumed to be perfectly dependent (in both the randomness and uncertainty), the HCLPF value for core melt is calculated to be 0.30g. It is recommended that bounding cases of perfect independence and perfect dependence be studied to establish the lowest plant-level HCLPF. If partial dependence between component failures must be quantified, plant-level HCLPF could be obtained using the method in Ref. 22.

For active equipment that has a chance of being unavailable (i.e., the result of random failure or testing or maintenance outage), the probability of failure given an earthquake peak ground acceleration is given by the probability of the union of the events, i.e., earthquake induced failure of equipment and the unavailability of the equipment.

4.8.3 Final Results

The results of this margin evaluation may be summarized as:

- Some components of Group A functions have capacities generically larger than the earthquake review level (Sec. 4.4 and 4.6).

- On the basis of plant-specific information and employing the methods described in Section 4.8.1, some other components have HCLPF values larger than the earthquake review level.

- If the plant level HCLPF is calculated to be larger than the earthquake review level (determined using the methods described in Section 4.8.2.), the plant has adequate seismic margin against the earthquake review level.

- If the plant level HCLPF is found to be lower than the earthquake review level, a numerical value of the plant seismic capacity at a high confidence and low probability of failure level will be provided.

TABLE 4-1

PLANT FUNCTIONS vs FRONT LINE SYSTEMS MATRIX

	Reactor Protection System	Power Conversion System	High Pressure Injection System	Low Pressure Injection System	Accumulator System	Auxiliary Feedwater System	Recirculation System	Quench Spray System	Primary SRV System	Secondary SRV System	Reactor Coolant Pump Seal Cooler
Reactor Subcriticality	X		X								
Normal Cooldown		X									
Emergency Core Cooling (Early)			X	X	X	X			X	X	X
Emergency Core Cooling (Late)			X				X				
Containment Heat Removal							X				
Containment Overpressure Protection (Early)								X			
Containment Overpressure Protection (Late)							X				

TABLE 4-2

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES LESS THAN 0.30G PGA

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Blockwalls	Are they reinforced or laterally supported?	Walkdown to identify any potential systems interaction.
	Can arching action be developed?	
Piping (typical safety-related piping runs that are accessible)	Look for piping details which could be potential problems including:	
	Short, stiff pipe sections between equipment, and between equipment and supports (including wall penetrations).	If equipment moves (e.g., flexes or slides) piping could fail.
	Long flexible pipe runs which could swing and impact other pipes or equipment.	Valve operators could be damaged if they impact other equipment or structures.
	Supports with expansion anchor bolts. (May have to be done by reviewing construction drawings.)	
	Small pipes connected between large flexible pipes.	
	Piping between buildings.	Failure could occur due to large relative displacement caused by rocking or sliding of buildings.
	Brittle connections, socket welds, eroded or corroded piping (if it can be identified), and brittle cast iron piping.	

TABLE 4-2

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES LESS THAN 0.30G PGA
(Continued)

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Heat Exchangers	Inspect the supports (i.e., number, size of anchor bolts) and verify the details shown on design drawings.	Failure (leak or rupture) may flood and fail some other essential equipment (i.e., potential systems interaction).
Tanks	Inspect number of hold-down bolts on flat bottom tanks. Document number of bolts, size and stiffener details. Inspect how the tank is anchored to concrete (with chairs and angles or simply bolted down). For elevated tanks which are supported on slender legs and with no diagonal bracing, check the attachment of tanks to legs; obtain sizes of support members if a HCLPF evaluation is required. Study the weld details at valve pit location; is there enough flexibility in piping? Where is the valve located with respect to the tank? Look for potential for syphoning of tanks by break of connecting pipe.	Look for lack of stiffeners to transfer forces into tanks.
Batteries and racks	Inspection should focus on battery support (i.e., spacers and shims), ruggedness of racks, and anchorage details.	Some racks in older plants have low seismic margin.

TABLE 4-2

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES LESS THAN 0.30G PGA
(Continued)

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Batteries and racks (Continued)	Check for potential failure of space heaters, hanging light, or blockwall enclosures that could fall and impact the batteries.	
Active Electrical Equipment	Inspect method and adequacy of anchorage for motor control centers, switchgears, instrumentation panels, cabinets and racks and other electrical components.	Look for cabinets not tied together which could impact each other.
	Look for slack in cables and air lines at equipment attachment points.	
	Look inside electrical cabinets to determine if relays or other electrical instruments are securely mounted.	If safe to do so, push on instruments to check anchorage and flexibility.
	Inspect lateral restraints on large breakers.	Breakers could dislodge relative to cabinet.
HVAC Systems	Fan and cooler units.	
	Check if the units are supported on vibration isolators. If so, needs a margin review.	
	Examine if there is a lateral stop.	

TABLE 4-2

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES LESS THAN 0.30G PGA
(Continued)

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Control Room Ceilings	Inspect for adequate bracing	
	Check area above lightweight control room ceiling units for heavy reflective panels or equipment (e.g., air conditioning equipment).	Reflective panels may fall through ceiling and damage control instruments or injure operators.
Dams, Levees and Dikes	If they exist and their potential failure may have significant consequence to plant safety, they should be inspected to assess their existing condition and vulnerability to earthquakes.	

TABLE 4-3

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES BETWEEN 0.3 AND 0.5G PGA

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Containment	Review major penetrations.	
NSSS Supports	Review the support details. Check if any tests were done to verify that snubbers can lock up.	
Structural Failures	a. Shear walls, diaphragms footings:	Needed only if ACI 318-71 or ACI 349-76 requirements are not met or if the plant was designed for less that 0.1g SSE.
	Inspect the connection details between panels and columns.	
	b. Special non-ductile details:	
	Check if large openings exist, in shear walls or in floor (roof) diaphragms.	
	Inspect the connection details between panels and columns.	
	c. Impact between buildings:	
	Check if the gap between buildings is as large as shown on drawings.	
	Inspect size of expansion and seismic joints between structures.	Confirm that joint size complies with plans and specs.

TABLE 4-3

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES BETWEEN 0.3 AND 0.5G PGA
(Continued)

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Blockwalls	Are they reinforced or laterally supported?	Walkdown to identify any potential systems interaction.
	Are they near any safety-related equipment? Can arching action be developed?	
Piping (all accessible runs in selected safety-related piping systems)	Look for piping details which could be potential problems, including:	
	Short, stiff pipe sections between equipment, and between equipment and supports (including wall penetrations).	If equipment moves (e.g., flexes or slides) piping could fail
	Long flexible pipe runs which could swing and impact other pipes or equipment.	Valve operators could be damaged if they impact other equipment or structures.
	Supports with expansion anchor bolts (may have to be done by reviewing construction drawings).	
	Small pipes connected between large flexible pipes.	
	Piping between buildings.	Failure could occur because of large relative displacement caused by rocking or sliding of buildings.

TABLE 4-3

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES BETWEEN 0.3 AND 0.5G PGA
(Continued)

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Piping (Cont.)	Brittle connections, eroded or corroded piping (if it can be identified), and brittle cast iron piping.	
Valves	Inspect representative motor-operated valves on small pipes (less than 2" in diameter).	Inspect support and/or yoke size of operators.
		Check if the yoke is made of cast iron.
	Look for cases where the operator is anchored to the structure but the valve or piping immediately adjacent to the valve is not anchored.	The concern is that large piping displacement may severely strain the yoke between the valve and operator and cause binding of the operator stem or leakage past the stem seals.
Heat Exchangers	Review details on the supports (i.e., number, size of anchor bolts)	Failure (leak or rupture) may flood and fail some other essential equipment (i.e., potential systems interaction).
Tanks	Inspect number of hold-down bolts on flat bottom tanks. Document number of bolts, size, and stiffener details.	Look for lack of stiffeners to transfer forces into tanks.

TABLE 4-3

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES BETWEEN 0.3 AND 0.5G PGA

(Continued)

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Tanks (Cont.)	Inspect how the tank is anchored to concrete (with chairs and angles or simply bolted down). For elevated tanks which are supported on slender legs and with no diagonal bracing, check the attachment of tanks to legs; obtain sizes of support members if HCLPF evaluation is required Study the weld details at valve pit location; is there enough flexibility in piping? Where is the valve located with respect to the tank? Look for potential for syphoning of tanks by break of connecting pipe.	
Batteries and racks	Inspection should focus on battery support (i.e., spacers and shims), ruggedness of racks, and anchorage details. Check for potential failure of space heaters, hanging light, or blockwall enclosures that could fall and impact the batteries.	Some racks in older plants have low seismic margin.
Active Electrical Equipment	Inspect method and adequacy of anchorage for motor control centers, switchgear, instrumentation panels, cabinets and racks and other electrical components.	Look for cabinets not tied together which could impact each other.

TABLE 4-3

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES BETWEEN 0.3 AND 0.5G PGA
(Continued)

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Active Electrical Equipment (Cont.)	Look for slack in cables and airlines at equipment attachment points.	
	Look inside electrical cabinets to determine if relays or other electrical instruments are securely mounted.	If safe to do so, push on instruments to check anchorage and flexibility.
	Inspect lateral restraints on large breakers	Breakers could dislodge relative to cabinet.
Diesel Generators	Inspect peripheral components necessary for operation of equipment: - Concentrate on anchorage - Check if the diesel generator is shock mounted	For example, diesel oil day tank, compressed air equipment, lube oil cooler, control cabinet, fuel oil transfer pumps, heat exchangers, heating and venting equipment.
HVAC Systems	Fan and cooler units: Check if the units are supported on vibration isolators. If so, it needs a margin review. Examine if a lateral stop exists. Check the adequacy of anchorage systems.	
Control Room Ceilings	Inspect for adequate bracing	
	Check area above lightweight control room ceiling units for heavy reflective panels or equipment (e.g., air conditioning equipment).	Reflective panels may fall through ceiling and damage control instruments or injure operators.

TABLE 4-3

EXAMPLE ITEMS TO FOCUS ON IN A PLANT WALKDOWN DURING MARGIN REVIEW FOR EARTHQUAKES BETWEEN 0.3 AND 0.5G PGA
(Continued)

COMPONENT	ITEM TO BE EXAMINED	COMMENTS
Dams, Levees and Dikes	If they exist and their potential failure may have significant consequence to plant safety, they should be inspected to assess their existing condition and vulnerability to earthquakes.	
Pumps	For vertical pumps, verify that shafts are supported at their lower ends or the casings have lateral supports less than 20 feet apart.	
Cable Trays and Cables	Inspect representative cable trays:	
	- Examine lateral support and anchorage details; focus on anchor plate weld connections, taut cables, and sharp edges at end of cable trays.	
	- Look for slack in electrical cables between structures or between cable trays and building penetration.	If structures or trays shift, cables may break.
Retaining Walls	Inspect existing conditions of retaining walls which support soil that contains safety-related lines (e.g., service water piping).	If wall fails, the buried piping may fail.

TABLE 4-4

FRAGILITY PARAMETERS OF COMPONENTS IN A HYPOTHETICAL PLANT

Component	A_m (g)	β_R	β_U	HCLPF (g) *
1	0.90	0.20	0.45	0.30
2	0.90	0.24	0.43	0.30
3	1.00	0.24	0.33	0.40
4	1.30	0.24	0.49	0.40
5	0.90	0.30	0.36	0.30
6	1.60	0.25	0.43	0.50

* HCLPF = A_m X exp -1.65 $(\beta_R + \beta_U)$

4.9 Seismic Margin Review Team Qualifications

The seismic margin review team should consist of a review team leader, several systems analysts, several fragility analysts, and at least one representative from the organization sponsoring the review. This section presents the general qualifications for the review team members.

Each review team member should have an overall understanding of the seismic margins review methodology presented in this report including the limitations and background discussed in Ref. 1.

The review team leader should have the following qualifications:

- Knowledge and understanding of nuclear plant operations and components, including having seen and toured a number of nuclear power plants.
- Some understanding of seismic risk assessment of nuclear power plants.
- Some understanding of systems analysis and component fragility-analysis techniques.
- Demonstrated ability to coordinate the activities of the team members in the performance of the various steps of the review so that proper information is efficiently processed as the steps are completed.
- Ability to organize and coordinate team meetings and make arrangements for the plant walkdowns.

The team of fragility analysts performing the review should have the following qualifications:

- The capability to perform earthquake engineering of nuclear power plants and/or heavy industrial facilities.
- The ability to comprehend and use nuclear design standards and practices for structures and equipment.
- The ability to identify failure modes of equipment and recognize systems interactions.
- Demonstrated capability to perform fragility/margins evaluations including structural/mechanical analysis.
- Participate in the plant walkdowns and be responsible for performing the HCLPF calculations.

It would also be useful if the fragility analysts have some general understanding of systems analysis, and seismic risk assessment.

The team of systems analysts performing the review should have the following qualifications:

- Experience in nuclear power plant design and analysis.

- Familiarity with nuclear power plant systems configurations and operations, and a good understanding of the characteristics and operation of nuclear power plant components.

- Previous participation in the performance of at least one nuclear power plant probabilistic risk assessment with experience in accident behavior and the development and analysis of fault trees and event trees.

It would also be helpful if the systems analysts have a knowledge of system interactions as well as knowledge of seismic risk assessment.

CHAPTER 5.

RESOURCES AND SCHEDULE

This chapter describes the engineering resources needed to perform a seismic margin review of a nuclear power plant following the methodology described in this report. These estimated staffing requirements may be revised based on the findings of the trial plant reviews. The staffing requirements for performing the margin review and for an independent peer review are also described.

5.1 Review Team

We assume that a earthquake review level has been defined. The major tasks of a margin review are:

- Collection of design information
- Plant walkdowns
- Systems analysis
- HCLPF calculations for the components and the plant
- Internal Review
- Report writing

The team doing the margin evaluation should consist of engineers experienced in plant systems analysis and structural and mechanical/electrical equipment fragility analysis. Their qualifications are described in Section 4.9. Table 5-1 gives an estimate of the manpower requirements for this review. For an earthquake review level up to 0.3g pga, the total engineering time could be between 27 months and 39 months. For earthquake review levels between 0.3g and 0.5g pga the manpower requirement could be 36 to 48 months. The time requirement depends on the extent of seismic response analysis and the number of screened-in components requiring detailed HCLPF calculations.

5.2 Independent Peer Review

An independent peer review is essential for any seismic margin study. Such a peer review should consist of:

- A review of the seismic margin study report.

- An independent check of the information and plant-specific data used in the study.

The peer review team is also expected to conduct a plant walkdown concurrent with the first walkdown described in Section 4.4.

The major tasks of the peer review team are:

- Review of the seismic margin study
  - Plant Familiarization
  - Report
- Plant walkdown

The peer review team should also consist of engineers experienced in plant systems analysis, structural and mechanical/electrical equipment, and fragility analysis.

5.3 Schedule

The seismic margin review is expected to be completed in eight months as shown in Figure 5-1.

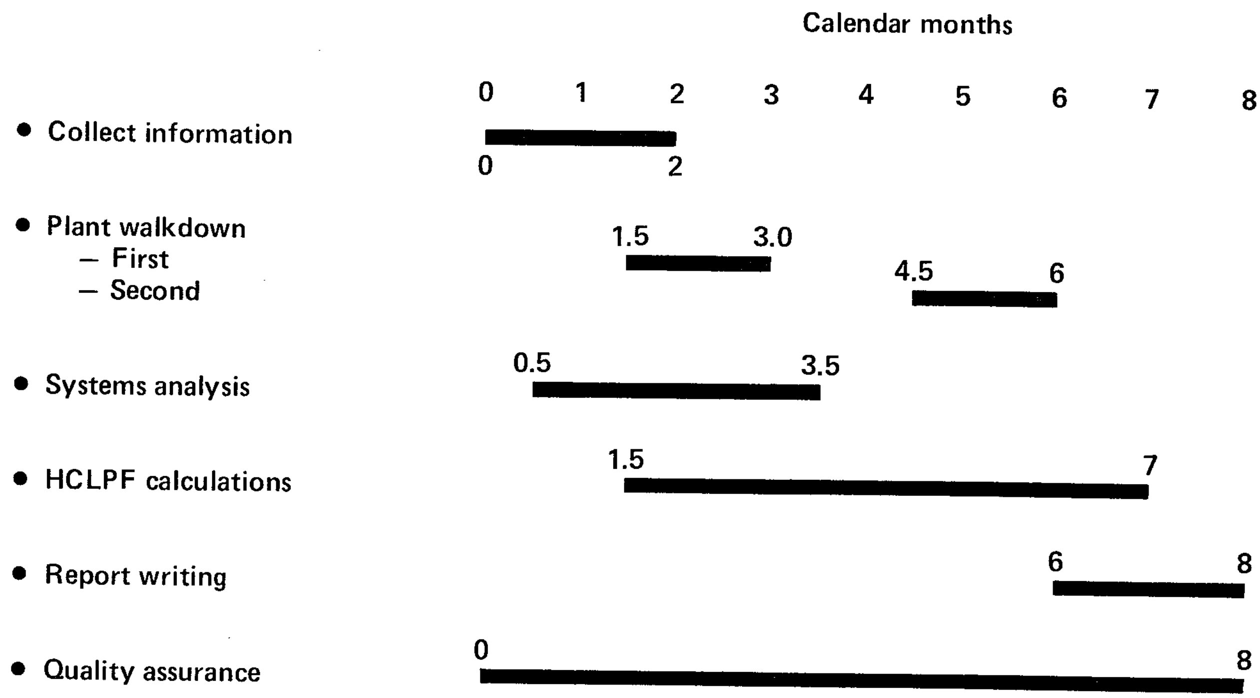


Fig. 5-1 Schedule for Seismic Margin Review.

TABLE 5-1

ESTIMATED STAFFING REQUIREMENTS FOR SEISMIC MARGIN REVIEW

	Engineering Months Needed	
	Systems	Fragilities
o Collect Information on Design	1.0	2.0
o Plant Walkdown		
- First	1.0	2.0
- Second	0.5	3.0
o Systems Analysis	3.0	
o HCLPF Calculations of Component and Plant	0.5	6.0[b,c]
o Internal Review	1.0	2.0
o Report Writing	1.0	4.0
	8.0	19.0

Total engineering time = 27 engineering-months (see d and e below).

NOTES

a. Staffing requirements for future seismic margin reviews may be revised based on the experience of trial plant reviews.

b. Depends on the number of screened in components requiring detailed HCLPF calculations. For earthquake review levels between 0.3g and 0.5g, 30 to 40 additional items (i.e. structures and equipments) may need detailed HCLPF calculations. An additional 9 engineering months are estimated as needed for this evaluation.

c. It is assumed that the design seismic responses can be scaled to obtain the response for the review level earthquake. If scaling cannot be done, reanalysis of structures and equipment is a big cost factor requiring an additional 12 engineering months. For soil sites, this reanalysis is considered necessary.

d. For rock sites scaling of design responses is permissible; otherwise, total engineering time = 39 engineering-months. For review level earthquakes between 0.3g and 0.5 pga, total engineering time = 36 engineering-months if scaling of design responses is permissible; otherwise, it is 48 engineering-months.

e. Additional engineering time needed by the utility to assist in data collection, plant walkdown, project review and coordination is estimated to be about 6 engineering-months.

TABLE 5-2

ESTIMATED STAFFING REQUIREMENTS FOR INDEPENDENT PEER REVIEW OF SEISMIC MARGIN REVIEW

	Engineering Months Needed	
	Systems	Fragilities
o Review of Seismic Margin Study		
- Plant Familiarization	1.0	2.0
- Review of Report	0.5	1.5
o Plant Walkdown	0.5	1.0
o Presentation of Review Results	0.5	1.0
	2.5	5.5

Total engineering time = 8.0 engineering-months.

CHAPTER 6.

FORMAT FOR A TYPICAL SEISMIC MARGIN REPORT

In this chapter we describe the contents and format of a seismic margin report. The objective of this format is to outline clearly the information that should be included in a typical margin report. The report should give a description of the plant safety systems and their design bases, the methodology utilized in the margin review, and the conclusions of the study. The information to be included in different sections of a margin report is discussed below.

Documentation of the trial plant review methods and results should be more extensive than in a typical seismic margin report as discussed in Section 6.8.

6.1 Purpose of Study

The report should describe the earthquake review level selected for margin review of the plant.

6.2 Plant Safety Systems and their Seismic Design Bases

The plant safety systems should be described with particular reference to their required performance during and after a large earthquake. The plant description should include a brief discussion of the principal design criteria, operating characteristics, and safety considerations for the nuclear steam supply system; the engineered safety features and emergency systems; the instrumentation, control, and electrical systems; the power conversion system; the fuel handling and storage systems; and the cooling water and other auxiliary systems. The components that comprise these systems should be tabulated as shown in Table 6-1.

The seismic design basis of the plant should be summarized; as a minimum, the following items should be covered:

- o Applicable design codes, standards and specifications (e.g., ACI 349, AISC Specifications, and ASME Boiler and Pressure Vessel Code).

- o Conformance to NRC Regulatory Guides (e.g., R.G. 1.60 and 1.61).

- o Seismic Design Ground Response Spectra; SSE and OBE peak ground acceleration values along with damping values for structures, systems, components and soil.

- o Seismic Qualification Methods. For equipment qualified by analysis, describe the methods of analysis (e.g., response-spectrum, time-history, static), of combining modes, of combining earthquake responses of directional components, of soil-structure interaction analysis, and of floor spectra generation.

For equipment qualified by testing, describe the type of test (e.g., sine-sweep, sine-beat, and complex-wave form; single or multi-directional), and the difference between the test response spectrum and the required response spectrum.

The quality and quantity of plant design data available for the seismic margin study should be described.

6.3 Screening of Systems and Components

The frontline and support systems that provide the Group A plant safety functions should be identified and presented in a tabular form (See Tables 6-3, 6-4, and 6-5 for examples). The components (i.e., structures, equipment, piping systems, HVAC systems, cable trays, and cabling) of these systems should be identified and tabulated (similar to Table 6-1). This list of components is a subset of the list developed in Section 6.2.

Components that belong in systems supporting Group A safety functions and that are screened out because of generic high seismic capacities should be identified. Methods used to confirm that the screened out high capacity components do in fact have no weaknesses should be documented. This may consist of review of the Panel recommendations (contained in Table 2-2), plant walkdown, and some limited plant-specific calculations. The list of components screened in should be augmented by including plant-unique features and system interactions identified in a plant review and walkdown.

6.4 Evaluation of Seismic Margins

Event trees and fault-trees developed using the screened-in components should be described and the accident (core melt) sequences identified. The "singles" (event) and "doubles" (event) cut sets or the Boolean expression should be tabulated for these accident sequences.

An estimate of the HCLPF of components that appear in the "singles" and "doubles" cut sets should be provided (See Table 6-2 for example). The procedures used in estimating the HCLPF of singles and doubles should be described. If a plant-level HCLPF is calculated, the procedure used should be described. Also, the methods used in the structural/mechanical analyses should be documented.

6.5 Plant Review and Walkdown

The results of a review of plant seismic design criteria and seismic design analyses should be discussed with particular emphasis on how similar the components in the plant are to those generic categories discussed in Table 2-1.

The procedure used, the personnel involved (including their background), and the findings of the plant walkdown should be described. The results of the first plant walkdown should be reported in terms of confirmation of the presence of safety systems per design documents and the absence of weaknesses in the assumed high seismic capacity components should be reported. Also, any plant-unique features, systems interactions, and deviations from design

documents should be documented. The results of the second plant walkdown should be described in terms of the physical data used for HCLPF evaluation.

6.6 Conclusions

The final chapter of the seismic margins report should contain the following:

- o Tables showing the screened out systems and components and basis for such screening.
- o Table showing the HCLPF values of screened-in components and estimate of plant level HCLPF.
- o Identification of low capacity elements in the plant and results of any studies performed to confirm their seismic capacities.
- o Statement as to whether the plant has capacity beyond the earthquake review level.

6.7 Documentation of Trial Plant Reviews

The objectives of trial plant reviews are to test the applicability of this seismic margin review methodology and to improve the methodology based on the results and insights gained in such trial plant reviews. In order to accomplish these objectives, the procedures used and the results obtained in the trial plant reviews should be thoroughly documented for the Expert Panel's peer review and subsequent refinement of seismic margin review guidelines.

Documentation should include, in addition to the items previously described in this chapter, the following items:

- o Basis for selection of the plant(s) for trial review and how the review proves the applicability of the seismic margin review methodology for other plants.
- o Screening tools used in plant reviews (e.g., screening tables and simplified analyses).
- o Assumptions made in the HCLPF calculations.
- o Comparisons between HCLPF values obtained using different candidate methods.
- o Detailed description of design review and plant walkdowns; include design documents, system descriptions, field notes, photos of equipment and sketches.
- o Engineering hours spent on various tasks of the trial plant review(s).
- o Any difficulties experienced in following the Panel's guidelines and deviations taken in the review.
- o Further specific guidelines and procedures needed.

TABLE 6-1

DESCRIPTION OF COMPONENTS OF SAFETY SYSTEMS

System	Component	Location	Seismic Qualification Method/Reference
NSSS	Steam Generator	Containment Lower Support El. 538' Upper Support El. 573'	Dynamic Analysis Calculation Package #
Residual Heat Removal	RHR Pumps	Aux. Bldg. El. 495'	Dynamic Analysis Calculation Package #
Electrical Power	4160V Switchgear	Aux. Bldg. El. 567'	Test Seismic Qualification Report #
. . .	. . .	. . .	. . .

TABLE 6-2

HCLPF VALUES OF CRITICAL COMPONENTS IN GROUP A FUNCTIONS

Cut set	HCLPF Values of Components (g)	HCLPF Value of Cut set (g)
Singles		
EGECLPSE	0.30	0.30
EDGOILCL	0.30	0.30
DFCNTBLD	0.39	0.39
Doubles		
RWST * COREGEOM	(0.30, 0.35)	0.35
EGECLPSE * CRDS	(0.30, 0.33)	0.33

TABLE 6-3

PLANT FUNCTIONS VS FRONT LINE SYSTEMS MATRIX

	Reactor Protection System	Power Conversion System	High Pressure Injection System	Low Pressure Injection System	Accumulator System	Auxiliary Feedwater System	Recirculation System	Quench Spray System	Primary SRV System	Secondary SRV System	Reactor Coolant Pump Seal Cooler
Reactor Subcriticality	X		X								
Normal Cooldown		X									
Emergency Core Cooling (Early)			X	X	X	X			X	X	X
Emergency Core Cooling (Late)			X				X				
Containment Heat Removal							X				
Containment Overpressure Protection (Early)								X			
Containment Overpressure Protection (Late)							X				

TABLE 6-4

FRONT LINE SYTEMS VS SUPPORT SYSTEMS MATRIX

	Eng. Safety Fact. Act. System	AC Power System	DC Power System	Charging Pump Cooling System	Safety Injection Pump Cooling System	Reactor Pump Comp. Coolant Water	Turbine Pump Comp. Coolant Water	Circulating Water System	Service Water System
Reactor Protection System	X		X						
Power Conversion System		*	X				X	X	
High Pressure Inj. System	X	X	X	X	X				
Low Pressure Inj. System	X	X	X						
Accumulator System									
Auxiliary Feedwater System	X	X	X						
Recirculation System	X	X	X						X
Quench Spray System	X	X	X						
Primary SRV System			X						
Secondary SRV System			X						
RCP Seal Cooling System						X			

* Requires offsite power

TABLE 6-5

SUPPORT SYSTEMS VS SUPPORT SYSTEMS MATRIX

	Eng. Safety Feat. Act.	AC Power System	DC Power System	Service Water System
Engineered Safety Features Actuation System	**		X	
AC Power System	X	**	X	X
DC Power System			**	
Charging Pump Cooling System	X	X	X	X
Safety Injection Pump Cooling System	X	X	X	X
Reactor Plant Component Cooling Water System		X	X	X
Turbine Plant Component Cooling Water System		X	X	X
Circulating Water System		*	X	
Service Water System	X	X	X	**

* Requires offsite power

** Not applicable

GLOSSARY

GLOSSARY

ACCIDENT SEQUENCE

A combination of failures and/or successes of plant systems or functions beginning with an initiating event. Accident sequences are branches of event trees. Several accident sequences are possible for a given event tree, each describing a different branch. Each accident sequence describes a unique combination of event failures and/or successes.

ASEP

Accident Sequence Evaluation Program. A long term NRC program intended to provide generic models of fault-trees and event-trees for simplified estimation of accident-sequence frequencies of core melt for U. S. nuclear power plants.

ATWS

Anticipated transient without scram. Transient initiating event with an associated failure to effectively stop the fission chain reaction.

BASIC EVENTS

The lowest level of event representation on a fault tree. A basic initiating fault requiring no further development.

BEST ESTIMATE VALUE

Any reasonable statistic can be considered as the proper choice of estimator of a given parameter. The question of choosing the best estimator is ambiguous until "best" is defined. Desirable properties of an estimator could be, for example, being unbiased or promising the minimum mean square error. The best estimate fragility is chosen by the Panel as the median curve.

BOOLEAN EXPRESSION

Boolean algebra deals in situations involving dichotomy. Switches are either open or closed, valves are either open or closed, events either occur or they do not occur. It is defined as the algebra of events the product of which is a set of logic equations that summarize the possible states of a system. A fault tree can be thought of as a pictorial representation of those Boolean relationships among fault events that cause the top event to occur.

CAPACITY

The ability of a component to sustain a load measured in terms of the load level (e.g., stress, moment, or acceleration) below which the component continues to perform its functions. Not to be confused with fragility. In Fig. G-1 on the following page, curve 2 is associated with a component having higher capacity (strong component) than the one associated with curve 1 (weak component).

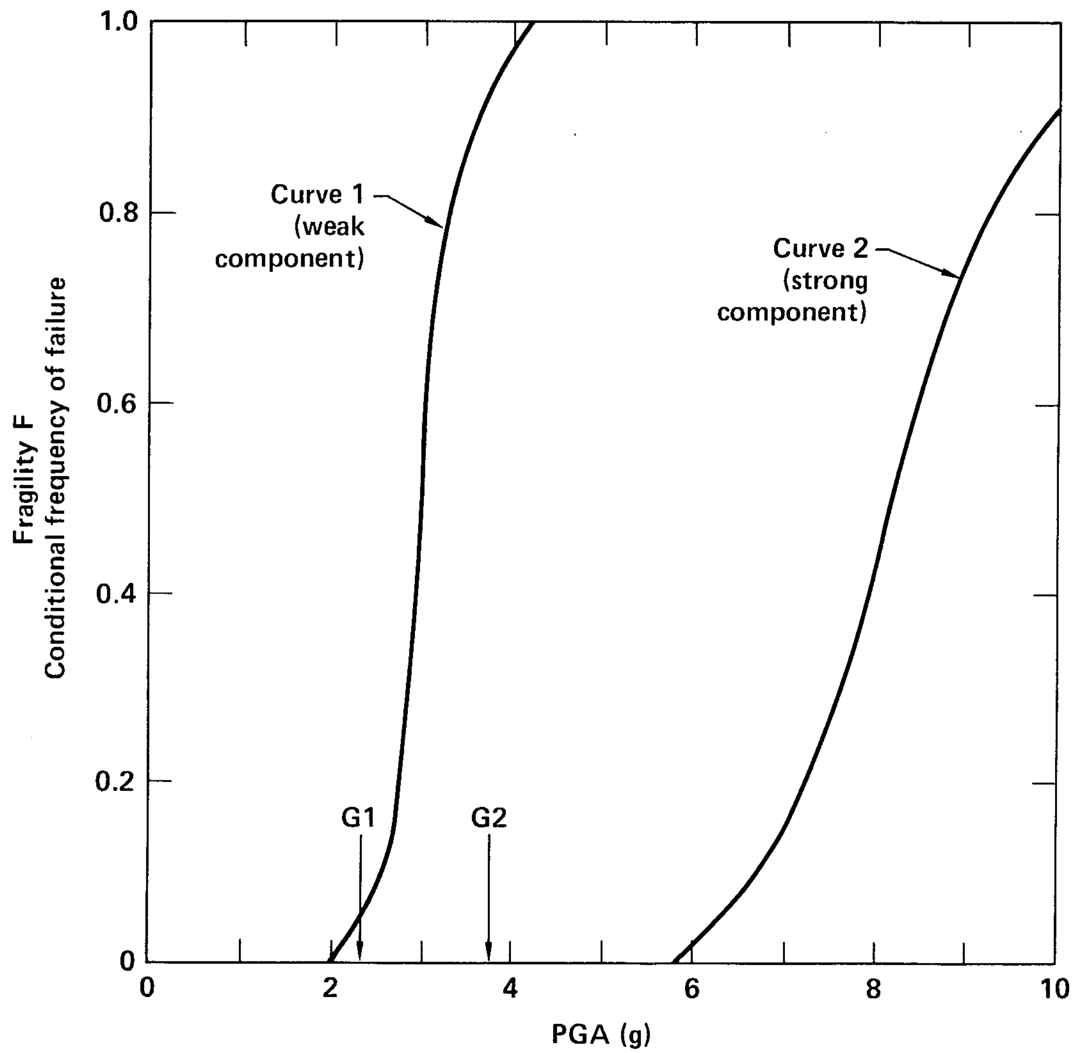


Fig. G-1. Fragility Curves. For a pga equal to G1, the weak and strong components both have a low probability of failure (i.e., low fragility). At level G2, the weak component (i.e., the one with low capacity) has a high probability of failure (i.e., high fragility) and the strong component (i.e., high capacity) has a low probability of failure (i.e., low fragility).

CODE MARGIN

Margin computed as follows:

a) using structural response parameters (such as damping) less stringent than those used in the Standard Review Plan,
b) using only normal loads plus seismic loads,
c) assuming essentially elastic behavior, and
d) assuming capacities defined by code and the Standard Review Plan.

COMMON CAUSE FAILURES

The primary failures (of components) on a fault tree are not necessarily independent. There may be a single cause for multiple component failures. When this cause is not intrinsic to the components, but is a result of an external effect, it is a common cause failure (e.g., an operator may miscalibrate all the sensors, a steam line break may cause all instrumentation to fail in a control panel or an earthquake may cause multiple component failures).

COMPONENT

A general term used for a structural element, system, an electrical or mechanical piece of equipment, a piping system, or an electrical cable system.

CORE MELT BINS

See "PLANT DAMAGE STATE."

CORE MELT FREQUENCY

An estimate of the annual frequency of occurrence of all accident sequences involving core melt.

CUT SET (Minimal)

A minimal cut set is the smallest combination of component failures that, if they all occur, will cause the top event to occur. It is a combination (intersection) of primary events sufficient for the top event. The combination is a "smallest" combination in that all the failurés are needed for the top event to occur; if one of the failures in the cut set does not occur, then the top event will not occur (by this combination).

DESIGN ACCELERATION

A specification of the ground acceleration, such as the SSE, at a site in terms of a single value such as the peak. It is used for the earthquake-resistant design of a structure (or as a base for deriving a design spectrum).

DESIGN EARTHQUAKE

A specification of the seismic ground motion at a site; used for the earthquake-resistant design of a structure.

DESIGN EVENT, DESIGN SEISMIC EVENT

A specification of one or more earthquake source parameters, usually magnitude and the location of energy release with respect to the site of interest; used for the earthquake-resistant design of a structure.

DESIGN GROUND MOTION

See "DESIGN EARTHQUAKE."

DESIGN SPECTRA

A set of response spectra for design purposes. (See "RESPONSE SPECTRA.")

DOMINANT PLANT DAMAGE STATE

The plant damage state that contributes most to the frequency of core melt.

DOUBLE, DOUBLET, DOUBLETON

A minimal cut set made of the intersection of two basic events (i.e., for the top event to occur, both basic events have to occur).

DURATION

A qualitative or quantitative description of the length of time during which ground motion at a site shows certain characteristics (e.g., perceptibility, violent shaking, etc.).

EARLY MELT

A term used to indicate the time at which the core melt event occurs in relation to the operation of safety systems designed to prevent it. Early core melt occurs as a result of failure of safety systems designed to provide core cooling immediately upon initiation of a severe accident.

EARTHQUAKE SIZE

As used here and in most nuclear-power-plant-related risk studies, the size of an earthquake is measured in terms of the pga, rather than magnitude, intensity or any other geophysical parameter. (See Chapter 2 of Ref. 1.)

EVENT TREE

This term defines sequences of system failures that may lead to the release of radioactive material. Each tree is associated with an initiating event. The event tree method begins with an initiating event, tracks subsequent events of various plant systems, and determines the probability of the various accident sequences.

EXCEEDANCE PROBABILITY

The probability that a specified level of ground motion or specified social or economic consequences of earthquakes will be exceeded at a site or in a region during a specified exposure time.

EXPOSURE TIME

The time period of interest for seismic-risk calculations, seismic-hazard calculations, or design of structures. For structures, the exposure time is often chosen to be equal to the design lifetime of the structure.

FAILURE MODE

Failures are basic abnormal occurrences. A failure mode specifies exactly which aspects of component failures are of concern. Failure modes should be realistic and consistent within the context of the system operational requirements, environmental factors, and the numerical data base.

FAULT TREE

A fault tree analysis can be simply described as an analytical technique, whereby an undesired state of the system is specified (usually a state that is critical from a safety standpoint), and the system is then analyzed in the context of its environment and operation to find all credible ways in which the undesired event can occur. The fault tree itself is a graphic representation of the various parallel and sequential combinations of faults that will result in the occurrence of the predefined undesired event.

FILTER

A series of criteria used to sort the components of a nuclear power plant into different classes. In this report, this filtering process is sometimes called "a screening operation" or "sorting." The criteria used to perform the filtering are based on the capacity level (HCLPF) of the components relative to the earthquake review level and a determination as to whether they support Group A functions.

FRAGILITY

Conditional probability that a component would fail for a specified ground motion or response-parameter value as a function of that value. (See "CAPACITY.")

FRAGILITY CUTOFF

The lower tail of a fragility curve may be truncated at some peak ground acceleration (PGA) value on the basis that seismically qualified components do not fail below this value.

FRONT LINE SYSTEMS

Systems that are necessary to perform a safety (Group A) function.

FREQUENCY (of occurrence)

In this report: rate per unit of time. The mean rate of occurrence of earthquakes is a measure of frequency.

GROSS ERRORS

Design and construction errors are those deviations that place the component capacity outside the range of natural variation; gross errors are those that have significant impact on components or plant margins.

GROUND ACCELERATION CAPACITY

The seismic capacity of a component measured in terms of the peak ground acceleration value at which the component would fail.

HAZARD CURVE (seismic)

Frequency of exceeding a pga versus pga, usually expressed on a per-year basis.

HCLPF

High Confidence of Low Probability of Failure. The concept of the HCLPF is discussed in Section 2.3 of Ref. 1.

INHERENT RANDOMNESS

The variability observed from sample to sample of a physical phenomenon; it cannot be reduced by more detailed evaluation or by gathering of more data.

INITIATING EVENTS

Events that exceed the operating allowable limits of a nuclear power plant, therefore requiring that the plant be shut down. An event tree is associated with each initiating event. Two major categories of initiating events are recognized: pressure-boundary rupture and transient initiation. These categories are subdivided according to the capabilities of the particular plant systems required to maintain the plant in a safe condition. An example of a pressure-boundary rupture is the rupture of a large pipe. A transient initiation does not involve rupture (e.g., the trip of a turbine).

INTERNAL INITIATING EVENT

An initiating event that is caused by occurrences or failures that are internal to the physical systems of the plant. These events are in contrast to external initiators that result from occurrences external to the physical plant systems such as earthquakes, tornados, floods, fires, aircraft impact, and explosions.

ISAP

Integrated Safety Assessment Program. A pilot program of the NRC intended to prioritize safety issues on a plant-specific basis using available analyses and PRA results.

LATE MELT

A term used to indicate the time at which the core melt event occurs relative to the operation of safety systems designed to prevent it and the amount of radioactivity present in the core. Core melt is "late" when it occurs after the successful operation of safety systems designed to provide core cooling immediately upon the initiation of a severe accident; and after the failure of those systems designed to provide core cooling or containment cooling in the long term.

LOCA

Loss-Of-Coolant Accident. An accident caused by a break in the reactor coolant system pressure boundary.

MAGNITUDE

Magnitude is a measure of the size of an earthquake and is related to the energy released in the form of seismic waves. It is given by the logarithm of the amplitude of the trace of a seismometer, corrected to represent a set of fixed conditions. There are several magnitude scales that are representative of different frequency ranges of the incoming seismic energy.

MEAN PEAK GROUND ACCELERATION (mean pga)

In this report, the mean value of the peaks of the two horizontal components of acceleration.

MEDIAN VALUE

The value of a variable that represents the 50th percentile of the probability distribution function. That is, for any given subset of the population, it is equally likely that the result of a trial will yield a value above or below the median. In this report, the median of a parameter, A, is indicated as A_m.

MINIMAL CUT SET

See "CUT SET."

OFF-SITE RISK

The probability that social or economic consequences of earthquakes will equal or exceed specified values at a site, at several sites, or in an area, during a specified exposure time.

OPERATING BASIS EARTHQUAKE (OBE)

An earthquake that, considering the regional and local geology, seismology, and specific characteristic of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. It is an earthquake that produces vibratory ground motion that will allow the continued operation of those features necessary for continued operation of a nuclear power plant without undue risk to the health and safety of the public. (See 10 CFR Part 100, Appendix A: Reactor Site Criteria.)

PEAK GROUND ACCELERATION (pga)

See "MEAN PEAK GROUND ACCELERATION."

PLANT DAMAGE STATE

A grouping of accident sequences causing similar physical responses in a plant. Characteristics of each plant damage state are used to define initial conditions for the subsequent analysis of containment failure mode and radiological release (see release categories). Same as plant damage bin, core melt bin, and core damage state.

PLANT LEVEL FRAGILITY

The conditional probability that plant systems would fail and lead to severe core damage (or release) for a specified earthquake ground-motion-parameter value.

PLANT SYSTEM AND SEQUENCE ANALYSIS

The identification of the sequences of events signifying the success or failure of safety systems, beginning with the specified initiating event, progressing through a planned reactor shutdown or severe core melt, and terminating in radiological releases to the environment. The frequencies of various events in these sequences are estimated using Boolean expressions and component fragilities.

PROBABILISTIC RISK ASSESSMENT (PRA)

See Ref. 1 Appendix E and Chapter 3 of this report.

RANDOM FAILURE

Those failures that occur normally during plant operation not affected by seismic events.

RELEASE CATEGORIES

A measure of the type, amount, energy, and timing of radioactive material released from the plant. These are functions of accident sequences and containment failure modes.

RELEASE FREQUENCY

Frequency of occurrence of a group of accident sequence resulting in a specific release category.

RESPONSE SPECTRA

A set of curves calculated from an acceleration time history that give the maximum values of response (acceleration, velocity, or displacement) of a damped linear oscillator, as a function of its natural period of vibration for given damping values.

REVIEW

As described in this report, the review constitutes the act of performing the necessary operations (i.e., analysis of available data, physical inspections of the plant, detailed mechanical and structural analyses, and systems analysis) to assess the seismic margins at the plant.

RMIEP

Risk Methods Integration and Evaluation Program. An NRC research program intended to integrate all available PRA knowledge and apply it (as a demonstration) to the LaSalle Nuclear Power Plant. The final result will be a complete PRA of LaSalle (Level 3 PRA, Ref. 4) and an integrated methodology for use in future PRAs.

SAFE SHUTDOWN EARTHQUAKE (SSE)

A regulatory design requirement defined as follows: that Earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology, and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are those necessary to assure: (1) the integrity of the reactor-coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequence of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10 CFR Part 100.

SCREENING

See "FILTER."

SEISMIC CAPACITY

See "CAPACITY."

SEISMIC CATEGORY I

For licensing purposes, nuclear plant structures and equipment are divided into two categories according to their function and the degree of integrity required to protect the public. These categories are Category I and non-Category I. Structures, systems, and components important to safety, as well as their foundations and supports, are designed to withstand the effects of an OBE and an SSE, and are thus designated as Seismic Category I. These plant design features are those necessary to assure: a) the integrity of the reactor coolant pressure boundary, b) the capability to shut down the reactor and maintain it in a safe shutdown condition, or c) the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guideline exposures of 10 CFR Part 100.

SEISMIC HAZARD

Any physical phenomenon (e.g., ground shaking, ground failure) associated with an earthquake that may produce adverse effects.

SINGLE, SINGLET, SINGLETON

A minimal cut set made of a single basic event.

SORTING

See "FILTER."

SQUG

Seismic Qualification Utility Group. A group of utilities joined together to gather experience data on component behavior in past earthquakes.

SSMRP

The Seismic Safety Margins Research Program was a U.S. Nuclear Regulatory Commission program conducted by Lawrence Livermore National Laboratory. It developed a complete, fully coupled analysis procedure for estimating the risk of an earthquake-induced radioactive release from a commercial nuclear power plant. The procedure traces the seismically induced failure modes in a reactor system down to the individual component level and takes into account the common-cause earthquake-induced failures at the component level.

SSRAP

Senior Seismic Review and Advisory Panel. A panel comprised of: R.P. Kennedy (chairman), Paul Ibanez, Ansel Schiff, Walt von Riesemann, and Loring Wyllie, whose function is to review and interpret the SQUG data.

STATION BLACKOUT

Total loss of all (on-site and off-site) AC power sources.

SUPPORT SYSTEMS

Systems that are necessary for the operation of front line systems (e.g., cooling systems, lubrication oil systems, pneumatic systems, hydraulic systems).

TOP EVENT

The undesired event on the top of a fault tree. A fault tree is constructed for each system. Under seismic excitation, components fail, leading to the system's inability to serve its functions. The system's inability to serve its functions is the top event for the fault tree.

WALKDOWN

A step in the review process during which data is gathered, assumptions on component capacities are checked, and analysis is performed. The two plant walkdowns described in this report rely on physical inspection of the plant as well as a review of written documentation, drawings, and the performance of mechanical and structural analyses.

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This report is the third report of the Expert Panel on the Quantification of Seismic Margins. The objective of this report is to present detailed guidelines for the performance of seismic margin reviews of nuclear power plants. The guidelines presented in this report are based on the Panel's second report entitled "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants." It is intended that these guidelines be used in at least one trial plant review to demonstrate whether the approach and the quantification techniques are adequate. Based on lessons learned from these trial reviews, the Panel can then be more prescriptive about defining guidelines for general use.

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