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Probabilistic safety assessment for seismic events

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FOREWORD

Probabilistic safety assessment (PSA) is increasingly important in the safe design and operation of nuclear power plants. The activities of the International Atomic Energy Agency in this area are focused on facilitating the use of PSA by reviewing the techniques developed in Member States, assisting in the formulation of procedures and helping Member States to apply such procedures to enhance the safety of nuclear power plants.

In this context a set of publications is being prepared to establish a consistent framework for conducting PSA and forms of documentation that would facilitate the review and utilization of the results. Since December 1986 several Advisory Group meetings, Technical Committee meetings and Consultants meetings have been convened by the IAEA in order to prepare the publications.

The lead publication for this set establishes the role of PSA and probabilistic safety criteria in nuclear power plant safety. Other publications present procedures for the conduct of PSA in nuclear power plants and recognized practices for specific areas of PSA, such as the analysis of common cause failures, human errors and external hazards and the collection and analysis of reliability data.

The publications are intended to assist technical persons performing or managing PSAs. They often refer to the existing PSA literature, which should be consulted for more specific information on the modelling details. Only those technical areas deemed to be less well documented in the literature have been expanded upon. The publications do not prescribe particular methods but they describe the advantages and limitations of various methods and indicate the ones most widely used to date. They are not intended to discourage the use of new or alternative methods; in fact the advancement of all methods that achieve the objectives of PSA is encouraged.

This Technical Document on Probabilistic Safety Assessment for Seismic Events is mainly associated with the Safety Practice on Treatment of External Hazards in PSA and discusses in detail one specific external hazard, i.e. earthquakes.

The IAEA wishes to convey its thanks to all those who participated in the drafting and review of the publication.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

IAEA Member States are showing increased interest and activity in probabilistic safety/risk assessment for nuclear power plants (NPPs). In response to this interest, the IAEA is leading a broad scope activity to prepare safety guides and safety practices for probabilistic safety assessment (PSA) studies [1, 2]. Within this context, the importance of external events in nuclear power plant safety and consequently in PSA studies is recognized [3].

According to the experience of Member States in PSA studies of operating NPPs, one of the most important external events affecting NPP safety is earthquakes. In some cases, the risk caused by earthquakes has been found to be comparable with the risk caused by internally initiated events.

The IAEA is undertaking activities to prepare guidelines for seismic PSA in conjunction with revision of earlier documents concerning seismic safety [4, 5]. A seismic PSA may be conducted for one or more reasons which include, but are not limited to:

- response to regulatory requirement for licensing a new plant;
- resolution of existing seismic issues;
- resolution of new seismic issues;
- development of a risk management program;
- performance of backfit cost/benefit studies;
- plant life extension.

The general method described herein is current practice and is applicable to all of the potential applications. The depth to which the study is conducted may depend upon the objective and the level of seismicity at the site. The general methodology is also applicable to all types of reactors and has been utilized to date on commercial PWRs, BWRs, LMFBRs, research reactors and military reactors worldwide.

1.2. OBJECTIVES OF SEISMIC PSA

The main objectives of seismic PSA are:

- to develop an appreciation of accident behaviour;
- to understand the most likely accident sequences;
- to gain an understanding of the overall likelihood of core damage;
- to identify the dominant seismic risk contributors;
- to identify the range of peak ground acceleration that contributes significantly to plant risk;
- to compare seismic risk to risk from other events.

Aside from giving an order of magnitude of the frequency of core damage associated with earthquakes, which might contribute appreciably to the total frequency of core damage, a seismic PSA involves a number of beneficial side effects. Basically, these benefits are all related to obtaining useful information which is difficult to obtain from conservative deterministic analyses that have been conventionally used in design and licensing.

In considering, for example, different possible spectral shapes, and by determining their separate effects on structures, systems and components, and by finally weighing the latter with their expected frequencies of occurrence, one has more rational bases either for checking the adequacy of existing seismic protections or for the optimal design of new ones.

The PSA makes use of alternative hypotheses regarding design basis earthquake parameters, soil properties, structural models, material characteristics, etc. In short, this sort of weighted sensitivity analysis is an instrument for detecting the weaker links in the chains leading to undesired events.

Finally, one should add that using PSA is an appropriate means for evaluating an existing NPP for a newly defined seismic input exceeding the original design basis.

1.3. PURPOSE OF THE DOCUMENT

The purpose of this document is to provide information and some measure of guidance and insight to those who are considering starting a seismic PSA. It tries to give an overall picture of the seismic PSA and attempts to bridge the gap between an internal event PSA and a seismic PSA.

This report is not intended to be an extensive manual or handbook but tries to help the reader by providing some references for further information on current practices and insights obtained by conducting seismic PSAs.

1.4. SCOPE OF THE DOCUMENT

In this document, a Level 1 PSA plus containment performance analysis is considered. Potential releases and off-site consequences are not included and are considered to be outside its scope. This document covers mainly the frequency of occurrence of ground motion, the seismic accident sequence initiators, the fragility analysis of safety related items, the capability of systems to mitigate accidents from seismic events and the integration of these aspects which might lead to a core damage. The reason for this simplification is to limit this subject to the essential aspects of the smooth integration of seismic related initiating events into the systematic framework of a PSA.

This does not imply, however, that effects of earthquakes after the event of core damage may be neglected. Some of the aspects which may have significant contribution to overall risk, but have not been considered in this document, are:

- increased probability of human error subsequent to the occurrence of a destructive earthquake,
- significant probability of damage to lifelines and other infrastructures which may have been planned for use in the context of emergency planning and evacuation.
- increased probability of delayed response to the nuclear accident (by authorities and the public) due to the interference of another catastrophic event.

This document concentrates on the PSA studies for NPPs. Other facilities such as research reactors, fuel cycle facilities, gamma irradiation facilities and fuel storage facilities are simpler in design and operation than NPPs. It will be possible to extrapolate the method used for NPPs to these other facilities.

2. OVERVIEW OF THE PROCEDURE FOR SEISMIC PSA

2.1. INTRODUCTION

This section provides an introductory overview of the seismic PSA procedure. The intent is to acquaint the reader with the various elements of the seismic PSA process, leaving detailed discussion of those elements for later sections. The framework which this section presents will help the readers' understanding of the nature and purpose of the various features and procedures described in the following sections, permitting the reader to proceed directly to the sections of greater interest.

Figure 1 illustrates schematically the method by which the seismic hazard and the seismic fragilities of individual components, systems and structures are combined to provide an estimate of the probability of various plant damage states. The seismic hazard is the initiator of the accident sequences. These sequences are considered as developing from an initiated fault condition, such as an anticipated transient without scram (ATWS) or a loss of coolant accident (LOCA). Some of these initiated fault conditions may arise from structural or mechanical failures produced directly by earthquake ground motion. The ability of the safety systems to mitigate the consequences of the initiated fault condition depends on the extent to which failures of the component plant items of those safety systems have degraded the functional capability of the system. Failure probabilities of components are derived from their seismic fragilities. Attention is called to the possibility of these initiated fault conditions arising from failures which are of negligible probability in an internal initiator PSA. These can include multiple failures whose simultaneous occurrence must be considered in the case of a seismic initiator because of its potential for generating common cause failures.

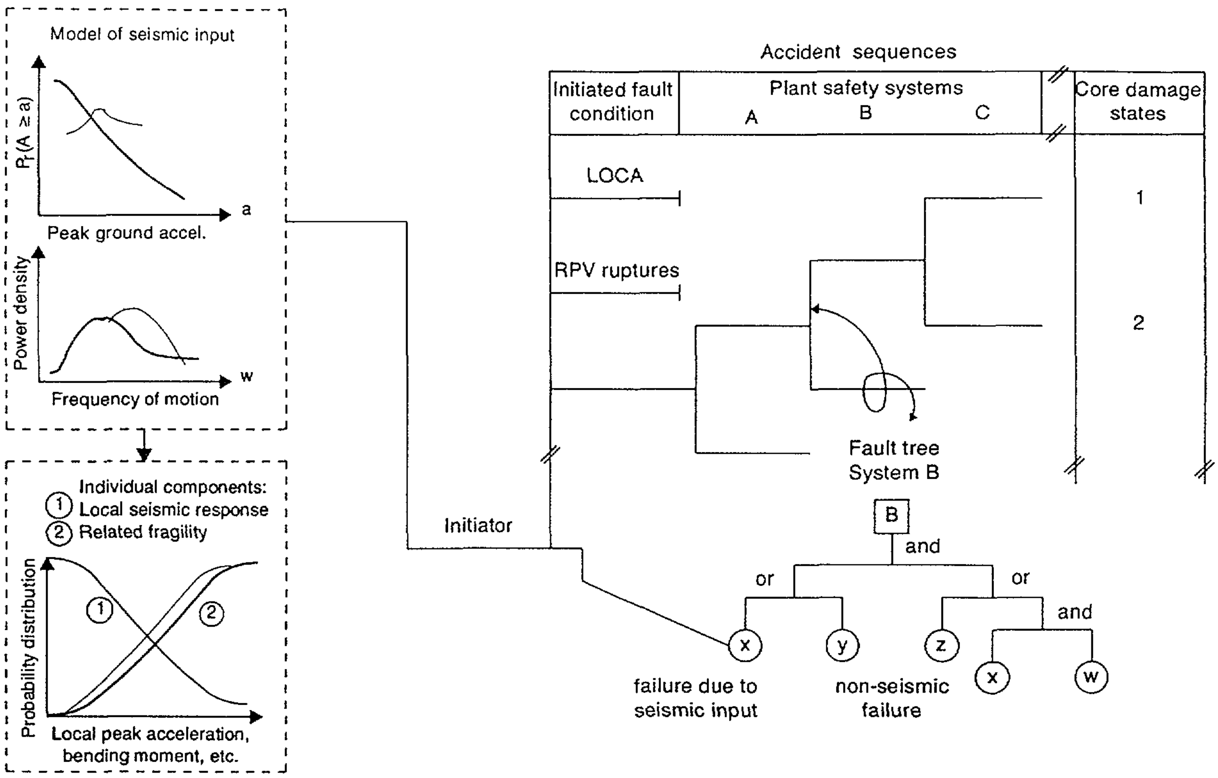


FIG. 1. Schematic overview of a seismic PSA.

For more detailed information the reader is advised to study other available literature listed in Section 8. NUREG/CR-2300 [6] and NUREG/CR-2815 [7] describe the overall process used in earlier seismic PSAs. Reference [8] contains a description of the Seismic Safety Margin Research Program (SSMRP) where sophisticated probabilistic response was calculated for structures, the primary system and selected auxiliary systems. References [9] and [10] provide details on how fragility curves are calculated for most commercial seismic PSAs using scaling of existing design information. Reference [11] gives the reader a good overview of both methods and includes discussions of partial correlations with examples. Reference [12] is a study by Sandia Laboratories that looks at risk from all events in five plants representative of US construction. There are some concepts in that document for using features of both the SSMRP program (very detailed) and the more commercial method (scaling of design information). Reference [13] is a discussion of how to use fault tree-event tree PSA methods to determine seismic margin. It is a PSA without convolution of the seismic hazard with the fragilities. Systems models and component fragilities are used to define a high confidence of low probability of failure value (HCLPF). A HCLPF is generally considered to be approximately 95% confidence of <5% probability of failure.

2.2. SEISMIC HAZARD

Knowledge of the seismic hazard is essential to the quantification process. The hazard gives the frequency of occurrence of earthquake motions at various levels of intensity at the site. The final output of the seismic hazard study should be a hazard curve (frequency of the vibratory ground motion parameter — usually acceleration at the site), and the response spectrum in the free field. A full discussion of the procedure for developing a complete model of the seismic hazard is given in Section 3.

It is convenient for the calculation to discretize the seismic hazard into a number of intervals. The upper limit of these intervals is chosen so as to be certain that no significant contributions to the assessed probability of core damage are omitted. If one is using peak ground acceleration as the intensity parameter one might consider a range of peak ground accelerations. This range might be divided into an appropriate number of subintervals, 0.1–0.25g, 0.26–0.5g, 0.51–0.75g, and 0.76–1.0g. The probability of each accident sequence would be quantified at the upper bound of each of these intervals, conditional on earthquake occurrence. To obtain the unconditional sequence probabilities, the probability of each intensity level is then factored into the conditional results. Since large intensity levels are more damaging but much less probable, the probability of damage is not an ever increasing function of intensity level. In fact, some studies have found the dominant contribution to many accident sequences to arise from lower level earthquakes because of their much greater probability of occurrence.

It is important, after the quantification process is complete, to examine the relative contribution made to the probabilities of the accident sequence, at each of the intervals into which the earthquake level is discretized. If one finds, for example, that the probability of accident sequences continues to increase from the lowest level through to the highest level, then one has not considered the entire earthquake range of importance. It is therefore necessary to do an additional calculation at an even higher level. One can only be sure that a sufficient range of earthquakes has been examined if the assessed probabilities reach a maximum within the range of levels considered and decrease as one moves to either extreme.

2.3. SEISMIC RESPONSE OF STRUCTURES, SYSTEMS AND COMPONENTS AND FRAGILITY DETERMINATION

The first step in determination of fragilities is to determine the response of structures, systems and components which give basic knowledge for future work. The final product of this step is the plant specific information which it is necessary to consider in seismic fragility analysis.

In order to obtain this plant specific information efficiently, one has to have practical experience in scaling of existing design information or to conduct selective new response analyses. As a simplifying measure it is important to focus attention initially only on the apparently most vulnerable items.

Ideally, the structural and system analysis should include definition of earthquake induced failure mode for each safety related structural element and component. One should also consider the possible systems interactions inside the plant. That is, failure of non-safety systems and components could induce the failure of safety related systems and components. Also other seismic secondary effects (like failure of a dam and possible flood at the site) may require modelling to obtain seismic response.

The development of responses must be made with consideration of the fragility quantities to be employed, and vice versa. The response data and interrelated fragilities are obtained by any of the methods described in Section 5. Here only a brief description of the role of response quantities in the quantification process is given.

There are three aspects of the response which affect the quantification of probability of core damage: the mean value of the response, its variance and the correlation between responses. Within a cut set, two individual plant items or components may not have independent failure probabilities because their responses are correlated. These correlations also enter into the calculation of the cut set probability. One can, of course, ignore the response correlation between components as a simplifying measure, but this may, in some cases, be highly unconservative.

The final fragility quantities are the complement to the response quantities developed for the seismic PSA. They are developed with full consideration of the method used to obtain the responses. For example, a certain equipment item may be much more sensitive to high frequency than low frequency motion. The fragility for that item will thus essentially depend upon the high frequency response.

As was the case for the response, the fragility data also have three aspects which enter into the quantification of probability of core damage (PCD). Two of these are the median, or mean fragility and the variance of the fragility. These quantities are combined with the median, or mean response and the response variance as shown in Fig. 2. The result would be an estimate of the conditional probability of seismically induced failure in the component or structure in question which is expressed as a fragility curve and referenced to a ground motion parameter such as acceleration.

The third aspect involves the correlation between equipment fragilities. Clearly, the failure of two pumps, both designed, manufactured, delivered, installed, tested, operated, maintained and stressed in parallel, cannot be assumed to be totally independent. They could of course be treated as such, but not without introducing some unconservatism. On the other hand, treating them as totally dependent might be excessively conservative.

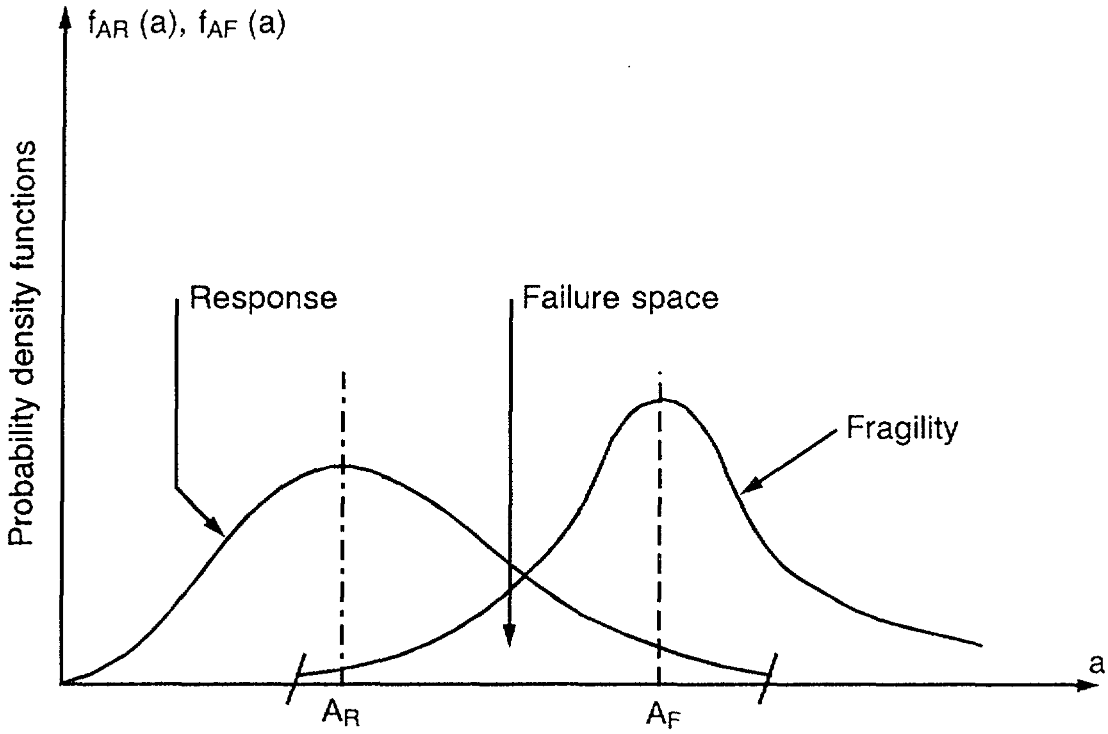


FIG. 2. Combinations of response and fragility values leading to failure.

The median, or mean fragility, its variance, and correlation between fragilities, enter into the quantification process of the cut set probabilities. These probabilities are, in turn, quantified as part of the estimation of system failure and accident sequence probabilities.

In the simulation method conducted in the SSMRP program, responses and component fragilities conditional on local responses were developed separately and combined as shown in Fig. 2. In most commercial seismic PSAs, the fragility curve is specified as a conditional probability relative to a ground motion parameter and the structural and equipment response is an integral part of the fragility description. Section 5 provides more detail on fragility curve development.

2.4. EVENT FAULT TREE CONSTRUCTION AND ACCIDENT SEQUENCE QUANTIFICATION

The method for obtaining safety system and accident sequence cut set probabilities from the logic models is discussed in Section 6.2. These cut sets must be quantified to obtain the probabilities of safety system failures or accident sequence occurrences. These quantities are useful in showing the relative contribution of systems and sequences to core damage. The calculation of cut set probabilities presents one of the most fundamental differences between seismic and internal initiator PSA. In internal initiator PSA, component failures are usually treated as independent and random events. Consequently, the probability of a cut set involving independent random failures would be evaluated simply from combinations of the random failure probabilities of each of the elements of that cut set. In a seismic PSA, the component failures represented in a cut set may be correlated through their respective responses and fragilities. This correlation is in fact one of the reasons why seismic initiators are of particular concern in nuclear power plants; earthquakes can cause the simultaneous failure of several redundant safety equipment items, and the correlation is a measure of the potential for this simultaneous failure. Calculation of the probabilities of cut sets containing correlated events involves multivariate integration of the joint probability density function of the cut set elements [14]. The process is considerably more complicated than simple multiplication but is necessary to account for the increased cut set probabilities which result from correlated failures.

2.4.1. Development of random (non-seismic) failure data

Random failures are treated in the same way in seismic PSA as they are in internal event PSA. In fact, if an internal event PSA has already been performed, then this step is quite straightforward for the seismic PSA since the data are already available for all of the randomly occuring base failures in the fault trees.

Though this step may be straightforward, it is by no means inconsequential. Some seismic PSAs have found that the dominant contributors to degraded core accident probabilities involve random failures of safety equipment to operate on demand. The failure is still considered to be seismic because the accident is initiated by an earthquake even though the subsequent equipment failures may be dominated by random failure modes. A common example is failure of the switchyard and failures of the emergency diesel generators to start or to run for a sustained time interval.

2.4.2. Level dependent failures

For this class of failure, it may not be desirable or possible to compute explicit responses and/or fragilities. Since the failures are nevertheless felt to be dependent on earthquake level, they cannot be treated simply as random failures because random failure probabilities do not vary with earthquake level.

Based on judgement or experimentation, a probability of failure is developed for these events for each of the earthquake levels which are being considered. When the quantification of cut sets is being carried out, which is done level by level, the appropriate probability for each earthquake level is used in the evaluation process. Relay chatter is an example of the type of failure which may be included here. Operator error might also be of concern but may be difficult to model and in general has not been included in a seismic PSA due to lack of data.

2.5. UNCERTAINTY ANALYSIS

2.5.1. Sources of variabilities

The estimate of the frequency of core damage produced from a seismic PSA has considerable uncertainty associated with it and, without a measure of the variability, the point estimate itself is almost meaningless. Two distinct sources of variability are recognized as making separate contributions to the overall variability. These two sources are generally termed random variability and uncertainty (modelling error).

Random variability

There is, of course, variation in every physical measurement and therefore even plant specific data used in a seismic PSA involve variability. This variability may arise in part from the stochastic nature of underlying physical processes and in part from the inability to measure precisely the parameters which characterize those processes.

Uncertainty (modelling error)

The uncertainty which enters into the estimates of the frequency of core damage from a seismic PSA as the result of the availability of a number of methods for modelling each step of the overall procedure is referred to as modelling uncertainty. It is distinguished from randomness because it originates with the methods used to model the seismic hazard and the plant response rather than as the result of inherent variability in the physical processes being modelled.

2.5.2. Distinguishing between sources of variability

The individual seismic hazard estimates represent the inherent variability in earthquake size and frequency at the site. If there were no inherent variability in earthquakes at the site, then it would be possible to state just how many earthquakes of a given intensity would occur in any one year. Since earthquakes cannot be predicted with anywhere near this level of certainty, their future occurrence can only be assessed with probabilistic statements exemplified by the distribution in the seismic hazard functions shown in Fig. 1.

The distributions of the seismic hazard function represents a different opinion about the seismic hazard. Since each of the hazard prediction experts had access to the same body of physical data for the site and surrounding region, the variability between the various hazards is due more to differences among the experts' methods for interpreting the physical data than to variability in the physical data.

The distinction between random and modelling variability is, at its root, an artificial one. Ultimately, all uncertainty stems from our lack of knowledge about physical processes. For example, if one had perfect knowledge about all of the factors affecting the seismicity of a site, it would be possible to predict just exactly when and what type and size of earthquake would occur at that site. Unfortunately, we have a very imperfect understanding of the factors involved. One can, however, distinguish to some degree the uncertainty which is practically irreducible from that which is not. The reducible uncertainty is that caused by our imperfect interpretation of the limited data that we have. The distribution of seismic hazard shown in Fig. 1 constitutes an expression of that imperfection.

2.5.3. Presentation of results

It is convenient to treat the two recognized sources of variability separately in the analysis of seismic risk. The contribution from randomness leads to a single point estimate of risk while that arising from modelling uncertainty introduces a dispersion around this point estimate.

The point to be noted here is that seismic PSA, like internal event PSA, is fraught with variabilities, both in the data and in the methodology used to analyse those data. Consequently, an uncertainty analysis is essential to the meaningful interpretation of a seismic PSA.

3. PROBABILISTIC SEISMIC HAZARD ANALYSIS

3.1. INTRODUCTION

The seismic input to be utilized within the scope of a PSA is derived from probabilistic considerations. Although full use is made of all deterministic information that can be made available, the probabilistic seismic hazard analysis (PSHA) has to deal with pronounced randomness and uncertainties. The aim is to develop hazard curves which characterize the seismic exposure of a given site to the so-called primary effects (vibratory ground motion). This analysis is based on historical earthquake reports and instrumental records, as well as the geology of the region, including physical evidence of past seismicity.

The procedure to follow is described in detail in the IAEA Safety Guide 50-SG-S1 (Rev.1) [4]. In the following, only a short overview is given stressing the probabilistic aspects (see Fig. 3).

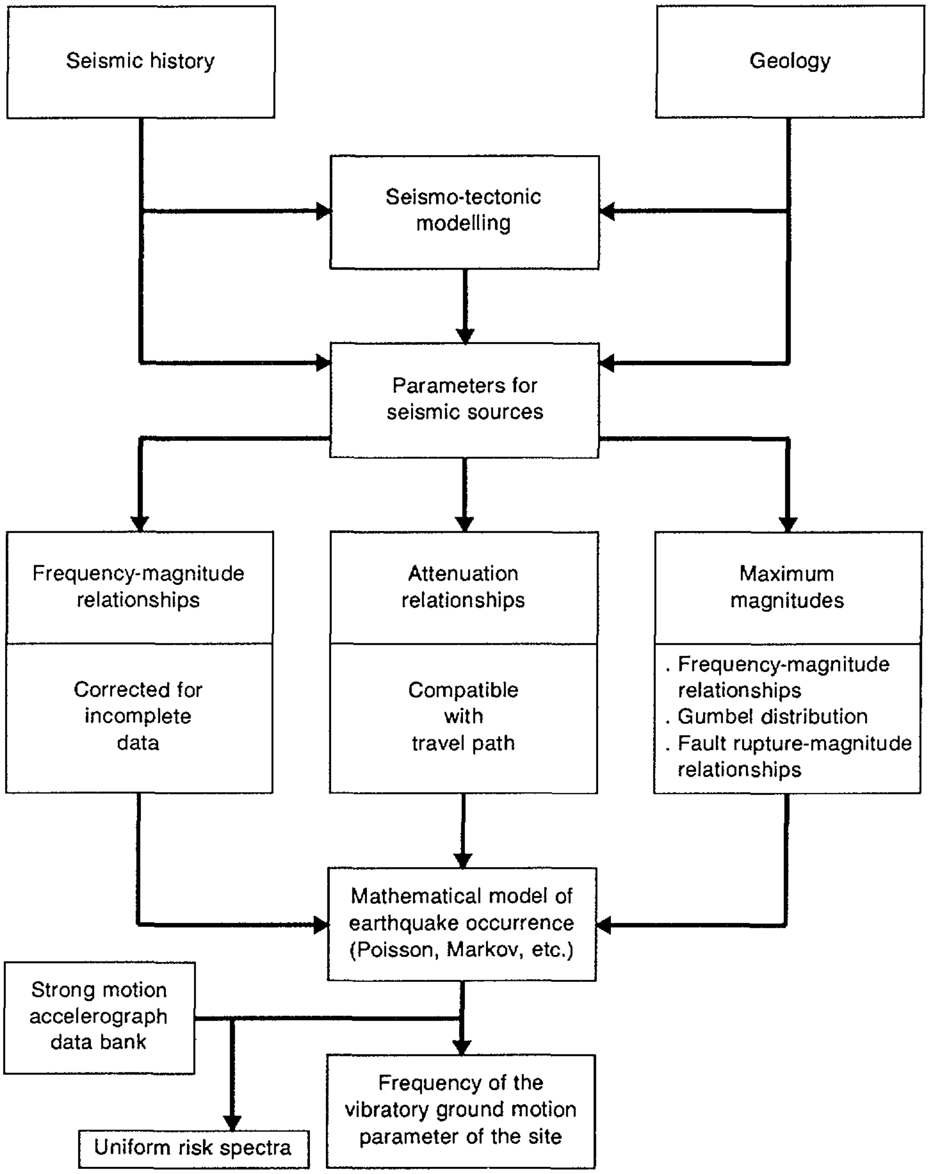


FIG. 3. Flow chart of PSHA.

3.2. HAZARD CURVES

3.2.1. Seismotectonic modelling

Based on all available data to define the seismic history and the tectonics of the region, seismogenic structures and provinces (active crustal volumes) are defined as earthquake (or seismic) sources of uniform seismicity.

Seismic sources can be modelled as volume, area or linear sources with simplified geometry. For each source seismic activity (magnitude–frequency or intensity–frequency relationship, maximum or cut-off magnitude or intensity) is assessed. Attenuation relationships are incorporated in the seismotectonic model.

One or more attenuation relationships are determined on the basis of the strong motion records and isoseismal maps for the region of concern. In case specific data are not enough, attenuation data and relationships for similar geological regions may be used, taken from the literature.

3.2.2. Stochastic model

Earthquake occurrence is represented as a random process. Types used include the Poisson process, Markov chains, renewal processes and cluster processes. The most widely used is the Poisson process which assumes time independence of earthquake occurrence. The arguments against this relate to the independent arrival assumption which is known to be contrary to the physical nature of seismogenic sources. However, the use of non-Poissonian processes is novel and not yet well proven although some applications have been published within the last decade. If the Poisson process is used, the aftershocks are normally removed from the catalogue.

The adequacy of the earthquake catalogue (i.e. length, completeness and reliability) has an important bearing on the accuracy of the stochastic model. This in turn contributes to the uncertainty inherent in the hazard curve.

3.2.3. Vibratory ground motion

Present practice is to characterize the severity of the ground motion by a single parameter such as intensity, acceleration, velocity, etc. It is used as the independent variable of the hazard curve relationship. The choice of the parameter depends mainly on the type and the reliability of available data. For example, in areas where historical seismicity records have been well kept, but instrumental seismicity is scarce, intensity might be a suitable parameter for the analysis. If, on the other hand, the available history of the area is comparatively short, but instrumental coverage is very good, and there are sufficient strong motion records, spectral acceleration could be used.

Although the hazard curve is defined in terms of a single parameter, it is essential that other parameters including duration and spectral content are considered when determining the seismic input. Seismic input to response analysis may be in the form of free field or bedrock time histories, response spectra or other spectral specifications. Time histories can be actual strong motion records selected from a suitable set of records, or artificially generated to conform to a prescribed spectrum.

If soil–structure interaction (SSI) is to be included in the response analysis, it will also be necessary to obtain data for the dynamical properties of the soils, such as shear wave velocity, shear modulus and so on. These parameters may have strong dependency on depth below its foundation, stratification, if any, and strain and these factors should be taken into account. Strain dependency is in most cases a function of the earthquake level.

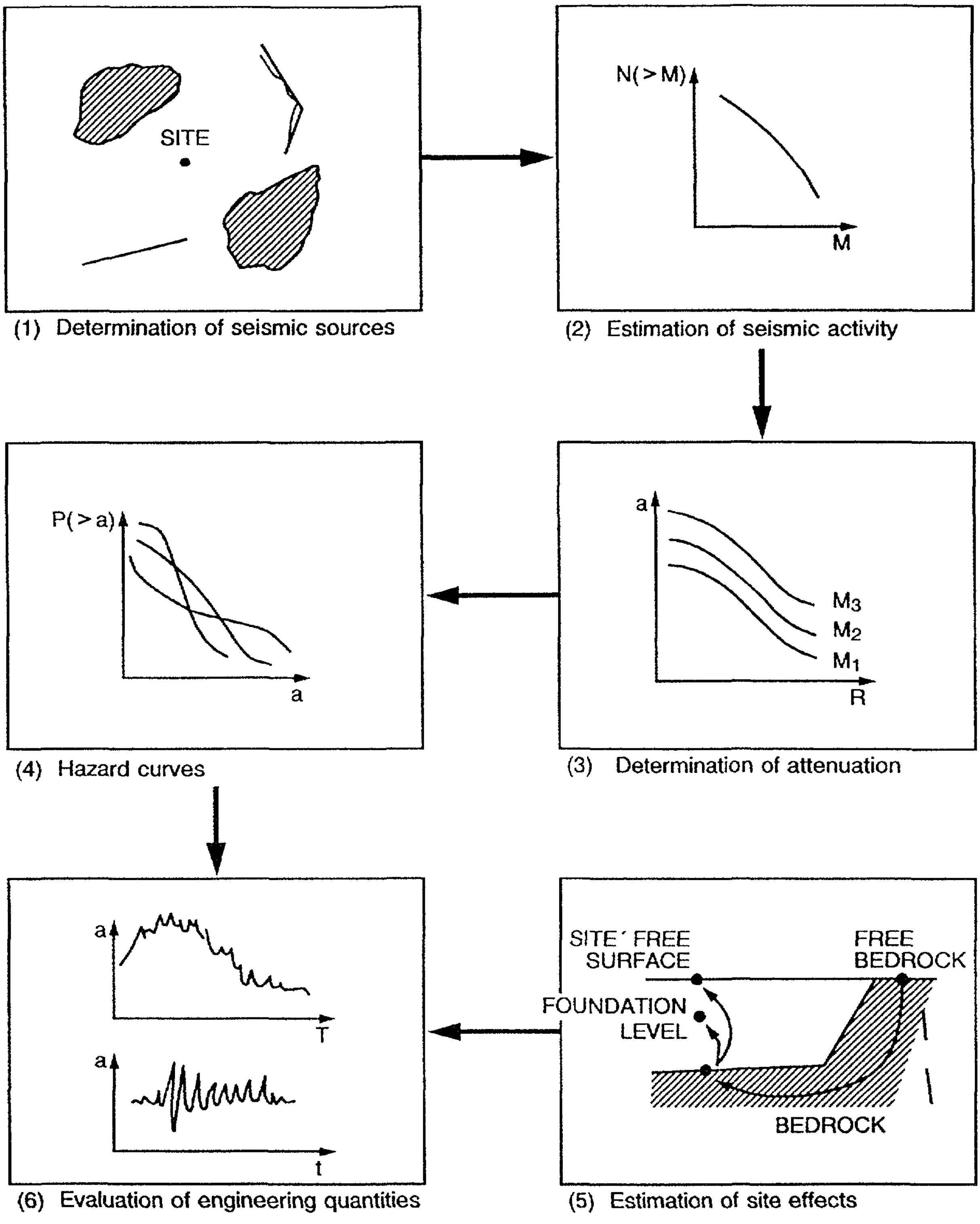


FIG. 4. Analytical task flow of PSHA.

Input parameters required for the calculation are:

- geometrical parameters (coordinates, depth, dips, etc.) of seismic sources;
- parameters of magnitude (or intensity) - frequency relationships (e.g. a and b in Gutenberg–Richter equation);
- maximum or cut-off magnitude (or intensity);
- attenuation relationships;
- uncertainty of the above quantities (given e.g. by two or more alternative values of parameters, and two or more alternative equations, together with corresponding probabilities).

A parameter sensitivity study will be helpful in the determination and choice of the alternatives.

By means of a computer programme, the cumulative annual frequency of exceedance of a given ground motion level in terms of peak ground acceleration (PGA), velocity or intensity on hypothetical outcrop or free field is calculated for combinations of alternative values of parameters.

The result is a number of hazard curves which in the following step are aggregated into a few curves (e.g. five), preserving the information related to uncertainty.

An analytical task flow of PSHA is given in Fig. 4.

3.2.4. Uncertainty

The main contributors to the uncertainty of the hazard curve are:

- the boundaries of seismogenic structures and provinces;
- the geometrical parameters of seismic sources;
- the specification of the seismic activity of the sources;
- the choice of attenuation relationships;
- the choice of the stochastic model.

An important source of uncertainty is the calculation of magnitude from intensity and transformation into acceleration or other vibratory ground motion parameter.

3.3. ENGINEERING QUANTITIES

Any hazard curve must ultimately be translated into engineering quantities. What is needed by the structural engineer is either a response spectrum or a set of reasonably independent time histories at explicitly defined locations, usually at the foundation level and the free field.

The main problem that arises is that hazard curves are defined in terms of a single parameter (PGA or intensity). Thus, information like duration of strong motion or frequency content have to be added. This can be done based on experience with similar tectonic and geological situations; inevitably, additional uncertainties are introduced that have to be estimated.

The variability in response spectra is particularly pronounced if different source mechanisms and focal distances have to be considered. A major earthquake at a large distance from a given site may result in the same intensity as a close moderate one. However, the duration and the long period accelerations will be considerably larger for the major earthquake, whereas the high frequency content of the moderate one will be more important.

Hazard curves usually give the probability of exceedance of PGA or intensity in the free field or a hypothetical outcrop. This means that the influence of the local geological situation on the ground motion, generally known as 'site effect', is not yet explicitly included.

Soft surface layers, marked lateral discontinuities in the immediate vicinity or accentuated changes in topography can result in considerable amplification or alteration of the ground motion. These effects have to be accounted for by a calculation of the dynamic response of the site's geological formation to incoming earthquake waves. The results will be affected by random variability due to the random characteristics of the arriving seismic waves. Uncertainties arise from both the lack of knowledge of the local geology (geometry, soil parameters) as well as from the use of an inevitably simplified model.

4. SECONDARY SEISMIC EFFECTS

4.1. INTRODUCTION

PSAs including seismic loads have been generally limited to the direct influence of the vibratory ground motion to different safety related items in the nuclear power plant, and the resulting influence of these to the annual frequency of core damage which is the major content of the subsequent sections. The concern for seismic loads in PSAs is their potential to initiate common cause failures or/and dependent failures.

The purpose of this section is to give some recommendations for identifying secondary effects which may appear as consequences of earthquakes.

Secondary seismic effects are often dismissed deterministically either as not being credible or because it is cumbersome to include these into the already complex framework of a PSA. This section will address some of the issues related to secondary seismic effects. Recent field observations have demonstrated the importance of secondary seismic effects in the damage distribution and subsequent impairment of function of plant systems. The most common secondary effects are what is termed as systems interactions. Other types of secondary effects include fire following an earthquake, inadvertent activation of fire protection systems and plant specific secondary effects such as flooding due to dam failures.

4.2. SYSTEMS INTERACTIONS

The inclusion of many types of systems interactions is common practice in current seismic PSAs. Systems interactions result when seismic failure of a non-safety system or component affects the performance of a safety related component or system. These interactions may be spatial or systematic. Spatial interaction includes falling, hammering, spray and internal flooding. A very common spatial interaction is failure of unreinforced masonry walls which may impact essential equipment.

Systematic interactions include such scenarios as failure of a non-safety heat exchanger which breaks the closed loop component cooling water systems resulting in loss of CCW. Modern plant designs preclude most potential systems interactions but they are quite common in earlier NPPs.

4.3. SEISMIC FIRE INTERACTIONS

This is currently a subject of interest and has been studied by USNRC contractors [15]. Fire protection systems are typically not designed for earthquakes and are often damaged in earthquakes. If a seismic event initiates a fire and the fire protection system is unavailable, the plant may be severely damaged. Fire protection system failures are often examined as a spatial system interaction source of spray or falling but in the event of a fire, the unavailability becomes even more important. Another current issue with fire protections systems is activation resulting from an earthquake. This is often a source of water damage to equipment in non-nuclear facilities and a potential threat in nuclear facilities.

Other consequences which could result from inadvertent activation of fire protection systems are uninhabitability of the control room and shutdown of emergency diesel generators upon a fire signal. These fire and fire protection systems issues are often difficult to identify and model.

4.4. PATHWAYS FOR OTHER SEISMIC SECONDARY EFFECTS ON NUCLEAR FACILITIES

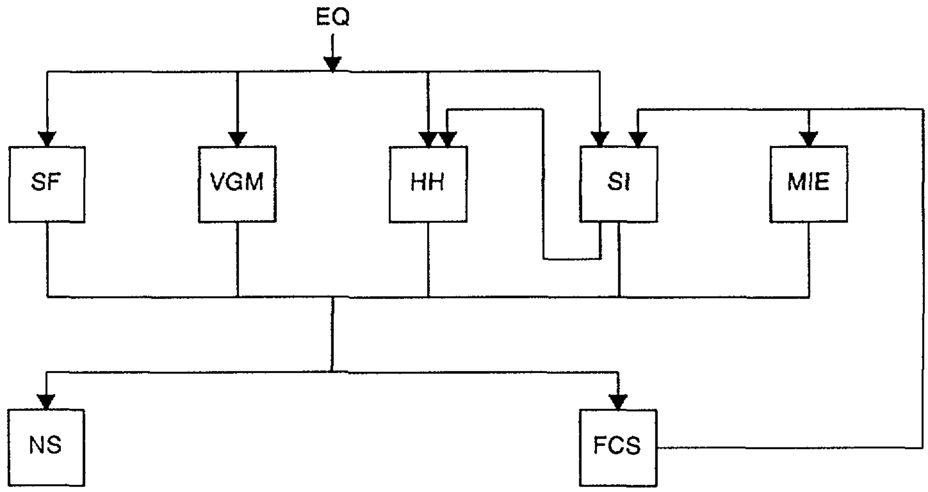
Seismic events may initiate an accident in various ways because of the number of intermediate effects which finally cause failure or loss of function of NPP components.

Figure 5 gives a condensed version of these possible pathways. Various elements may be located in each block; see the legend of Fig. 5.

It should be noted that many of the seismically induced secondary events may not constitute an immediate input into a 'seismic' PSA. For example consider the pathway EQ → VGM → FCS → MIE → NS, i.e. vibratory ground motion due to earthquake causes the failure of a dam and consequently the nuclear power plant site is flooded. Although the original trigger is seismic, the eventual effect to the nuclear power plant site is in the form of a flood and the additional due risk can be accounted for within the framework of a 'flood' PSA.

It may be necessary to produce a new seismic hazard curve for some secondary effects if they take place some distance away from the site. For example, the loss of off-site power might be initiated at a switching station remote from the NPP. The time aspect of secondary effects should not be forgotten; the most obvious example is impact due to aftershocks. Some of these possible pathways are discussed in more detail in the following sections.

In practice, most of the pathways which can be generated using Fig. 5 would have such a low probability of occurrence that these scenarios would be considered 'impossible' and taken out of further consideration. However, in performing a PSA it may be discovered that secondary effects have not been excluded when siting the plant.



EQ: Earthquake
VGM: Vibratory ground motion
SF: Surface faulting
HH: Hydrological hazards; Tsunami, Seiche
SI: Soil instabilities: Liquefaction, slope instability, subsidence, collapse
MIE: Man-induced events: Flood, fire, drifting cloud, explosion
FCS: Failure of conventional structures (dams, pipelines, storage facilities for explosive material)
NS: Nuclear structures (containment, reactor building, buildings neighbouring reactor, power lines, switchyard, water intake structures, ultimate heat sink, diesel generator building, fuel storage facility).

Hazards considered for NS: Fire, explosion, flood, settlement, uplift, loss of bearing capacity, overturning, tilting, sliding, foundation rupture, impact due to collapse of other structures, drawdown.

FIG. 5. Block diagram for pathways.

Some of the reasons are as follows:

- New (i.e. after design/construction of the nuclear power plant) information and evidence may be revealed indicating an increase in a particular natural hazard. An example of this is the discovery of a fault in the site vicinity.

- New general information and evidence may lead to a different interpretation of seismic hazard at the site. This may be due to more data in terms of additional epicentres in the site region or strong motion records obtained elsewhere but which may have applicability to the site. This could lead to a higher design acceleration or higher design spectral ordinates.

- The above items also apply to conventional structures in the site vicinity whose failures may adversely affect the nuclear power plant safety. Furthermore, safe siting criteria for these structures are generally less strict and the design basis vibratory ground motion for these is lower in comparison with nuclear power plants.

- There may also have been construction of new conventional structures in the site vicinity whose failure may adversely affect nuclear power plant safety, such as a pipeline, petrochemical facility or a dam.

- Finally, new understanding and interpretation of nuclear safety concepts may also initiate considerations for earthquake levels beyond design basis.

4.5. EXAMPLE SCENARIOS FOR OTHER SEISMIC SECONDARY EFFECTS

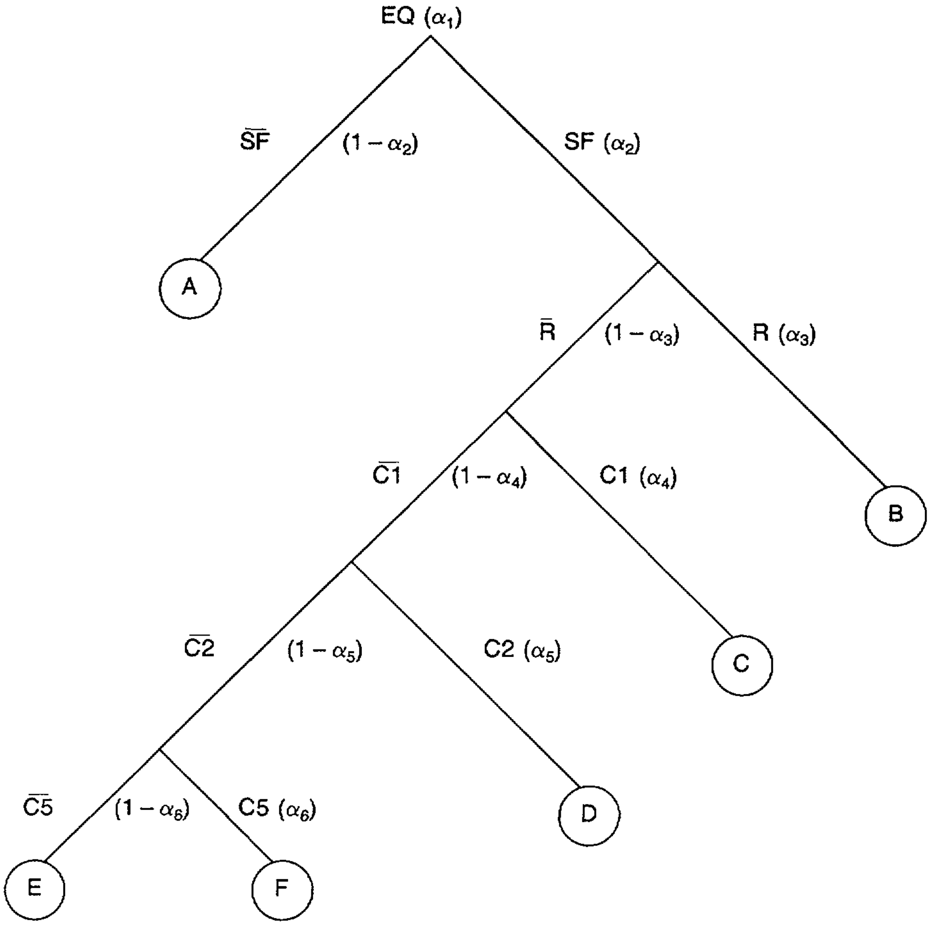
Ten example scenarios have been compiled in Table I. This is not an exhaustive list and the analyst must insure that all possible sequences for any specific sites are covered.

Each of the sequences could be developed into an event tree. This has been done in Fig. 6 as an example for sequence number 3 (surface faulting).

Although all the pathways are initiated by seismic events, this does not mean that the risk from all of the pathways is in all cases appropriately included in the estimation of the seismic risk increment for the facility; nor does it mean that a detailed analysis of a seismically initiated pathway need necessarily be performed as part of the seismic PSA. It would be perhaps more appropriate for several of the indirect pathways to be evaluated as part of other external event analyses.

TABLE I. EXAMPLES OF SCENARIOS FOR OTHER SEISMIC SECONDARY EFFECTS

SECONDARY SEISMIC EFFECT	CONSEQUENTIAL EFFECTS
1A. Soil liquefaction under NPP structures	Severe structural damage leading to core damage
1B. Soil liquefaction not under NPP structures	Loss of off-site power Loss of cooling water Worsened access
2. Slope instability, subsidence or ground collapse	Direct damage to NPP structures Blockage of river causing flood Damage to water retaining structures
3. Surface faulting	Direct damage to NPP structures Loss of ultimate heat sink
4. Structural damage to power lines, switchyards, etc.	Loss of off-site power
5. Tsunami, seiche or dam failure	Flooding of safety related items Damage to power lines, etc. — loss of off-site power Damage to NPP structures
6. Damage to pipelines or hazardous storage	Fire Explosion Toxic effects Missiles



α_1: annual probability of an earthquake of $M \geq M^*$ in the site vicinity
α_2: probability of surface faulting SF
α_3: probability of SF intersecting the reactor building R
α_4: probability of SF intersecting other category 1 structures C1
α_5: probability of SF intersecting category 2 structures C2
α_6: probability of SF intersecting conventional structures C5

FIG. 6. Event tree for surface faulting.

For example, if a seismic PSA is being performed for a nuclear facility which is in the flood plain of a dam, then it is almost certainly the case that a flood analysis for the facility is being performed as well. This flood analysis will consider non-seismically induced modes of flooding arising from such phenomena as random failure of the dam, dam failure due to severe weather, improper operation of the dam, etc. It would be entirely possible to increase the frequency of the gross failure of the dam by the amount due to seismic events and, in this way, account for the risk to the nuclear facility from seismically induced dam failure. In fact, this seems to be a much more desirable approach since the analysis of plant damage will be similar for seismically induced failure of the dam as for non-seismically induced modes of failure.

The word similar is used because there is one important distinction between seismically induced flooding at a nuclear facility site and other flood events at the site, and that is the fact that the facility will also very likely be subjected to some seismic loading as well as flood damage, depending on the epicentral location of the earthquake, the proximity of the dam to the facility, the surrounding geology, etc. This problem of having to consider two simultaneous or closely sequenced external events impacting a facility is a very difficult one. It has not been extensively addressed and may, in fact, defy adequate treatment. One may wish, instead, to calculate the probability of sustaining two severe external events simultaneously and assume some maximum damage state to result. The frequency of such a damage state arising from this simultaneous occurrence would then be simply equal to the expected frequency of simultaneous occurrence of the two external events. This frequency may be well within whatever is deemded to be an acceptable level of risk for the facility. If so, then no further analysis would be necessary. If not, then it might be necessary to reevaluate the situation to reduce conservatism and, possibly, resort to a specialized analysis which deals with the effects of two simultaneous external events.

4.6. RISK CATEGORIES

Once all of the consequential effects have been considered and event trees prepared, the risks can be categorized and ranked in order of probability. Then each one can be considered for inclusion in the PSA, or eliminated if the risk is acceptably small.

5. FRAGILITY

5.1. INTRODUCTION

The derivation of appropriate fragility curves for structures, equipment and piping systems can require substantial effort. In this section, simplified as well as more sophisticated methods will be discussed that are used in the derivation of fragility curves.

Various methods may be used for the derivation of fragility curves. Some are well established analytical procedures while others are very subjective estimates. In the "PRA Procedures Guide" [6], these methods are briefly described. Other informative references on fragility derivation are [9], [10] and [11]. These papers describe the methods of developing fragility curves that have been used in over 30 seismic PRAs to date. Reference [8] describes the more sophisticated methodology developed in the USNRC sponsored Seismic Safety Margins Research Program. The level of effort for development of fragility curves for structures and equipment is variable. The most important and usually the most seismic resistant building is the reactor building, and the number of buildings which require fragility analysis is limited, therefore more detailed analytical methods are generally applied to structures. The number of safety related equipments and piping is large, and it is not practical to apply detailed methods in most cases. In this section, details of the techniques will not be described, only their main features will be presented along with guidance on which methods are usually appropriate for generic classes of equipment.

A flow chart of the development of fragility curves for structures and equipment is shown in Fig. 7. The development of the fragility curves for structures and equipment encompasses the dynamic response to the ground input motion and the capacity relative to the load. This section will focus on response from the free field ground motions up through the equipment, dominant failure modes and capacity. A discussion of the use of generic fragility descriptions, expert judgement and earthquake experience to focus the detailed analytical activity on the most important and most vulnerable structures and equipment is also presented.

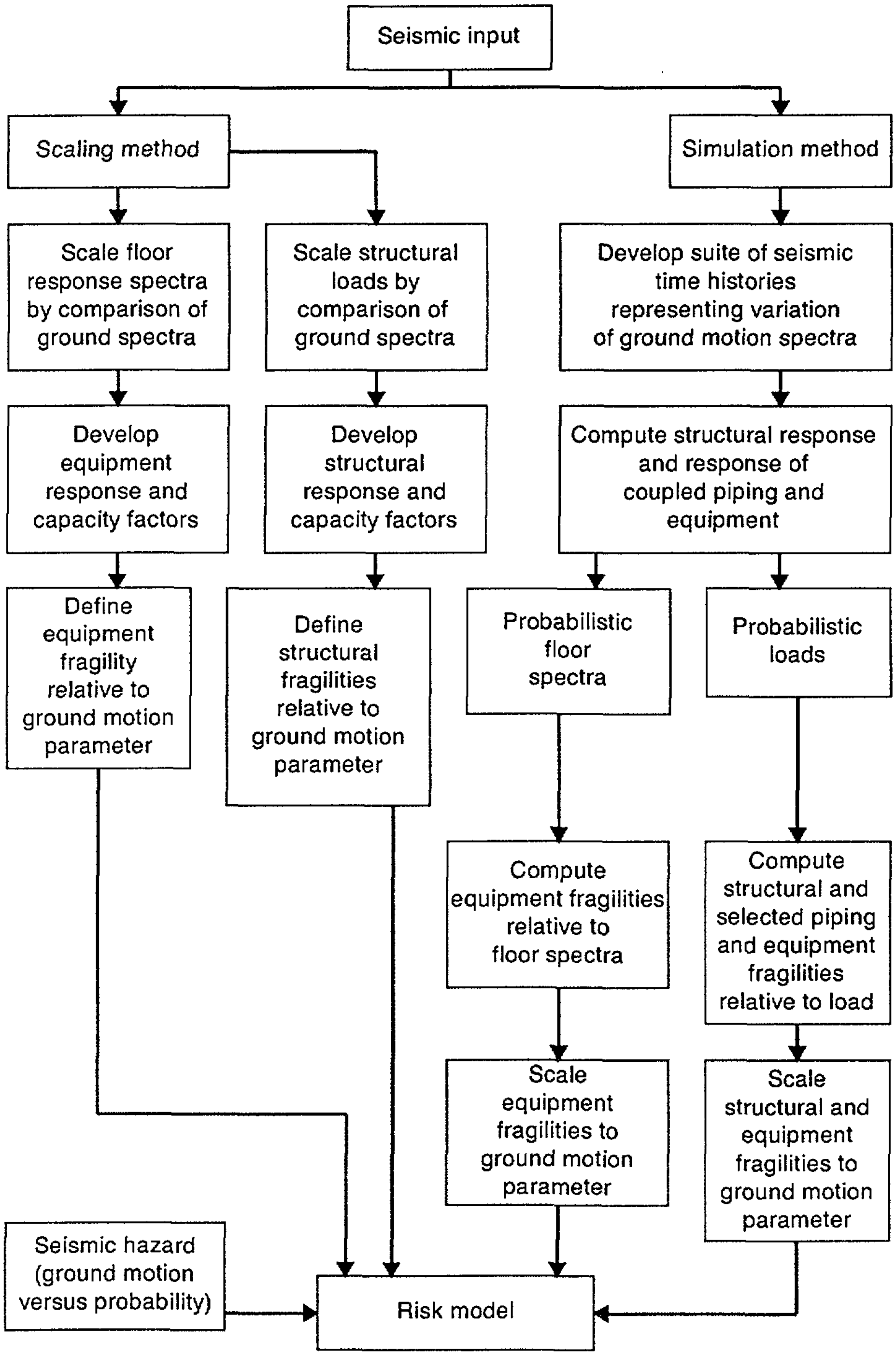


FIG. 7. Flow chart for seismic fragility development.

5.2. FRAGILITY DESCRIPTION

Seismic fragility of a component or system is defined by a curve that gives conditional probability of failure for a given value of a seismic input motion parameter. The input motion parameter may be defined at the structure/component interface (component support location) or at the base of the supporting structure (ground level) depending upon the detail employed in the PSA to define response at the component level.

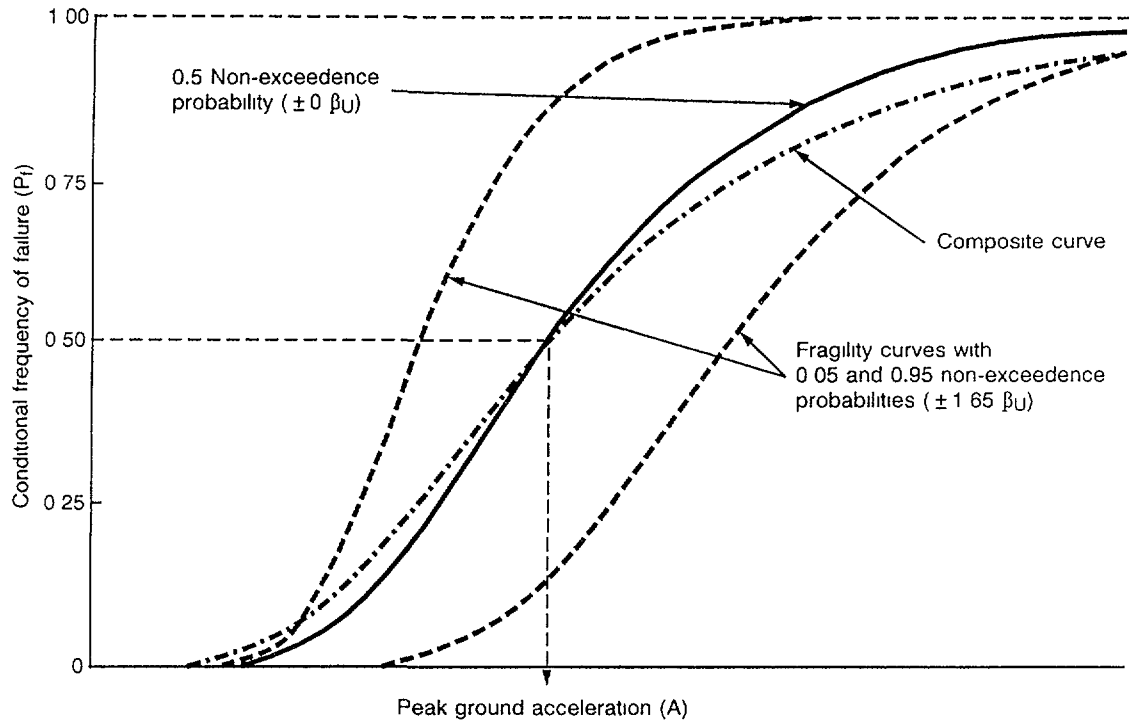


FIG. 8 Fragility curve representations.

Figure 8 shows a typical representation of fragility curves for a component. Because there are many sources of variability in developing a component fragility description, a single fragility curve is not appropriate, although a single composite curve is sometimes used, as also shown in Fig. 8.

The entire fragility family for a component corresponding to a particular failure mode can be expressed in terms of the best estimate of a median input motion parameter and two random variables. For convenience, we will use acceleration, A, as the input motion parameter. Thus, the acceleration capacity A is given by:

$$A = A_m \epsilon_R \epsilon_U \quad (1)$$

in which A_m is the median capacity and ϵ_R and ϵ_U are random variables with unit medians. They represent, respectively, inherent randomness about the median and uncertainty in the median value. Sources of randomness and uncertainty are discussed in Section 5.3. Randomness is fundamental to the phenomenon being represented. Regardless of the amount of detail expended on fragility development there remains some random variability that cannot be isolated. The slope of the fragility curves represent the inherent randomness. Uncertainty represents the analyst's incomplete knowledge. With further detailed study, testing and analysis, the uncertainty could theoretically be eliminated. The positions of the family of fragility curves represents the analyst's uncertainty as to where the true fragility curve lies. In this model, we typically assume that both ϵ_R and ϵ_U are log-normally distributed with logarithmic standard deviations, β_R and β_U, respectively. Material strengths tend to be log-normally distributed. Also, the central limit theorem states that the products and quotients of random variables tend to be lognormally distributed regardless of the distribution of individual variables. The formulation for fragility given by Eq. (1) and the assumption of log-normal distributions enable easy development of the family of fragility curves while reasonably representing their randomness and uncertainty.

In Fig. 8 a median (50% non-exceedance probability) and the 90% confidence bounds (95% and 5% non-exceedance probabilities) are shown. Non-exceedance probability is defined in this case as the probability at which the true fragility curve will lie above the position stated.

A best estimate fragility curve may be defined using a composite of the randomness and uncertainty variabilities. The composite variability β_c is defined as:

$$\beta_c = (\beta^2{}_R + \beta^2{}_U)^{1/2} \tag{2}$$

The composite fragility curve may be used for computing point estimates whereas the complete family of fragility curves is used if the distribution on core damage frequency is computed.

5.3. SOURCES OF RANDOMNESS AND UNCERTAINTY

Randomness is typically considered to be a source of variability in response or strength that is a direct function of the earthquake input motion. No matter how much the earthquake is studied, the exact time history of a future event cannot be predicted. Uncertainty arises from modelling of the structural or equipment response or its capacity. Each variable that contributes to the response or strength of a component has associated with it some random variability or uncertainty which results in randomness and uncertainty in the response and strength. Following is some general guidance on the variables of most importance and the dominant source of variability.

Response variable	Dominant variability categorization
Peak to peak variation in input motion	Randomness
Phasing of earthquake components	Randomness
Phasing of modal responses	Randomness
Vertical/horizontal accelerations ratio	Randomness
Soil stiffness	Uncertainty
Soil damping	Uncertainty + randomness
Structural stiffness	Uncertainty
Structural damping	Uncertainty + randomness
Non-linearities	Uncertainty
Soil–structure interaction	Uncertainty

Capacity variable	Dominant variability categorizations
Material strength (force-deformation)	Uncertainty
Ductility	Uncertainty + randomness
Instability point (elastic or inelastic)	Uncertainty
Load combination (normal plus seismic)	Uncertainty

The fragility analyst may often assign both randomness and uncertainty to some parameters whose variability is affected by both the input motion time history and the lack of knowledge in the modelling of the phenomenon. Damping and ductility are examples where the variability is usually defined as a combination of randomness and uncertainty.

5.4. FAILURE MODES

It is important in developing a fragility description that the risk modeller understands the failure mode associated with the fragility curve and the consequences of the failure mode. Structural failure modes may have different meaning. In general, the global structural failure mode is a degree of distress in the structure beyond which the analyst cannot be assured that equipment, piping, etc. can remain properly anchored. If a shear wall is severely cracked, pipe supports in the wall may pull out and no longer support the piping. This degree of distress is very judgemental and must reflect the location of the structural distress and the attached equipment. In some cases a structural failure mode may mean complete collapse such as in the case of an interior non-structural masonry wall that may collapse and impact essential equipment. This is a system interaction where a non-essential structural component fails and affects the function of an essential component.

Equipment failure modes may consist of failure to function or failure of the pressure boundary. The consequences of each type of failure mode may have a totally different effect upon the plant system response.

It is very important that a close interface be maintained between the fragility analyst and the systems analyst so that the effects of the failure mode are properly modelled.

5.5. WALKDOWN AND SCREENING

It has been recognized that a detailed walkdown of a NPP is an effective way of uncovering seismic vulnerabilities in essential equipment and in identifying seismic spatial systems interactions. The walkdown performed as part of a seismic PSA can be a very effective means of determining which components will require a detailed fragility assessment, which may be represented by generic fragility curves and which may be screened out. The walkdowns should be conducted by experienced engineers who can make engineering judgement decisions on the relative seismic capacity of equipment.

In general, rigid equipment that is well anchored is not vulnerable to seismic events. Such equipment are pumps, valves, compressors, diesel generators, chillers and many tanks and heat exchangers. Sometimes though, there may be a potentially vulnerable ancillary item that governs the failure mode of a rigid equipment. Examples are the diesel fuel day tank for a diesel generator, impact of a valve operator in close proximity to a structural element, or vertical air receiver tanks for a compressor system.

Other items which have been demonstrated by earthquake experience to be seismically rugged are piping, cable raceways and electrical conduit. These distributive systems are not vulnerable to inertial loading as long as the support systems do not fail in a brittle manner. The major concern with distributive systems is seismic anchor movement due to differential building motions or at a flexible anchor point.

Electrical and control cabinets tend to be the most troublesome since their functional capacity cannot be ranked by visual inspection. The structural features though can often be assessed by judgement during the walkdown.

Most of the emphasis in a walkdown is to judgementally rank the structural capacity of essential equipment based upon its anchorage and its ancillary equipment, identify potential systems interactions such as failure of masonry walls in the near vicinity of essential equipment and to look for other seismic issues that could affect the function of essential equipment. Guidance on conducting walkdowns is provided in Refs [16] and [17].

5.6. FRAGILITY CURVE DEVELOPMENT

Two methods have been used to develop fragility curves for use in seismic PSAs, the simulation method and the scaling method. Both methods have been described in the US Nuclear Regulatory Commission Procedures Guide [6]. Reference [8] describes the simulation method in detail and References [9] and [10] describe the scaling method. Reference [11] provides examples of each method. In either method, the hazard definition and the system models are the same. The difference lies in the details of development of the fragility description for structures and components. In either case, though, the final fragility description is a conditional probability of failure relative to a hazard parameter defined at ground.

Figure 7 is a flow chart of each of the methods. In the scaling method, existing structural analyses are utilized and the fragility curves are referenced to a ground motion parameter. In the simulation method, new structural and equipment analyses are conducted and the fragility curves are referenced to the component/structure interface response parameter (local response) but must ultimately be scaled to reference a ground motion input parameter.

5.6.1. Scaling method

In the scaling method an overall factor of safety is developed which represents the best estimate of the actual response and capacity as opposed to the design response and capacity. In structures, the overall safety factor is the product of a capacity factor and a response factor.

$$F = F_C F_{RS} \tag{3}$$

The median (best estimate) of the fragility relative to the hazard, which in this example is defined as peak ground acceleration, is:

$$A_m = F_m(A_{SSE}) \tag{4}$$

where F_m is the median value of the safety factor and A_{SSE} is the safe shutdown earthquake upon which the design is based.

The capacity factor F_C and response factor F_{RS} are each comprised of the product of the factors of conservatism or unconservatism in each important variable contributing to the response and capacity. Each of these important variables has randomness and/or uncertainty which result in randomness and uncertainty in the final factor, F_m, and the median fragility, A_m.

For equipment mounted in a structure, the overall factor of safety is the product of three factors.

$$F = F_{RS} F_{RE} F_{CE} \tag{5}$$

where F_{RS} is the structural response factor, F_{RE} is the equipment response factor and F_{CE} is the equipment capacity factor. Each factor is in turn the product of the factors of conservatism or unconservatism in each important variable that contribute to response and capacity. The randomness and uncertainty in each variable is combined to define an overall randomness and uncertainty in the overall factor of safety, thus the median fragility level, A_m. Derivation of these factors is detailed in Refs [9], [10] and [11].

Often, for older NPPs, a new deterministic analysis is conducted of selected structures and the scaling method is used in conjunction with the new deterministic analysis and the ground motion spectra selected for the new deterministic analysis.

Since the original analyses or new deterministic analyses are scaled in lieu of conducting probabilistic analyses by the simulation method, the uncertainty is consequently larger than if the more sophisticated simulation method is used. In this method, the factors are developed relative to the design basis defined as a ground motion spectrum, thus the fragility curve is always defined in terms of the ground motion input.

5.6.2. Simulation method

In the simulation method, multiple response analyses are conducted from the ground up for structures and selected equipment and piping. The equipment and piping models may be coupled or uncoupled from the structural models. Multiple time history analyses are conducted in combination with variations of the important variables contributing to response. The resulting response, whether load in a structural or equipment element, or a floor response spectrum, is defined probabilistically so that factors associated with response are not required. The capacity of the structure or component is then computed and a capacity factor relative to the local input motion parameters is derived. The randomness and uncertainty associated with the capacity is estimated and a fragility curve relative to the local input motion parameter is developed. Since the hazard is defined at the ground, this fragility description must be transferred to the ground by dividing by the ratio of the local response parameter used as a reference for fragility curve development to the ground input parameter used to define the hazard.

The randomness and uncertainty computed for the new response must also be combined with that defined for the capacity to quantify the randomness and uncertainty relative to the ground motion input.

The simulation method is often truncated at the structural response level without propagating to the equipment or piping level. The simulation method to piping and equipment has been used to a great advantage for some higher seismicity sites to improve accuracy of the response prediction and reduce uncertainty. Such detail is usually not warranted for the lower seismicity sites. Use of the simulation method to compute probabilistic spectra and structural loads is, however, quite common, even for lower seismicity sites.

5.6.3. Supplemental methods

Often, generic fragility curves are used directly. Reference [18] contains a compilation of seismic fragilities developed in PSAs of numerous nuclear plants. These may be used when results of the walkdown and a review of the design basis suggest that they are appropriate. Some of these generic fragilities have been used in conjunction with earthquake experience data using Bayesian methods [19]. This proved in some cases to be of benefit in reducing the uncertainty in the generic fragility descriptions.

Fragility descriptions for components qualified by testing may be approximated from the test results but, unfortunately, the true capacity is unknown. It is usually assumed that the test level represents approximately a 95% confidence of less than 5% probability of failure. This has been termed in the literature to be a high confidence of low probability of failure (HCLPF) value. There is usually conservatism in the structural response and in the testing input so that the HCLPF relative to ground is substantially greater than the safe shutdown earthquake which defines the seismic design basis. These conservatisms define the structural response factor and an overtesting factor. If test data are used to develop the fragility curve, the randomness β_R and uncertainty β_U on the capacity must be estimated. They are usually estimated to be of the same magnitude as the β_R and β_U developed for the capacity of other components. The fragility curve is initially defined relative to the floor response of the structure and then transferred to ground by incorporating the structural response factor and its randomness and uncertainty.

The development of fragility curves for most components is quite subjective, but if conducted within the guidelines of this technical document and supporting references, they should be reasonably representative of the seismic capacity of NPP components. Usually, in earlier NPPs, there were some weak links in the structural design or in a few equipment items that dominated seismic risk so that the use of generic fragilities or fragilities estimated from qualification test data did not result in a major risk contributor. For the higher seismic zone sites though, more detailed study is often required to more accurately define the median capacity and to reduce the uncertainty. It is usually most efficient to use the information readily available to make scoping estimates of the fragilities for items included in the risk models. Then after a first iteration on the seismic risk computation, those structures and components which are the dominant contributors are identified. Then it may be appropriate to conduct more detailed evaluations of certain fragility descriptions. Sensitivity studies may also be conducted to determine the benefit of further fragility evaluations. Insights that are useful for conducting detailed analysis are presented in the next section.

In some instances, it has been cost beneficial to conduct non-linear analysis of selected structures or equipment in order to improve the fragility description. In other cases, it is easier to perform simple backfits to increase the capacity. The risk model can be used to optimize the expenditures for improvement of structural and equipment fragilities.

5.7. BUILDINGS AND CONTAINMENT STRUCTURE

General experience and suggested methods are described in the following sections.

5.7.1. Reinforced concrete and pre-stressed or post-stressed concrete buildings

Reinforced concrete (RC) and pre-stressed or post-stressed concrete (PC) buildings are very common structures to be analysed and lumped mass-spring models are often used in the design process. To know which part of the section is affected by a spring failure, i.e. shear wall, beam-column connection, part of a particular slab, etc., additional information is needed for the expected value of the ductility factor or some other parameter indicating the degree of the elastoplastic state of the section. Even when it is not necessary to evaluate the fragility curve of a particular section, we may need to evaluate the back-bone curve in case the load-bearing element of the section changes by redistribution of the load within the section.

5.7.2. Hybrid building: Reinforced concrete and pre-stressed or post-stressed concrete buildings with steel frame structure

For a BWR reactor building, a hybrid building structure is often used. For example, a reactor building up to the operation floor is a reinforced concrete structure, but above this level a steel frame structure is often employed. In this case, plastic hinges may form on some sections of the steel frame prior to distress in the concrete structure. In defining failure of the structure, one must estimate the limit of inelastic deformation that is tolerable in each section before affecting safety related items.

5.7.3. Masonry structure

Masonry structures are frequently used as interior walls for auxiliary buildings. Most vibratory energy should be absorbed by local failure of the wall elements. The failure of load bearing masonry walls may cause catastrophic failure to the structure. Non load bearing walls are a threat to adjacent equipment. Spalling or collapse of a masonry wall may cause a fragment missile or a crashing of the wall against essential components. Non-linear analyses are often used to demonstrate the capacity of unreinforced masonry walls.

5.7.4. Prefabricated building

A structure composed of pre-cast concrete elements is sometimes used for auxiliary buildings. If a roof slab or a slab floor element is heavy, there is a possibility that it will fall on components in the building due to local failure of joints within walls or columns. Such designs and failures have been observed in conventional steam power plants and other industrial buildings. For the fragility assessment, this failure mode may be an abrupt and complete failure or collapse.

5.7.5. Steel containment structure

Complete collapse of a cylindrical steel shell by an external load is very unlikely, except for very thin shell structures like a hopper for grain or a tall stack without supporting structure. Therefore the complete collapse of a steel containment structure under seismic conditions is not considered credible. The definition of its failure should then be the loss of integrity. In most critical cases of failure, a penetration or an equipment hatch may be separated from its shell, or a large crack on the shell may be created by local buckling. This constitutes a complete loss of integrity.

5.7.6. Pre-stressed concrete containment structure

This has a higher probability of collapse than a steel containment, if the pre-stressed cable is broken. However, there is almost no possibility of breakage of pre-stressing cables. The definition of the failure of pre-stressed concrete pressure vessel is still not clear. Its loss of integrity is not easy to estimate, because the behaviour of its liner plate in the ultimate condition is not well known.

The calculation of the fragility curve for both types of structures, i.e. a steel containment or a PC containment, is not difficult using either a lumped mass-spring model or a finite element model, if the criteria for failure is well defined. The most difficult part is determining the failure mode and the threshold of failure.

5.7.7. Foundation

Failure of foundations of safety related structures, as well as safety related mechanical components like cooling water intake pumps is important in sites underlain by unstable soils or rock.

The failure of a foundation may be defined by differential settlement under earthquake conditions. A cracking or breakage separation of foundation mat is rare for large structures, but is possible for a machine foundation.

5.7.8. Pools and ponds

Pools or ponds in nuclear power plants are of two types, the spent fuel pool in the reactor building and above ground, the other separate from massive buildings and situated at the ground level. The failure of a spent fuel pool situated in the reactor building may be defined by the failure of the reactor building itself, by failure of the pool structure or by failure of connecting piping.

Even if it has a metallic liner plate, a pool may lose its integrity by cracking. The behaviour of the pool or pond, on or under the ground level, is similar to that of a foundation. But its wall is thinner than the mat, and has the possibility of cracking or failing. This is also governed by soil conditions, including the possibility of liquefaction.

Overflow caused by sloshing may cause failures of some items on or near the edge of the pool. Overflow itself may be only a partial functional failure.

5.7.9. Appendage structures

'Appendage structure' means a structure which may be expressed by a small mass-spring system attached to the building, like part of a stack, a small elevated tank, etc. Such structures may fail easily by a whipping effect in relation to the building itself. A steel frame structure located on the upper part of the reactor building as described in Section 5.7.2 is one such example.

The way of developing the fragility curve is the same as for other structures or for equipment mounted in a structure.

5.8. COMPONENTS

Most component fragilities are derived by the scaling method using qualification information (analysis or test data), or are represented by generic fragility curves. The walkdown is an effective tool to decide which method is appropriate.

5.8.1. Vessels

The focus is usually on the anchorage and supports. Nozzles and tank wall failure are usually not credible.

5.8.2. Reactor vessels

Two main types of reactor vessels are considered. One is skirt supported for BWR, and the other is supported by piping nozzles for PWR. The reactor vessel for BWR can be treated as an ordinary vessel as described in Section 5.8.1. It also has supports on the upper part of the reactor vessel. For this, also refer to Section 5.8.5, Steam Generator and Section 5.9.7, Supporting Structure and Devices.

Failure of the reactor vessel for a PWR would be very rare under an ordinary seismic event. For its fragility curve, a fracture mechanics approach may be used. An extensive study of PWR primary coolant system supports was conducted for the USNRC [20–22] and it was determined that the probability of support failure was so low that it is no longer considered in seismic PSAs of US plants. Some PSAs of BWRs have revealed a high capacity of the reactor vessel supports but not so high that the failure mode can be ruled out. Reference [23] revealed a high capacity for an example US BWR.

In very high seismicity sites, uplifting of nozzle supported vessels may require consideration.

5.8.3. Core internals and fuel

Failure is not well defined, especially for the fuel. Failure to insert the control rods is the greatest concern for the core internals. Situations which cause the state of a core assembly to prevent normal water circulation for cooling the core partially or as a whole is usually not considered as a credible failure mode. Rupture of a small number of fuel pins is also not modelled as a failure.

The core is highly non-linear and complex analyses are conducted in the design stage to verify core stability under seismic and LOCA loading. Generally, the LOCA loading governs and there is substantial margin for seismic loading.

5.8.4. Heat exchangers

There are various types of heat exchangers in NPPs. A fragility curve of an upright heat exchanger is similar to that of a vertical type vessel. Tube bundles are usually well supported and the emphasis is on the supports and anchorage.

Horizontal cylindrical type heat exchangers are also frequently used. They are rather rigid, and usually fixed to their foundation by saddles. The failure analysis should be performed for the saddle and anchorage. The tube bundles are usually well supported to protect against flow induced vibration and are generally not a seismic issue.

5.8.5. Steam generator and pressurizer of PWR

In principle, there is not much difference from ordinary vertical tanks or heat exchangers; however, they are supported by devices fixed on RC shielding walls, and also supported by legs with hinges. Therefore, we must assume several failure modes; RC wall failure, leg failure, vessel failure, etc., as well as tube bundle failure. Vessel failure at the connection to supporting devices is similar to that for nozzles, and a limit state or a fracture mechanics approach may be employed. Undetectable flaws in this part are a significant issue for fracture mechanics analysis. PWR steam generator supports for US plants were examined in Refs [20–22] and it was found that the probability of seismic induced failure was very low. For other designs, especially in high seismic areas, such failure modes cannot be screened out.

Pressurizers are treated as other vertical vessels. The emphasis is on supports and anchorage. The role of the electric heaters in the pressurizer is typically not modelled for seismic PSA. If required, the fragility curve may be obtained as described for tube bundles.

5.8.6. Above ground storage and thin walled vessels

These storage tanks, which are related to the safety of the NPP, are rather few in number. In older NPPs failure of these tanks has been a dominant source of seismic risk. The failure mechanism has been extensively studied for oil storage tanks, but the exact mechanism of failure for so-called ‘elephant foot buckling’ or ‘bulging’ is not clear. For smaller sized (maybe less than 5 m in diameter) tanks, it is rather simple because the consideration of elephant foot buckling is not necessary and the ordinary buckling criteria should be applied, if its proportion requires it.

5.8.7. Control rod and control rod drive mechanism

Control rod drive (CRD) mechanisms for both BWRs and PWRs are subjected to the effect of relative deformation between the reactor vessel and the supporting structure. This deformation may be caused by the support deformation in the case of a BWR, and by a failure of RC internal structure or a slip of primary coolant loop with reactor vessel in the case of deformations of nozzle supports in a PWR. For CANDU reactors this may occur by relative deformation of a main building floor or partial or complete failure of anchorage of the reactor vessel. In some earlier NPPs, lateral supports were not provided and the CRDs acted as cantilever beams subjected to seismic inertia loading.

The lack of insertion of control rod is a key mode of failure. There are several modes of failure, like loss of driving function of CRD system in a BWR, bending of control rod drive housing (PWR), jamming of control rod cover plate (BWR), deformation of the core assembly, etc. The CRD systems fragilities require scaling of design information or possibly a new analysis.

5.8.8. Pump and turbines

Dynamic effect of rotating parts should be considered as well as the non-linearity of their bearing characteristics. Some of them, like a vertical shaft pump, a high speed turbine, have lower eigen-frequencies due to the rotating part, and their response to the earthquake input should be seriously considered. The method of evaluating their behaviour is mostly by simulation. The analysis of the failure of their casing is almost similar to that for an ordinary pressurized vessel, but their configuration is more complicated. Most important is to take into account the anchoring. Earthquake experience has shown most pumps to be seismically rugged as long as they are properly anchored. For most sites, pumps are treated generically or screened out. High seismic sites may require more specific evaluation.

5.8.9. Compressors

Compressors are rigid and have almost no weak point in their body. Exceptions may be attached equipment. In general, they are very strong and are usually modelled using generic fragility curves unless a walkdown reveals some unusual feature which requires specific modelling.

5.8.10. Emergency generator

Emergency diesel generators like compressors are rigid and seismically rugged except possibly some of the auxiliary systems such as the fuel system, exhaust system, cooling water lines, shock isolators, if any, etc. Typical failure modes include the air start receiver tanks, diesel fuel oil day tanks, the exhaust system and supporting electrical and control panels. These must be carefully checked on the drawing or in the field by a walkdown survey.

This is a typical case to figure out a global fragility curve based on several fragility curves of components.

A gas turbine generator can be treated like the emergency diesel generator.

5.8.11. Instrumentation system

Instrumentation systems include relays, switches, panels and racks, and cable trays as major items. They have different characteristics for the fragility analysis. Relays, switches, and instruments can be examined like an appendage to a supporting panel. A fragility curve for these items may be drawn based on the input acceleration at the panel point. To estimate their fragility characteristics analytically including malfunctioning is not practical. Most cases registered in existing data banks are those obtained by testings. It should be noted that some of the relays which consist of induction type discs may malfunction only under two dimensional excitation. Other types which have certain non-linear characteristics may malfunction under other particular operating conditions. Solid state relays are inherently rugged and are usually screened out.

The fragility analysis for a panel or rack is similar to other components. Their anchorage is generally their weakest points.

5.8.12. AC and DC power systems

Failure of insulators in the switchyard are common and result in an initiating event. The safety related switchgear, motor control centres and DC breaker panels are generally rugged as long as they are well anchored. These items are typically qualified by test and functional fragilities are derived from the test data as described in Section 5.6.3. Batteries and racks are usually rugged as long as the racks are well anchored and braced and the batteries securely nested in the racks with clearances taken up by spacers. Focus is usually on the rack structure and anchorage.

5.8.13. Overhead crane and refuelling machines

The possibility of falling of the whole system or parts of it may be analysed as for an ordinary structure.

5.9. PIPING

The fundamental procedures for evaluating the fragility curve for piping systems have been described in the previous discussions of development of fragility curves by scaling.

They are suitable for almost all pressurized, thick wall pipings. At present, there are many discussions on utilizing fracture mechanics and LBB concepts. It is almost certain that the compliance is high, and therefore there is no possibility of the occurrence of the DEGB before LBB occurs. However, in a seismic event, if LBB occurs several stress cycles before DEGB, the concept is invalid because of the short timing. DEGB is mainly caused by the relative movement of supporting points, but some test results show the possibility of DEGB induced by seismic inertia forces. Reference [8] describes a simplified fracture mechanics approach used in conjunction with the simulation method. A master fragility curve was developed based upon a simple fracture mechanics model. This master curve is then modified based on piping size, configuration, welding conditions, etc. The fragility curve was defined in terms of bending moment for comparison to bending moment calculated by simulation.

Simulation based on fracture mechanics is awkward, and there is some uncertainty for the assumption of existing undetected cracks. Therefore, this method should be used for only the most important cases. Another point of estimating the fragility curve of a piping system is the assumption of whether or not supporting structure and devices may fail.

In general, the reliability of engineered supporting systems is high. The assumptions that no failure of supporting devices would occur may be unrealistic for evaluating the failure of many piping systems in older NPPs, but earthquake experience has shown that support failure is uncommon and when it does occur the piping is rarely damaged.

Low cycle fatigue analysis is also a key technique for developing piping fragility curves. Other progressive type of damage such as fatigue ratcheting may be evaluated in the same way. The technique to evaluate such cumulative damage is established but is labour intensive.

Extensive studies using probabilistic fracture mechanics have been conducted for US plants [24–27] and the probability of DEGB in primary systems was found to be very low. Extensive testing of piping elements and systems shows large margin relative to inertial loading [28]. Based upon numerous studies of piping behaviour in earthquakes [29], piping systems are usually screened out unless undesirable features are observed in a plant walkdown. The major concern is seismic anchor motion, brittle joints and poor supports.

5.9.1. Large scale piping

Primary coolant loops are the most important piping in the NPP. They contain not only straight parts, but also elbows, tees, nozzles, pumps, valves, etc. Fundamental techniques to analyse their failure rate is as described in the previous section. In this case, the role of a supporting system is highly significant, and the failure rate of the piping may be governed by the reliability of the supporting section. In the US DEGB program [20–27] it was found that failure of primary equipment supports in PWRs was controlling. Similar results were obtained for BWRs [23, 27] except in the case of BWR piping with intergranular stress corrosion cracking (IGSCC). If IGSCC is present, direct failure of the primary piping may be controlling.

Large scale piping for a main steam line of a BWR is designed like a conventional pipeline on the outside of the containment vessel. It has more flexible vibration characteristics than PCL in the containment vessel, and successive failures of supporting devices should be considered in its failure analysis. However, these lines should be evaluated by simpler procedures, because their importance as a safety related item is not as significant.

Estimation of damping coefficients of the system is very important to the result of analysis, and constitutes a major system uncertainty.

5.9.2. Small bore piping

Small scale piping typically means a pipe whose diameter is less than 5 cm (2 in). Generic information will be sufficient in most instances. The main concern with small bore piping is seismic anchor motion which may result from a small stiff line being connected to a flexible larger line.

5.9.3. Piping connection

Piping connections are the most likely failure point. Mechanical couplings and threaded screws in connection are the most critical part for small scale piping. These types of connections are typically used on non-safety fire protection piping but the breakage could result in a spray or falling hazard to other safety related equipment. Elbows and tees of large scale piping are critical points as well as nozzles to vessel, pump and valve. This criticality is generally described by indices for local stress.

In general, piping is not a significant contributor to seismic risk and not much is expended in seismic PSAs in development of detailed fragility curves. Only specific issues uncovered in a walkdown are addressed. As an example of a specific concern is the connection of a small scale piping to a large scale piping if the large pipe is flexible and the smaller branch is stiff. Building differential motion effects on stiff piping is another example of concern.

5.9.4. Valves

Valves may be divided mainly into four groups, that is, large bore mechanically operated valves, small bore mechanically operated valves, small bore manually operated valves and force balanced types of safety valves. Points of failure are different for each group. The body or casing itself is considered to be a pressurized part as other parts of a piping or a pressure vessel, and the way of analysis is similar in general.

Most large valves are seismically rugged and are either screened out or generic fragility curves are used. Unusual eccentricity of heavy motor or hydraulic operators warrant a specific evaluation.

Small bore mechanically operated valves are potentially more vulnerable due to the eccentric operator mechanisms. This makes a structural weak point in the valve itself as well as the connecting portion of the piping. Generally, it is required to support the driving part and the piping near the valve body, and some are not designed in this way. In such cases, they may have a potential for failing due to vibratory excitation. Impact of the operators on adjacent structural members is a potential failure concern. In fact, this is the only observed type of valve failure in an extensive survey of piping systems subjected to strong motion earthquakes [29].

Manually operated small bore valves do not constitute a problem and are almost always screened out.

Force balanced safety valves may cause slippage of valve heads and sheet, if they are flat, because the force between them is reduced to almost zero under operational pressure conditions. This slippage may bring scratches on the sheet by the vibratory response of its driving mechanism, and some leakage could result after the event. Earthquake experience has not revealed any issues with these types of valves; however, some have been observed to have large eccentricity and appear to be marginal for large seismic events.

5.9.5. Small pipes on rack and cable tray

Small bore pipes, especially for instrumentation including in-core monitoring system and control rod driving system, are often formed into a bundle or mounted onto a rack. Cables are also mounted on a rack. Their behaviour is usually similar to that of piping systems, but in some cases, those which are not well fixed on a rack drop from the rack or the tray, and could break. Evaluation of such situations may be performed based on the peak response acceleration of rack or tray as usual. It should be mentioned that the damping coefficient of bundled systems are high compared with a single pipe. The critical failure mode is usually the rack or its anchorage.

5.9.6. Air conditioning system

Air conditioning systems consist of air ducts, a blower, heat exchangers and air dampers. The behaviour of air ducts is similar to that of cable trays as described in the previous section. The behaviour of a blower is similar to that of a turbine, and the behaviour of heat exchangers and dampers that of vessels and thin shell storage tanks. Buckling of thin shells, separation of corner joint of plates, buckling and excess deformation of the frame are typical failure modes of this system. Usually, the critical failure modes are associated with anchorage of the heavier blower and heat exchanger assemblies. Blowers on vibration isolators have typically failed in large seismic events.

5.9.7. Supporting structure and its devices

Failure of these can be controlling and is a key to the failure of the total piping system. They can be divided into several items, such as, concrete wall, steel frames, a rigid restraint, a rod-type restraint, a constant load hanger, a spring support, a mechanical snubber, a hydraulic snubber, etc. The first four are treated as a structure including the buckling problem. The last two have the problem of reliability even under normal conditions. Their failure mode is not clear because they have never been analysed or tested sufficiently, but some data show the fact that the possibility of their failure under seismic conditions is not negligible. Their margin is very variable depending on size. If we try to estimate their failure using simulation of the total system of a particular piping, we need information on the failure of supporting devices which are used for the system based on tests if possible.

The relative displacement of two or more fixing points of the system may cause additional strain on a piping system. The value of the relative displacement of a supporting structure may be calculated based on its response.

5.10. OTHER CONCEPTS

References [30] and [31] present some new ideas in the development of fragilities which have not classically been used in past seismic PSAs. In general, the development of fragility curves for most components is highly judgemental based on historical performance of piping and equipment in past earthquakes or observations during the walkdown and simple scaling of design analyses or generic research results. For high seismic zones such as in Japan, more sophisticated simulation and fracture mechanics methods may be required.

6. SYSTEMS ANALYSIS AND SAFETY EVALUATION

6.1. INTRODUCTION

This section is mainly concerned with the differences between seismic PSA and internal initiator PSA. It is not intended that a comprehensive and detailed review of internal initiator PSA be presented here. There are numerous references which provide such a detailed review to which the reader is referred (e.g. NUREG/CR-2300, NUREG/CR-2815 [6, 7]). It is highly recommended that the reader be well acquainted with internal initiator PSA before attempting to comprehend this section.

In Section 2 the general procedure for performing a seismic PSA has been outlined (see also Fig. 1). Figure 1 shows that the major steps for accomplishing a seismic PSA are:

(1) development of a seismic hazard curve;
(2) structure and component seismic response determination;
(3) assignment of structure and component fragility;
(4) random failure data development;
(5) event/fault tree construction and solution;
(6) risk quantification incorporating results of steps 1 through 5.

This section focuses on steps 5 and 6. Each of steps 1 to 3 have been addressed in Sections 3 and 5. Step 4 is identical to that for internal initiator PSA.

6.2. ACCIDENT SEQUENCE DEFINITION/EVENT TREES

In this document the use of well established event tree-fault tree methods, widely used for modelling systems behaviour in the case of internal accident initiators, is described. Other methods such as concern tree-fault tree modelling, which is being developed to avoid the impractically large event trees which can arise when multiple simultaneous failures are represented, are not described.

6.2.1. Initiated plant states

Before accident sequences can be defined, the initiated failure states which prompt the response of the safety systems must be clearly and uniquely identified. For seismic PSA, this process is essentially the same as for internal initiator PSA. However, this does not mean that the set of initiated states used for internal initiator PSA is automatically sufficient for a seismic PSA. In particular, some types of initiated states may be considered too improbable for an internal initiator event analysis and yet not be negligible in the seismic PSA. An example of this could be reactor vessel rupture (RVR).

The result of the identification of initiated failure states should be a set of well defined and quantifiable failures of plant components or structures. For example, in a nuclear power plant, a typical initiated state is a large LOCA. This state is defined by a particular range of pipe ruptures in the primary cooling loop of the reactor, and necessitates a particular response of plant safety systems for an appropriate mitigation of the consequences of a large LOCA. The result of initiated state identification could be a set of pipe rupture combinations, each of which would result in a large LOCA.

In Section 4, discussion was made of the many types of indirect failures, such as those produced by systems interactions, tsunami flood, avalanche, etc., which might also be induced by the seismic event. For other than the systems interaction case these indirect failures, although caused by the earthquake, are (typically) not treated in the accident sequence resulting from the seismic initiators. They are usually treated in the external initiator PSA using methods which have been developed for them specifically. The effects of a seismically induced flood may be quite different from those of other external floods because the plant may be subjected to more than one threat to safety. Thus, the damage to the plant due to the combination of seismic and flood effects may be greater than that arising from either occurring separately. Perhaps more important and more problematical than the possible associated external initiators are internal initiators, such as fire and possibly also floods, which may be initiated by the seismic event. In the case of fires, the seismic initiator may fail fire barriers and fire detection/mitigation systems, and the combination of fire and structural motion may together produce more damage in other safety systems than if they had occurred separately. This aspect of indirect seismic failures is obviously quite complex and has yet to be fully modelled.

6.2.2. Facility functional response

The description of the facility functional response to initiators of various classes is carried out in the same way in seismic PSA as in internal initiator PSA; once an accident has been initiated, the facility should attempt to mitigate the consequences of the initiated state (e.g. large LOCA) in the same way, regardless of whether it was induced randomly or as the result of an earthquake. Thus, one would expect that the functional event trees developed for seismic PSA would be similar, if not identical, to those used in the internal initiator PSA.

It is possible that operator procedures, or even plant equipment, could be designed to respond differently to seismically initiated failures than those initiated randomly. In such a case, the event trees developed for the seismic PSA would be different from those developed under the internal initiator analysis. Operator error could be enhanced due to a seismic event. This is difficult to quantify due to lack of data and is usually ignored in a seismic PSA. Also, as mentioned above, additional types of initiators might be considered for seismic PSAs which were not considered for internal initiator PSA. In this case, additional event trees would be needed for the seismic PSA to accompany those used in the internal initiator analysis.

6.3. SYSTEMS ANALYSIS AND FAULT TREE DEVELOPMENT

6.3.1. Safety system failure criteria

Once the functional response and the associated event trees have been developed, the failure criteria for the safety systems can be established. These criteria provide the definition of equipment failures within the safety systems which result in loss of system function. This definition is vital to the development of clear top event definitions in the fault tree development stage.

The development of safety system failure criteria in seismic PSA should proceed exactly as in an internal initiator PSA. In fact, unless the facility under study has systems which are designed to respond differently for seismic initiators, one would expect that the safety system failure criteria will be identical to those of the internal initiator PSA, for corresponding initiated fault conditions.

We note that, if simplification is necessary or desired, it may be possible to reduce the effort during the safety system failure criteria process by assuming success or failure of particular safety systems. For example, off-site electrical power, and facility systems dependent on it, may be quite vulnerable to seismic loads, making it possible to assume loss of these systems, especially if this is a conservatism. This would eliminate the need for detailed analysis of the function and design of those systems.

6.3.2. Fault tree development

Undoubtedly, one of the largest efforts in seismic PSA, as well as internal initiator PSA, is the construction of fault trees to describe logically the combinations of equipment failures which can lead to failure of safety systems to carry out their function adequately. There is, of course, considerable leeway in the degree of detail, or 'depth', to which the fault trees are developed. However, it is usually desired to model the safety systems down to the level of discrete equipment items (e.g. pumps, valves, circuit breakers, etc.), which, in turn, determines a minimum depth in the fault trees.

If no fault trees exist for internal initiators, then their development can be a major effort for seismic PSA. However, it is probably the case with most PSAs that, if a seismic PSA is being carried out, an internal initiator PSA has already been, or is being, carried out. For this reason, and because numerous references exist which describe fault tree development and analysis, we focus here on the steps in fault tree construction which are particular to seismic PSA.

There are two major modifications to internal initiator fault trees which must be made to adapt them to a seismic PSA. First, it is necessary to incorporate those failures which, because of their extreme improbability, are not included in the internal initiator fault trees. This would include failures of structures (e.g. wall or roof collapse, severe concrete spallation, basemat uplift, tilting, etc.), or relay chattering. These events are so improbable without seismic (or other energetic external) excitation that they are most likely disregarded in the internal initiator PSA.

The second major modification involves those failures which are considered in the internal initiator analysis but which must have a second, seismic failure mode included for them in the fault tree. For example, an internal initiator PSA might include, as a primary failure, the random failure of a motor operated valve to close on demand. The seismic PSA fault tree could be modified to change the motor operated valve failure to close from a primary failure into a gate failure. This gate failure would be an OR gate having two primary failures as input, the random failure of the valve and the seismically induced failure of the valve.

This second type of modification has the potential for increasing greatly the size of the fault trees, easily doubling the number of primary failures. Fortunately, not all random failures would have to be paired with a seismic failure when modifying the internal initiator fault trees. Some failures, such as maintenance errors, are not affected by earthquakes. Other types of equipment, such as fuses or electrical cabling, might be considered so highly resistant to seismically induced failure that their probability of seismically initiated failure can be neglected. One must be careful in making these types of a priori decisions since they are tantamount to probabilistic culling, which is discussed below.

Lastly, some types of failure, such as operator error, are dependent on earthquake level but not as easily modelled with two separate failure modes, i.e. random and seismic, as are equipment items. For these, the basic failures can be left as they are in the internal initiator fault trees; however, when quantifying the fault trees, these earthquake level dependent failures can be assigned probabilities which vary with earthquake level but which are estimated without necessarily employing explicit response/fragility models.

7. INTERPRETATION OF THE RESULTS

Within the scope of the present document, the results of a Level 1 seismic PSA consists of the occurrence frequency of core damage. It is customary to synthesize this information into the median or mean value of the frequency and two fractile values (lower and upper) defining a range of frequencies within 90% (or any other desired fraction) of the frequency contained.

The central value (median or mean) can be roughly thought to reflect the contribution to the risk due to intrinsic randomness while the 'confidence interval' gives the measure of uncertainty with which the core damage frequency is obtained.

TABLE II. CONTRIBUTION OF INITIATING EVENTS TO MEAN ANNUAL CORE MELT FREQUENCY FOR PUBLISHED PSAS WITH COMPLETE SEISMIC ANALYSIS

Plant	Date	Contribution (%)					
		Seismic	Internal	Fire	Wind	External	Mean annual core melt frequency
Zion	1981	8	85	7	–	–	6.7×10^{-5}
IP2	1983	6	58	10	26	–	1.4×10^{-4}
IP3	1983	2	88	9	1	–	1.4×10^{-4}
Seabrook	1983	13	75	11	–	1	2.3×10^{-4}
Limerick	1983	13	34	53	–	–	4.4×10^{-5}
Millstone 3	1984	15	77	8	–	–	5.9×10^{-5}
Oconee 3	1984	25	56	4	5	10	2.5×10^{-4}

Notes:
- Contribution to core melt is not necessarily indicative of public health risk contribution.
- Seismic events that initiate core melt accident sequences are generally more likely to also cause damage to containment than other initiating event.
- Comparison of median (rather than mean) seismic risk to median core melt frequency would indicate in most (but not all) cases lower seismic contribution.

Results of a seismic PSA are typically compared to results from internal events and other external events (Table II from Ref. [32] shows a comparison of mean core melt frequencies). In the IAEA Code on the Safety of Nuclear Power Plants: Siting [33] — General Criteria 305, it is stated that the radiological risk associated with external events should not exceed the range of radiological risk associated with the accidents of internal origin. This comparison when made by a PSA requires a Level 3 analysis. When comparing Level 1 results, exact comparisons may be misleading due to the fact that the uncertainty in the seismic induced core damage frequency may be much greater than for internal events or other external events. Comparisons of the means, medians and 84 percentile values may place different emphasis on the major contributors to core damage and the analyst must be objective in his evaluation of the results and actions that he may recommend for improving plant safety. Containment performance is discussed below and should also play an important role in any safety improvement recommendations.

Once the PSA results for internal and external events are available, it is customary to conduct sensitivity studies and cost benefit studies before making any decisions on backfitting measures. The completed PSA can be a very effective tool for optimizing the resources that are expended to improve plant safety.

Initial results from the seismic PSA may indicate that relatively inexpensive backfits are in order that will significantly reduce the seismic contribution to risk. In other cases, any backfits to significantly reduce the seismic contribution to risk may be quite costly and it may be justified to recompute selected seismic fragilities using more sophisticated non-linear methods to account for redundancy, conduct fracture mechanics analysis, to conduct experiments to determine the actual capacity of a dominant contributor or to examine recovery actions.

If the seismic PSA includes a containment performance evaluation, useful information will be obtained on potential sources of seismic induced containment failure or containment bypass. This information along with sensitivity studies may then be useful in deciding whether any backfits are justified. In NUREG 4334 [13], it was concluded that events that led to core melt also resulted in the loss of accident mitigations systems and ultimate containment failure, thus the supporting systems that are common for core cooling and containment cooling may have more importance from a risk viewpoint than systems dedicated to one or the other.

In some countries, the accident mitigation systems are designated seismic category 2 and designed to a lower level of earthquake than the category 1 systems associated with the core cooling, pressure control and reactivity control. For a seismic induced core melt incident the mitigating systems may already have failed before the failure that results in core melt curves.

In interpreting the results of Level 1 analyses, one should also be examining the failure mode and its affect or containment performance so that decisions can be made that focus beyond Level 1 results.

Results of the PSA of internal and external events should be compared to results for other NPPs of similar design as an aid in interpreting the results. Significant differences in results may be rational if there are unique vulnerabilities in one of the plants. If these unique vulnerabilities are not apparent, then there may be some modelling differences that may warrant further examination.

The PSA, including seismic and other external events is a very useful tool to study plant response to initiating events, develop accident management programs and manage the expenditure of resources for backfitting. With more and more PSA studies being conducted, the usefulness will increase for enhancing the safety of NPPs in a logical and economic manner.

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LIST OF ABBREVIATIONS

AC	Alternating current
ATWS	Anticipated transient without scram
BWR	Boiling water reactor
CCW	Component cooling water
CD	Core damage
CRD	Control rod drive
DC	Direct current
DEGB	Double ended guillotine break
HCLPF	High confidence of low probability of failure
LBB	Leak before break
LOCA	Loss of coolant accident
NPP	Nuclear power plant
PC	Pre-stressed or post-stressed concrete
PCD	Probability of core damage
PCL	Primary coolant loop
PGA	Peak ground acceleration
PRA	Probabilistic risk analysis
PSA	Probabilistic safety assessment
PSHA	Probabilistic seismic hazard analysis
PWR	Pressurized water reactor
RC	Reinforced concrete
RVR	Reactor vessel rupture
SSI	Soil-structure interaction
SSMRP	Seismic Safety Margin Research Program

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