EXTERNAL EVENTS PRA METHODOLOGY STANDARD

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SECTION 1 ---- INTRODUCTION

1.1 Objectives

The objectives of this Standard are to set forth requirements for external-event probabilistic risk assessments (PRAs) used to support risk-informed decisions for commercial nuclear power plants, and to prescribe a method for adapting these requirements for specific applications.

1.2 Coordination with Other PRA Standards

The Standard is intended to be used together with other PRA standards that cover different aspects of PRA scope. Specifically, this Standard is intended to be used directly with the PRA Standard developed by ASME (ASME, 2000) that covers an "internal events PRA" at full power. (See Section 1.3 below for a more complete description of the ASME scope.) Similarly, this Standard is intended to be used with the American Nuclear Society standard covering low-power/shutdown operations, when that standard, now under development, is completed. However, additions and modifications to the technical requirements will be necessary to cover low-power/shutdown applications.

This Standard is also intended to be used together with Standard ANS 2.27, "Standard covering Guidelines for Investigations of Nuclear Materials Facilities Sites for Seismic Hazard Assessments" (ANS, 2000), and "Standard ANS 2.29, "Standard for Probabilistic Analysis of Natural Phenomena Hazards for Nuclear Facilities" (ANS, 1997), when those standards, now in draft form, are completed. ANS 2.27 and ANS 2.29, which will have more detail than this Standard in certain technical areas, are referred to in the appropriate places in this Standard that cover requirements related to hazard analysis.

1.3 Scope

The PRA scope covered by this Standard is limited to analyzing accident sequences initiated by external events that might occur while a nuclear power plant is at nominal full power. It is further limited to requirements for (i) a full Level 1 analysis of the core damage frequency (CDF) and (ii) a limited Level 2 analysis sufficient to evaluate the large early release frequency (LERF).

External events covered within the Standard's scope include both natural external events (e.g., earthquakes, high winds, and external flooding) and human-made external events (e.g., airplane crashes, explosions at nearby industrial facilities, and impacts from nearby transportation activities).

In contrast, the scope of the ASME Standard (ASME, 2000) covers internal plant initiators (except internal fires) that might occur while the nuclear power plant is at full power. Accidents initiated by internal flooding are explicitly included in the ASME standard, as are accidents initiated by a loss of offsite power. Therefore, this ANS Standard and the ASME Standard, when used together, cover all potential accident initiators arising at full power, except internal fires. The only initiators explicitly excluded are accidents resulting from purposeful human-induced security threats (e.g., sabotage). Although (as discussed above in section 1.2), this Standard is intended ultimately to be used with the American Nuclear Society standard covering low-power/shutdown operations when that standard is completed, accidents initiated by external events occurring during low-power/shutdown conditions are explicitly <u>not</u> covered by the requirements herein. Additions and modifications to the technical requirements will be necessary to cover such applications.

1.3.1 Scope: Screening Analysis and Demonstrably Conservative or Bounding Analysis

The scope of this Standard includes not only traditional PRA analyses, which are intended to be realistic, but also screening analyses and demonstrably conservative or bounding approaches that use aspects of PRA methodology but are not full-scope PRAs themselves (see Section 3.6, for example). Many risk-informed applications can and do use such analyses. (Herein, the phrases "bounding analysis" and "demonstrably conservative analysis" are used interchangeably.)

1.3.2 Scope: Seismic Margin Assessment Methodology

The scope of this Standard includes not only a traditional external events PRA, but also the widely-used Seismic Margin Assessment (SMA) methodology (see Section 3.5). SMA methods employ many of the same tools as a seismic PRA, and the decision to include SMA methods here is motivated by the desire to allow an SMA to be used for some risk-informed applications. The scope of an SMA is more limited than the scope of a seismic PRA, so some risk-informed applications cannot be supported by an SMA, or can be supported only to a limited extent.

In particular, an SMA using the so-called "EPRI SMA method" (EPRI, 1991), which is the approach used for almost all of the SMAs that have been performed, does not employ a systems model that permits the development of a full core-damage frequency (CDF), nor does the systems-analysis approach account for non-seismic unavailabilities and human errors in a systematic manner. This means that there are important limitations to the types of risk-informed applications that such an SMA can support. While various proposals have been advanced to remedy some of the aspects of this limitation, and work to evaluate these proposals is underway in the seismic-

PRA/seismic-SMA community, this work has not been completed as of the date of the publication of this standard. [See the second paragraph in Section 3.5 for a discussion of how this Standard applies to an SMA using the so-called "NRC SMA method" (Budnitz et al., 1985; Prassinos, Ravindra, and Savy, 1986).]

Another particular limitation is very important: The systems analysis aspect of an SMA contemplates only the evaluation of success paths that would prevent a core-damage accident sequence. Within an SMA, there is no explicit way to separate those core-damage accident sequences that might lead to a "large early release" (see the discussion of LERF in the next section, 1.3.3) from other sequences that would not lead to such a release. Hence the entire area of applications related to LERF is beyond the capabilities of an SMA unless explicit enhancements are undertaken.

Throughout the Standard, the phrase "PRA" is used in a generic sense. Often (for example, in most of the language in this introductory Section 1), the intent is to include SMA methods as well as PRA methods within the scope of the phrase "PRA." For example, both PRA and SMA methods are definitely contemplated as being considered together for the purposes of Sections 1.4 to 1.10 below, even though the language generally uses "PRA" throughout.

1.3.3 Scope: The LERF Endpoint

As discussed above in Section 1.3, the Requirements herein cover "(i) a full Level 1 analysis of the core-damage frequency (CDF) and (ii) a limited Level 2 analysis sufficient to evaluate the large early release frequency (LERF)."

The approach to any external events PRA typically uses as its starting point the internal-events PRA model, to which must be added a number of SSCs not included in that model, but which could fail due to the fact that the accident initiator is an external event. Some "trimming" of that model is also common, to eliminate parts of it not relevant to the external-events analysis. (See REQ. SA-A3 in Section 3.4.2 and the second paragraph under Section 3.4.2.1 (under seismic PRA) for more discussion of these issues.) Both the part of the internal-events model dealing with CDF and the part dealing with LERF are used as starting points.

The analysis of the LERF endpoint proceeds in the same way as the analysis of the CDF endpoint, with one major exception, as follows: There are some accident sequences, leading to core damage but not to large early releases in the internal-events PRA model, that need to be elevated to potential LERF sequences when the initiator is an external event. These are sequences in which offsite protective action (specifically, the evacuation of nearby populations) is impeded due to the external event. The same sequence that might not be a LERF sequence due to any internal initiator may perhaps affect nearby populations who cannot evacuate as effectively.

These sequences would fall into the LERF category because the word "early" in the definition of "LERF" does not refer to a specific point in time, but rather to the issue of whether a large release might occur before effective protective actions (e.g., evacuation and sheltering) can be implemented to protect surrounding populations.

For example, suppose that an earthquake or tornado that initiates an accident sequence at the nuclear plant were to damage the only road available to evacuate close-in populations. Without effective evacuation, these populations may be exposed to radioactive releases that they would not be exposed to, were the same accident sequence to arise from an internal plant fault.

Therefore, in analyzing external events that have the potential to impede effective emergency evacuation, the analysis must examine whether any accident sequences that are not in the "LERF" category in the internal-events PRA model need to be included in that category for the particular external event being evaluated. The LERF part of the PRA analysis would require expansion accordingly.

1.3.4 Scope: Light Water Reactors in the Design or Construction Phase

This Standard is based mainly upon PRA methodologies and applications that have evaluated U.S. light-water nuclear power reactors that are already in commercial service, and specifically contemplates applications for those reactors. It is also applicable, with appropriate adaptations, to similar LWRs in the design or construction phase. Of course, at these earlier stages, the available information is not as complete as for an operating unit, so generic information must be used for certain inputs, which limits the usefulness of the resulting PRA.

1.3.5 Scope: Other Types of Nuclear Power Reactors

Although this Standard is based mainly upon PRA methodologies and applications that have evaluated U.S. light-water nuclear power reactors, and specifically contemplates applications for those reactors, it is also applicable, with appropriate adaptations, to other types of nuclear power reactors.

1.4 Types of Applications

The types of risk-informed PRA applications contemplated under this Standard are very broad, and include applications related to design, procurement, construction, licensing, operation, and maintenance. Both regulatory risk-informed applications and applications not involving the NRC's regulations are contemplated. In this regard, the approach is intended to be identical to that used in the ASME Standard (ASME, 2000).

The ASME Standard was not written to support any specific applications, but is concerned only with the capability of a PRA to support an application. PRA capabilities fall on a continuum, but for convenience the ASME Standard has identified three different capability levels, described in its Section 1. These three different capability levels (called "Categories" I, II, and III) manifest themselves in the ASME Standard through the presence, for each technical area covered, of three different Supporting Requirements written to cover the three different capability levels. To quote the ASME Standard in Section 1.3, "PRA Capabilities are evaluated for each Supporting Requirement, rather than by specifying a 'capability level' for the whole PRA. Therefore, only those aspects of a PRA element required to support the application in question need the capability level appropriate for that application. For a given application, supplementary analyses may be used in place of, or to augment, those aspects of PRA elements that do not fully meet the requirements...." The ASME Standard's Chapters 1 and 4 have a more complete explanation. Although ASME's Supporting Technical Requirements are different for each Category, all of ASME's Supporting Technical Requirements fall under a single set of High Level Requirements, independent of which Category they fit.

However, the three-category approach has not been used here. Rather, the requirements in this Standard are written for only one "category" of PRA capability, corresponding to the ASME Standard's Category II. The user should be aware that in developing this Standard the ANS standards-development working group has tried to adhere closely to the ASME Standard's Category II, which represents a high-quality PRA useful for a broad range of risk-informed decisions. A PRA meeting this Standard should have the capability to be used for the same sorts of applications contemplated for the ASME Standard's Category II capabilities (see ASME Chapters 1 and 4 for a more detailed discussion of typical applications). Such a PRA should also be capable of supporting Category I applications.

An analysis using the "Seismic Margin Assessment" methodology does not produce the same types of results as a seismic PRA --- for example, it does not produce a core damage frequency estimate --- and cannot support certain applications contemplated under the ASME Standard's Categories I and II. However, a well-executed SMA represents a good fit to many of the applications contemplated for ASME's "Category I," especially insofar as an SMA is generally well-suited to the categorization of structures, systems, and components (SSCs) according to their seismic capacity, and to the screening of SSCs according to their safety significance. A well-executed SMA is also a good fit for some applications contemplated for ASME's "Category II", but a judgment must be made for each application on a case-by-case basis.

1.5 The Nature of the Requirements

Shall, Should, and May: The requirements contained herein are all phrased in the usual language of standards, namely the language of "shall," "should," or "may." These three terms are defined in Section 2. These definitions are repeated here:

shall – used to state a mandatory requirement

should – used to state a recommendation

may – used to state an option to be implemented at the user's discretion.

The Phrase "Shall Consider": As used herein, the phrase "shall consider" is distinctly different from the word "shall." The word "shall" requires that something be done. When a requirement uses the phrase "shall consider X" (where X is some activity), the intent of the Standard is to require the consideration, but to allow the analyst not to proceed to perform the full study or task if a case can be made to support such an approach. The phrase "shall consider" also requires that the documentation substantiate the way in which the consideration was accomplished. For example, Section 3.4.2 under seismic PRA, REQ. SA-B6 states: "The systems analysis SHALL consider the possibility that a large earthquake can cause damage that blocks personnel access to safety equipment and controls, thereby inhibiting operability actions that might otherwise be credited." The intent is that the analysis must consider access-blockage issues for each important accident sequence, and also must document how this was done, for example by looking for access-blockage issues during the seismic-PRA walkdown. The documentation must be adequate for the purposes of peer review, and it is understood that the peer-review team will pay particular attention to this topic.

The Phrase "Acceptable Method": In many places, the Commentary contains words such as, "Reference X provides an acceptable method for performing this aspect of the analysis." The plain meaning of this wording should be clear, namely that using the methodology or data or approach in Reference X is one way to meet the Standard. The intent of any Requirement that uses this language is to be permissive, meaning that the analysis team can use another method without prejudice. However, it is important to understand that the intent of the Standard goes beyond the plain meaning, as follows: Whenever the phrasing "acceptable method" is used herein, the intent is that if the analysis uses another method, the other method must accomplish the stated objective with a comparable level of detail, a comparable scope, etc. It is not acceptable to use another method that does not accomplish the intent of the Requirement at least as well as the "acceptable method" would accomplish it. Whenever an alternative to the "acceptable method" is selected, it is understood that the peer-review team will pay particular attention to this topic.

1.6 Risk Assessment Technical Requirements: Section 3

Section 3 provides specific technical requirements for each PRA technical element.

The approach to developing the PRA technical requirements has concentrated on "what to do" rather than "how to do it." In that sense, specific PRA methods for satisfying the technical requirements are not prescribed, although certain established PRA methods were contemplated by the Working Group authors as the technical requirements were being developed.

Therefore, alternative methods and approaches to meet the technical requirements of this Standard may be used if they provide results that are equivalent or superior to the methods usually used. The use of any particular method to meet a technical requirement SHALL be justified and documented, and SHALL be subject to review by the peer-review process described in Section 5.

1.7 PRA Configuration Control: Section 4

In order to conform to this Standard, a PRA SHALL be maintained under a PRA Configuration Control Program, the requirements for which are provided in Section 4. The objective of the PRA Configuration Control Program is to ensure that the PRA reflects the as-built, as-operated facility to a degree sufficient for the application in which the PRA is used.

1.8 Peer Review: Section 5

In order to conform to this Standard, a PRA SHALL be peer-reviewed to evaluate the capability of each of its elements to support intended applications. Section 5 provides the requirements for the peer review. General peer-review requirements are supplemented by specific requirements applicable to seismic PRA, seismic-margin assessment, and the PRA analysis of other external events.

1.9 Risk-Assessment Application Process: Section 6

Section 6, which incorporates by reference the requirements found in Section 3 ("Risk Assessment Application Process") of the ASME PRA Standard (ASME, 2000), describes requirements for a process that SHALL be used to determine the capability of a PRA to support various applications. The use of a PRA will be different from application to application. The Standard, which is application-non-specific, is concerned only with the capability of the PRA to support an application. The PRA's technical capabilities are evaluated against the Standard requirement-by-requirement, rather than by evaluating whether the PRA as-a-whole has all of the appropriate technical capabilities to "meet the Standard." Therefore, only those PRA elements required to support the application in question need to meet the technical capability

level of the Standard. As set forth in Section 6, for any given application, supplementary analyses may be used in place of, or to augment, those elements which do not fully meet the technical capabilities represented by the requirements in ASME's Section 3.

Although ASME's Section 3 was written with a PRA in mind, the requirements therein apply equally well to applications using a Seismic Margin Assessment that meets this Standard.

1.10 Documentation Requirements: Section 7

Section 7 contains several general documentation requirements that apply through the Standard. In addition, under the Technical Requirements for each external event, there are a few additional documentation requirements specific to that external event.

SECTION 2 ---- DEFINITIONS

2.1 Acronyms and Initialisms

BWR - Boiling Water Reactor
CCF - Common Cause Failure
CCDP – Conditional Core Damage Probability
CDF - Core Damage Frequency
CDFM – Conservative Deterministic Failure Margin Method
CEUS – Central and Eastern United States
DOE – U.S. Department of Energy
ECCS - Emergency Core Cooling System
EDG - Emergency Diesel Generator
EPRI – Electric Power Research Institute
GIP – Generic Implementation Procedure
HCLPF – High Confidence of Low Probability of Failure
HVAC - Heating, Ventilation, And Air Conditioning
I&C - Instrumentation and Control
IE – Initiating Event
IPE – Individual Plant Examination
IPEEE – Individual Plant Examination of External Events
ISLOCA – Interfacing Systems Loss Of Coolant Accident
LERF – Large Early Release Frequency
LOCA - Loss Of Coolant Accident
MMI – Modified Mercalli Intensity
LOSP – Loss Of Offsite Power
NRC - United States Nuclear Regulatory Commission
OBE – Operating Basis Earthquake
PCS – Power Conversion System
pga – Peak Ground Acceleration
PMF – Probable Maximum Flood
PRA – Probabilistic Risk Assessment
PSHA – Probabilistic Seismic Hazard Analysis
PWR - Pressurized Water Reactor
QA - Quality Assurance
RCS - Reactor Coolant System
RLE – Review Level Earthquake
RPV - Reactor Pressure Vessel
SAR - Safety Analysis Report
SEL – Seismic Equipment List
SMA – Seismic Margin Assessment
SPLD – Success Path Logic Diagram
SPRA – Seismic Probabilistic Risk Assessment
SSC - Structure, System, or Component
SSE – Safe Shutdown Earthquake

SSEL – Safe Shutdown Equipment List
SSHAC – Senior Seismic Hazard Analysis Committee
SSI – Soil Structure Interaction
UHS – Uniform Hazard Response Spectrum

2.2 Definitions of Terms

accident consequences - the extent of plant damage or the radiological release and health effects to the public or the economic costs of a core damage accident

accident sequence - a combination of events, beginning with an initiating event, that challenges safety systems and resulting in an undesired consequence (such as core damage or large early release). An accident sequence may contain many unique variations of events (cut sets) that are similar.

accident sequence analysis - the process to determine the combinations of initiating events, safety functions, and system failures and successes that may lead to core damage or large early release

aleatory uncertainty - the uncertainty inherent in a non-deterministic (stochastic, random) phenomenon. Aleatory uncertainty is reflected by modeling the phenomenon in terms of a probabilistic model (which also must treat epistemic uncertainty.) In principle, aleatory uncertainty cannot be reduced by the accumulation of more data or additional information. (Sometimes called "randomness").

at power - those plant operating states characterized by the reactor being critical and producing power, with automatic actuation of critical safety systems not blocked and with essential support systems aligned in their normal power operation configuration

basic event - an event in a fault tree model that requires no further development, because the appropriate limit of resolution has been reached

CDFM method - refers to the Conservative Deterministic Failure Margin (CDFM) method as described in EPRI NP-6041 (EPRI, 1991) wherein the seismic margin of the component is calculated using a set of deterministic rules that are more realistic than the design procedures

common cause failure (CCF) - a failure of two or more components during a short period of time as a result of a shared cause

component - an item in a nuclear power plant, such as a vessel, pump, valve, or a circuit breaker

composite variability - The composite variability includes the randomness variability

and the uncertainty. The logarithmic standard deviation of composite variability, β_c, is expressed as $(\beta_R^2 + \beta_U^2)^{1/2}$.

containment analysis - the process to evaluate the failure thresholds or leakage rates of the containment

containment failure - loss of integrity of the containment pressure boundary that results in unacceptable leakage to the environment

core damage - uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated representing the onset of gap release of radionuclides

core damage frequency (CDF) – frequency of core damage per unit of time

deaggregation - determination of the functional contribution of each magnitude-distance pair to the total seismic hazard. To accomplish this, a set of magnitude and distance bins are selected and the annual probability of exceeding selected ground motion parameters from each magnitude-distance pair is computed and divided by the total probability.

dependency - requirement external to an item and upon which its function depends

distribution system - piping, raceway, duct, or tubing that carries or conducts fluids, electricity, or signals from one point to another

dominant contributor – a component, a system, an accident class, or an accident sequence that has a major impact on the CDF or on the LERF

epistemic uncertainty - the uncertainty attributable to incomplete knowledge about a phenomenon that affects our ability to model it. Epistemic uncertainty is reflected in a range of viable models, the level of model detail, multiple expert interpretations, and statistical confidence. In principle, epistemic uncertainty can be reduced by the accumulation of additional information. (Also called "modeling uncertainty").

event tree - a quantifiable, logical network that begins with an initiating event or condition and progresses through a series of branches that represent expected system or operator performance that either succeeds or fails and arrives at either a successful or failed end state

external event - an initiating event originating outside a nuclear power plant that, in combination with safety system failures, operator errors, or both, may lead to core damage or large early release. Events such as earthquakes, tornadoes, and floods from sources outside the plant and fires from sources inside or outside the plant are considered external events (see also internal event). By convention, loss of offsite

power and internal fires are considered to be "internal events."

failure mechanism – a physical explanation of why a failure has occurred. It can be characterized in many different ways, for example by the type of agent causing the failure (e.g., chemical mechanical, physical, thermal, human error) or by the physical process (e.g., vibration, corrosion.)

failure mode – a specific functional manifestation of a failure, i.e., the means by which an observer can determine that a failure has occurred (e.g., fails to start, fails to run, leaks).

failure probability - the expected number of failures per demand expressed as the ratio of the number of failures to the number of type of actions requested (demands)

failure rate - expected number of failures per unit of time expressed as the ratio of the number of failures to a selected unit of time

fault tree - a deductive logic diagram that depicts how a particular undesired event can occur as a logical combination of other undesired events

fractile hazard curves - a set of hazard curves used to reflect the uncertainties associated with estimating seismic hazard. A common family of hazard curves used in describing the results of a PSHA is curves of fractiles of the probability distributions of estimated seismic hazard as a function of the level of ground motion parameter.

fragility - Fragility of a system, structure or component is the conditional probability of its failure at a given hazard input level. The input could be earthquake motion, wind speed, or flood level. The fragility model used in seismic PRA is known as a double lognormal model with three parameters, A_m, β_R and β_U which are respectively, the median acceleration capacity, logarithmic standard deviation of randomness in capacity and logarithmic standard deviation of the uncertainty in the median capacity.

ground acceleration - acceleration at the ground surface produced by seismic waves, typically expressed in units of g, the acceleration of gravity at the earth's surface

hazard – the physical effects of a natural phenomenon such as flooding, tornado, or earthquake that can pose potential danger (for example, the physical effects such as ground shaking, faulting, landsliding, and liquefaction that underlie an earthquake's potential danger)

hazard (as used in probabilistic hazard assessment) – represents the estimate of expected frequency of exceedance (over some specified time interval) of various levels of some characteristic measure of a natural phenomenon (for example, peak ground acceleration to characterize ground shaking from earthquakes). The time period of interest is often taken as one year, in which case the estimate is called the annual

frequency of exceedance.

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

high winds -- tornadoes, hurricanes (or cyclones or typhoons as they are known outside the US), extra-tropical (thunderstorm) winds, and other wind phenomena depending on the site location

initiating event - any event either internal or external to the plant that perturbs the steady state operation of the plant, if operating, thereby initiating an abnormal event such as transient or LOCA within the plant. Initiating events trigger sequences of events that challenge plant control and safety systems potentially leading to core damage or large early release.

intensity - a measure of the observable effects of an earthquake at a particular place. Commonly used scales to specify intensity are the Modified Mercalli Intensity, Rossi-Forel, MSK, and JMA scales.

interfacing systems LOCA (ISLOCA) - a LOCA when a breach occurs in a system that interfaces with the RCS, where isolation between the breached system and the RCS fails. An ISLOCA is usually characterized by the over-pressurization of a low pressure system when subjected to RCS pressure and can result in containment bypass.

internal event - an event originating within a nuclear power plant that, in combination with safety system failures, operator errors, or both, can effect the operability of plant systems and may lead to core damage or large early release. By convention, loss of offsite power is considered to be an internal event, and internal fire is considered to be an external event.

large early release - the rapid, unscrubbed release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions

large early release frequency (LERF) - frequency of a large early release per unit of time

Level 1 analysis - identification and quantification of the sequences of events leading to the onset of core damage

Level 2 analysis - evaluation of containment response to severe accident challenges and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment

magnitude - a measure of the size of an earthquake. It is related to the energy released in the form of seismic waves. Magnitude means the numerical value on a standardized scale such as but not limited to Moment Magnitude, Surface Wave Magnitude, Body Wave Magnitude, or Richter Magnitude scale.

may - used to state an option to be implemented at the user's discretion

peak ground acceleration - maximum value of acceleration displayed on an accelerogram; the largest ground acceleration produced by an earthquake at a site

plant - a general term used to refer to a nuclear power facility (for example, plant could be used to refer to a single unit or multi-unit site)

point estimate - estimate of a parameter in the form of a single number

probabilistic risk assessment (PRA) - a qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public (also referred to as a probabilistic safety assessment, PSA)

PRA configuration control plan - the process and document used by the owner of the PRA to define the PRA technical elements that are to be periodically updated and to document the methods and strategies for maintenance of those PRA technical elements

probability of exceedance (as used in seismic hazard analysis) - the probability that a specified level of ground motion for at least one earthquake will be exceeded at a site or in a region during a specified exposure time

randomness (as used in seismic-fragility analysis) - the variability in seismic capacity arising from the randomness of the earthquake characteristics for the same acceleration and to the structural response parameters that relate to these characteristics

response spectrum - a curve calculated from an earthquake accelerogram that gives the value of peak response in terms of acceleration, velocity, or displacement of a damped linear oscillator (with a given damping ratio) as a function of its period (or

frequency)

review level earthquake (RLE) - an earthquake larger than the plant SSE and is chosen in SMA for initial screening purposes. Typically, the RLE is defined in terms of a ground motion spectrum. [Note: A majority of plants in the Eastern and Midwestern United States have conducted SMA reviews for an RLE of 0.3g pga anchored to a median NUREG/CR-0098 spectrum (Newmark and Hall, 1978).]

safe shutdown equipment list (SSEL) - The list of all SSCs that require evaluation in the seismic-fragilities task of an SMA (seismic margin assessment). Note that this list can be different from the SEL ("Seismic Equipment List") used in an SPRA (seismic probabilistic risk assessment.)

safety function - function that must be performed to control the sources of energy in the plant and radiation hazards

safety-related - structures, systems, and components that are relied upon to remain functional during and following design basis events to assure: *(1)* the integrity of the reactor coolant pressure boundary, *(2)* the capability to shut down the reactor and maintain it in a safe shut down condition; or *(3)* the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable exposures established by the regulatory authority

safety systems - those systems that are designed to prevent or mitigate a design-basis accident

screening analysis - an analysis that eliminates items from further consideration based on their negligible contribution to the frequency of a significant accident or its consequences

screening criteria - the values and conditions used to screen results to determine whether an item is a negligible contributor to the frequency of an accident sequence or its consequences

seismic equipment list (SEL) - the list of all SSCs that require evaluation in the seismic-fragilities task of an SPRA (seismic probabilistic risk assessment). Note that this list can be different from the SSEL ("Safe Shutdown Equipment List") used in an SMA (seismic margin assessment).

seismic margin - Seismic margin is expressed in terms of the earthquake motion level that compromises plant safety, specifically leading to severe core damage. The margin concept can also be extended to any particular structure, function, system, equipment item, or component for which "compromising safety" means sufficient loss of safety function to contribute to core damage either independently or in combination with other failures.

seismic margin assessment - the process or activity to estimate the seismic margin of the plant and to identify any seismic vulnerabilities in the plant.

seismic source - a general term referring to both seismogenic sources and capable tectonic sources. A seismogenic source is a portion of the earth assumed to have a uniform earthquake potential (same expected maximum earthquake and recurrence frequency), distinct from the seismicity of the surrounding regions. A capable tectonic source is a tectonic structure that can generate both vibratory ground motion and tectonic surface deformation such as faulting or folding at or near the earth's surface. In a PSHA, all seismic sources in the site region with a potential to contribute to the frequency of ground motions (i.e., the hazard) are considered.

seismic spatial interaction - an interaction that could cause an equipment item to fail to perform its intended safety function. It is the physical interaction of a structure, pipe, distribution system, or other equipment item with a nearby item of safety equipment caused by relative motions from an earthquake. The interactions of concern are (1) proximity effects, (2) structural failure and falling, and (3) flexibility of attached lines and cables.

severe accident - an accident that usually involves extensive core damage and fission product release into the reactor vessel, containment, or the environment

shall - used to state a mandatory requirement

should - used to state a recommendation

success path (as used in Seismic Margin Assessments; see Section 3.5) - a set of components that can be used to bring the plant to a stable hot or cold condition and maintain this condition for at least 72 hours

support system - a system that provides a support function (e.g., electric power, control power, or cooling) for one or more other systems

spectral acceleration - pseudo-absolute response spectral acceleration, given as a function of period or frequency and damping ratio (typically 5%). It is equal to the peak relative displacement of a linear oscillator of frequency f attached to the ground, times the quantity $(2\pi f)^2$. It is expressed in g or cm/s^2.

system failure - termination of the ability of a system to perform any one of its designed functions. Note: Failure of a line/train within a system may occur in such a way that the system retains its ability to perform all its required functions; in this case, the system has not failed.

systems analysis - that portion of the external-events PRA analysis that applies to

evaluating the impact of external events within the plant PRA model. In this context, the term "systems analysis" encompasses the tasks related to identification of the SSCs to be included in the analysis, event sequence modeling, analysis of the failure of individual system functions within the sequences, and the integration and quantification of the overall PRA model.

uncertainty - a representation (usually numerical) of the state of knowledge about data, a model, or process, usually associated with random variability of a parameter, lack of knowledge about data, a model, or process, or imprecision in the model or process

uncertainty (as used in seismic-fragility analysis) - the variability in the median seismic capacity arising from imperfect knowledge about the models and model parameters used to calculate the median capacity

uniform hazard response spectrum - a plot of a ground response parameter (for example, spectral acceleration or spectral velocity) that has an equal likelihood of exceedance at different frequencies

verify - to determine that a particular action has been performed in accordance with the rules and requirements of this standard, either by witnessing the action or by reviewing records

walkdown - inspection of local areas in a nuclear power plant where structures, systems, and components are physically located in order to ensure accuracy of procedures and drawings, equipment location, operating status, and environmental effects or system interaction effects on the equipment which could occur during accident conditions. For seismic-PRA and seismic-margin-assessment reviews, the walkdown is explicitly used to confirm preliminary screening and to collect additional information for fragility or margin calculations.

SECTION 3 ---- PRA TECHNICAL REQUIREMENTS

3.1 Scope

This Section provides requirements for each of the elements that comprise an external events PRA. As discussed previously (see Section 1.3), "The PRA scope covered by this Standard is limited to analyzing accident sequences initiated by external events that might occur while a nuclear power plant is at nominal full power. It is further limited to requirements for (i) a full Level 1 analysis of the core-damage frequency (CDF) and (ii) a limited Level 2 analysis sufficient to evaluate the large early release frequency (LERF)."

3.2 Fidelity: Plant vs. PRA

It is important that the PRA reasonably reflect the actual as-built, as-operated nuclear power plant being analyzed. Several mechanisms are used to achieve this fidelity between plant and analysis. One key mechanism is called "plant familiarization." During this phase, plant information is collected and examined. This involves (i) information sources, including design information, operational information, maintenance information, and engineering information, and (ii) plant walkdowns, both inside and outside the plant. Later, if the PRA is modified, it remains important to assure that fidelity is preserved, and hence further plant-familiarization work is necessary.

Throughout both this Standard and the ASME internal-events, full-power PRA standard (ASME, 2000) with which this standard is coordinated, requirements can be found whose objective is to assure fidelity between plant and analysis. Because external events PRAs depend critically on plant walkdowns, both inside and outside the plant, to ascertain the physical configurations of important SSCs and the environments they are exposed to, this standard places special emphasis on <u>walkdowns</u>, through requirements in the relevant sections dealing with SSC fragilities due to earthquakes (see Sections 3.4.3 and 3.5 below), the sections dealing with other external events (see Sections 3.6 through 3.9), and the section dealing with peer review (Section 5).

3.3 Technical Requirements - General

For each technical element that comprises an external events PRA, this Standard includes both High Level Requirements and Supporting Requirements.

The High Level Requirements are a set of requirements that encompass beneath them all of the Supporting Requirements. The High Level Requirements are general in their language, in recognition of the diversity of approaches that have been used to develop

the existing industry PRAs and the need to allow for technological innovations in the future. Highly prescriptive High Level Requirements are judged undesirable, and perhaps even unworkable. These High Level Requirements are intended to be used by both the PRA analyst team and the peer-review team (see Section 5, "PRA Peer Review.")

The High Level Requirements and the Supporting Requirements, taken together, are formulated in a way that is intended to support the applications being considered. Specifically, a PRA can meet the High Level Requirements and Supporting Requirements at various levels-of-detail and various scopes, that need not extend beyond what is adequate to support the intended application.

3.4 Seismic PRA – Technical Requirements

The technical requirements for seismic PRA have been developed based on a wealth of experience over the past twenty years, including a very large number of full-scope seismic PRAs for nuclear power plants, and a large number of methodology guidance documents and methodology reviews. Appendix A contains a short introduction and review of the seismic-PRA methodology. Other useful references include (NRC, 1983), (Brookhaven, 1985), (Cummings, 1986), (Bohn and Lambright, 1988), (Reed and Kennedy, 1994), and (Budnitz, 1998). The earliest important guidance on seismic PRA methods is described in (SSMRP, 1981), (Shieh et al., 1985), and (Cummings, 1986). The proceedings of an international conference sponsored by the Nuclear Energy Agency in Tokyo (OECD-NEA, 1999) contain a number of methodological advances. The principal guidance on seismic-hazard analysis is in (Budnitz et al., 1997) and (Reiter, 1990). The major elements of a seismic PRA are seismic hazard analysis, systems analysis including quantification, and fragility evaluation. The technical requirements for each of these are given in the following.

Seismic PRA is an integrated activity requiring close interactions among specialists from different fields (for example, seismic hazard analysis, systems analysis, and fragility evaluation). Although the methodology for seismic PRA and the supporting data have evolved and advanced over the past twenty years, the analysis still requires judgment and extrapolation beyond observed data. Therefore, the analyst is strongly urged to review published seismic PRA reports and to compare his/her plant-specific seismic PRA to the published studies of similar reactor types and system designs. This will promote consistency among similar PRAs and risk informed applications, and will also promote reasonableness in the numerical results and risk insights. The peer review is also directed in part toward this same objective.

3.4.1 SEISMIC PRA: TECHNICAL REQUIREMENTS FOR PROBABILISTIC SEISMIC HAZARD ANALYSIS

3.4.1.1 INTRODUCTION

Requirements for the probabilistic seismic hazard analysis (PSHA) address two situations. The first situation deals with cases where no prior study exists and the site specific PSHA must be generated anew. In the second situation, the PSHA analyst may have the option to use an existing study to form the basis for a site-specific assessment. For example, the Lawrence Livermore National Laboratory and Electrical Power Research Institute regional hazard studies (NRC, 1993 and EPRI, 1989) for east of the Rocky Mountains can be used to develop site-specific PSHA for most of the CEUS sites.

As discussed in the high-level requirement HLR-HA-H below, these stidues and many hazard studies conducted for plant-specific PRAs are considered to meet the overall

requirements of this Standard, subject to any updating as necessary.

The primary objective of the PSHA for most sites is to estimate the probability or frequency of exceeding different levels of vibratory ground motion, and the requirements described in this standard address this objective in detail. If site conditions make it necessary to include other seismic hazards, such as fault displacement, landsliding, soil liquefaction, soil settlement, and earthquake-induced flooding, the objective is similar -- to estimate the probability or frequency of either hazard occurrence as a function of its size, intensity, and/or hazard consequences.

The "level" (complexity and efforts related to use of expert judgment, expert elicitation, integration, etc.) of hazard analysis depends on two primary considerations: (1) intended use of the SPRA; and (2) the complexity of seismic environment. The NRC/EPRI/DOE Senior Seismic Hazard Analysis Committee's so-called "SSHAC" report (Budnitz et.al., 1997) lists the following factors which affect the choice of level for the hazard analysis.

- The significance of the issue to the final results of the PSHA.
- The issue's technical complexity and level of uncertainty.
- The amount of technical contention about the issue in the technical community.
- Important non-technical considerations such as budgetary, regulatory, scheduling, or other concerns.

Based on considerations of the above and other factors, the SSHAC report has identified and provided guidance for four levels of hazard analysis.

The detailed description of these four levels is contained in the SSHAC report (Budnitz et.al., 1997). While basic constituent elements of a PSHA are the same in all applications, the SSHAC levels are roughly in order of increasing resources and sophistication. It is important, ultimately, to show that the PSHA characterization is robust for the intended application and accounts for the uncertainties.

3.4.1.2 HIGH-LEVEL REQUIREMENTS – PROBABILISTIC SEISMIC HAZARD ANALYSIS

The ANS 2.29 and 2.27 standards (ANS, 1997 and ANS, 2000), both currently in draft form, will be governing documents which will provide detailed requirements and guidance to perform the probabilistic seismic hazard analysis. The intent of this standard is to reflect these requirements at a higher level and put them in the context of

an SPRA and intended applications of the SPRA.

There are ten High Level Requirements for Probabilistic Seismic Hazard Analysis, as follows:

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT A -- SCOPE
(HLR-HA-A): The frequency of earthquakes at the site SHALL be based on a site-specific probabilistic seismic hazard analysis (PSHA) (existing or new) that reflects the composite distribution of the informed technical community. The level of analysis SHALL be determined based on the intended application and on site-specific complexity.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT B - DATA COLLECTION
(HLR-HA-B): To provide inputs to the PSHA, a comprehensive up-to-date data base including: regional geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties, SHALL be compiled. A catalogue of historical, instrumental, and paleoseismicity SHALL also be compiled.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT C - SEISMIC SOURCES AND SOURCE CHARACTERIZATION
(HLR-HA-C): To account for the frequency of occurrence of earthquakes in the site region, the PSHA SHALL consider all credible sources of potentially damaging earthquakes. Both the aleatory and epistemic uncertainties SHALL be considered in characterizing the seismic sources.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT D - GROUND MOTION CHARACTERIZATION
(HLR-HA-D): The PSHA SHALL account for all credible mechanisms influencing estimates of vibratory ground motion that can occur at a site given the occurrence of an earthquake of a certain magnitude at a certain location. Both the aleatory and epistemic uncertainties SHALL be considered in characterizing the ground motion propagation.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT E - LOCAL SITE EFFECTS
(HLR-HA-E): The PSHA SHALL account for the effects of local site response.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT F - AGGREGATION AND QUANTIFICATION
(HLR-HA-F): Uncertainties in each step of the hazard analysis SHALL be propagated and displayed in the final quantification of hazard estimates for the site. The results SHALL include fractile hazard curves, median and mean hazard curves, and uniform hazard response spectra (UHS). For certain applications, the PSHA SHALL include

seismic source deaggregation and magnitude-distance deaggregation.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT G - SPECTRAL SHAPE (HLR-HA-G): For further use in the SPRA, the spectral shape SHALL be based on a site-specific evaluation taking into account the contributions of deaggregated magnitude-distance results of the PSHA. Broad-band, smooth spectral shapes, such as those presented in NUREG/CR-0098 (Newmark and Hall, 1978) (for lower-seismicity sites such as most of those east of the U.S. Rocky Mountains) may also be used taking into account the site conditions. The use of UHS may also be appropriate if it reflects the site-specific shape.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT H - USE OF EXISTING STUDIES
(HLR-HA-H): When use is made of an existing study for PSHA purposes, it SHALL be confirmed that the basic data and interpretations are still valid in light of current information, the study meets the requirements outlined in A through G above, and the study is suitable for the intended application.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT I - OTHER SEISMIC HAZARDS
(HLR-HA-I): A screening analysis SHALL be performed to assess whether, in addition to the vibratory ground motion, other seismic hazards, such as fault displacement, landslide, soil liquefaction, or soil settlement need to be included in the SPRA for specific application. If so, the SPRA SHALL address the effect of these hazards through assessment of probability or frequency of either hazard occurrence and/or hazard consequences.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT J- DOCUMENTATION (HLR-HA-J): The PSHA SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

3.4.1.3 SUPPORTING TECHNICAL REQUIREMENTS -- PROBABILISTIC SEISMIC HAZARD ANALYSIS

The Supporting Technical Requirements for "Probabilistic Seismic Hazard Analysis" follow:

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT A: SCOPE

(HLR-HA-A): The frequency of earthquakes at the site SHALL be based on a site-specific probabilistic seismic hazard analysis (PSHA) (existing or *new)* that reflects the composite distribution of the informed technical community. The level of analysis SHALL be determined based on the intended application and on site-specific complexity.

NOTE: The need for determining the composite distribution is discussed in (Budnitz et al., 1997.)

REQUIREMENT	COMMENTARY
(REQ. HA-A1) The PSHA SHALL be based on and consist of collection and evaluation of available information and data; evaluation of the uncertainties in each element of the PSHA; and a defined process and documentation to make the PSHA traceable and transparent.	**NOTE (HA-A1):** The guidance and process given in (Budnitz et.al., 1997) and in (ANS, 1997) addresses the above requirement, and either of these MAY be usedb as an acceptable methodology. Existing LLNL (NRC, 1993) and EPRI (EPRI, 1989) hazard studies and many hazard studies conducted for plant-specific PRAs also meet this overall requirement, subject to updating as necessary. (See Requirement HLR-HA-H below.)
(REQ. HA-A2) The spectral accelerations, or the average spectral acceleration over a selected band of frequencies, SHALL be used as the parameter to characterize both hazard and fragilities. The selection of frequencies to determine spectral accelerations or average spectral acceleration SHALL capture the	**NOTE (HA-A2):** While the use of peak ground acceleration as a parameter to characterize both hazard and fragility has been a common practice in the past and is acceptable, the use of spectral accelerations is preferable.

frequencies of those SSCs that are of interest and are dominant contributors to the PRA results and insights. The use of peak ground acceleration is also acceptable.	
(REQ. HA-A3) The PSHA results, whether they are characterized by spectral accelerations, peak ground accelerations or both, SHALL extend to high enough values (consistent with the physical data and interpretations) so that the final numerical results, such as core damage frequency, reflect accurate estimates of risk, and delineation and ranking of seismic-initiated sequences are not affected.	**NOTE (HA-A3):** It is necessary to make sure that the hazard estimation is carried out to large enough values (consistent with the physical data and interpretations) so that when convolved with the plant or component level fragility, the resulting failure frequencies are robust estimates, and do not change if the acceleration range is extended. A sensitivity study can be conducted to define the upper bound value.
(REQ. HA-A4) A lower bound magnitude SHALL be specified for use in the hazard analysis such that earthquakes of magnitude less than this value are not expected to cause significant damage to the engineered structures or equipment.	**NOTE HA-A4:** The value of the lower bound magnitude used in analyzing the site-specific hazard is based on engineering considerations (EPRI, 1988). Based on the evaluation of earthquake experience data, earthquakes with magnitudes less than 5.0 are not expected to cause damage to safety-related structures, systems, and components. A lower bound magnitude value of 5.0 was used for both LLNL and EPRI studies. Note that this lower bound only applies to the magnitude range considered in the final hazard quantification, not to the characterization and determination of seismicity parameters for the sources. The choice of magnitude scale is left at the discretion of the analyst, but whichever magnitude scale is used should be documented.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT B: DATA COLLECTION

(HLR-HA-B): To provide inputs to the PSHA, a comprehensive up-to-date data base including: regional geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties SHALL be compiled. A catalogue of historical, instrumental, and paleoseismicity SHALL also be compiled.

REQUIREMENT	COMMENTARY
(REQ. HA-B1) The PSHA	**NOTE HA-B1:** It is important that a comprehensive database is shared and

SHALL develop or be based on a comprehensive geological, seismological, and geophysical database that reflects the current state-of-the-knowledge, and that is used by experts/analysts to develop interpretations and inputs to the PSHA.	used by all experts in developing the interpretations. The availability of the data base also facilitates the review process. (NRC, 1997a) and (EPRI, 1986) give acceptable guidance on the scope and types of data required for use in the seismic source characterization, ground motion modeling, and local site response evaluations to meet this requirement.
(REQ. HA-B2) The size of the region to be investigated and the scope of investigations SHALL be adequate to characterize all credible seismic sources that may contribute to the frequency of occurrence of vibratory ground motion at a site, considering regional attenuation of ground motions and local site effects. If the existing PSHA studies are to be used in the SPRA, the investigations SHALL be of sufficient scope to determine whether there are new data or interpretations that are not adequately incorporated in the existing data bases and analysis.	**NOTE HA-B2:** (NRC, 1997a) defines four levels of investigations, with the degree of their detail based on distance from the site, the nature of the Quaternary tectonic regime, the geological complexity of the site and region, the existence of potential seismic sources, the nature of sources, the potential for surface deformation, etc.. This guidance can be used to determine scope and size of region for investigations. The guidance in (NRC, 1997b) may be used to meet this requirement.
(REQ. HA-B3) As a part of data collection, a catalog of historically reported, geologically identified, and instrumentally recorded earthquakes SHALL be compiled. ANS 2.29 (1997) and ANS 2.27 (ANS, 2000) requirements or equivalent SHOULD be met.	**NOTE HA-B3:** In general, the catalog typically includes events of size MMI Intensity (or equivalent) greater than or equal to IV and magnitude greater than or equal to 3.0 that have occurred within a radius of 320 km of a site (NRC, 1997b). For the earthquakes listed, the catalog typically contains information such as event date and time, epicentral location, earthquake magnitudes (measured and calculated), magnitude uncertainty, uncertainty in the event location, epicentral intensity, intensity uncertainty, hypocentral depth, references, and data sources.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT C: SEISMIC SOURCES AND SOURCE CHARACTERIZATION

(HLR-HA-C): To account for the frequency of occurrence of earthquakes in the site region, the PSHA SHALL consider all credible sources of potentially damaging earthquakes. Both the aleatory and epistemic uncertainties SHALL be considered in characterizing the seismic sources.

REQUIREMENT	COMMENTARY
(REQ. HA-C1) The PSHA SHALL consider all potential sources of earthquakes that affect the probabilistic hazard at the site. Identification and characterization of seismic sources SHALL be based on regional and site geological and geophysical data, historical and instrumental seismicity data, the regional stress field, and geological evidence of prehistoric earthquakes.	**NOTE HA-C1:** A useful reference is (NRC, 1997a).
(REQ. HA-C2) The expert elicitation process to characterize the seismic sources SHALL be compatible with the level of analysis discussed in Req. HA-A4, and SHALL follow a structured approach.	**NOTE HA-C2:** Guidance given in (Budnitz et.al., 1997) is one acceptable way to meet this requirement.
(REQ. HA-C3) The seismic sources are characterized by: source location and geometry, maximum earthquake magnitude, and earthquake recurrence. The aleatory and epistemic uncertainties in these characterizations SHALL be addressed in accordance with the level of analysis identified for REQ. HA-A4.	**NOTE HA-C3:** While in some applications, the explicit display of the uncertainties or the distinction between aleatory or epistemic uncertainties (see Definition Section and Appendix A to this standard for brief explanations of these terms) in the final results may not be necessary, it is essential in the PSHA to characterize the uncertainties properly, so as to make the process transparent and results interpretable. Uncertainties in the hazard estimates dominate the uncertainties in the final SPRA results, and it is therefore important to understand the sources and nature of these uncertainties in making application decisions. (Budnitz et.al., 1997) gives detailed discussion and acceptable guidance on a process to be used for determination and quantification of uncertainties to meet this requirement. A National Research Council Committee has reviewed (Budnitz et al., 1997) and that review is in an Appendix to (Budnitz et al., 1997.)

(REQ. HA-C4) If an existing PSHA study is used, any seismic sources that were previously unknown or uncharacterized SHALL be shown to be not significant or SHALL be included in the update of the hazard estimates.	**NOTE HA-C4:** (NRC, 1997a) gives detailed guidance on how to assess the significance of new information including new interpretations, and this is one acceptable method. Specific case studies were also conducted by the industry during NRC's revision to the 10CFR Part 100 siting rules. These studies are referred to in (NRC, 1997a).

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT D: GROUND MOTION CHARACTERIZATION

(HLR-HA-D): The PSHA SHALL account for all credible mechanisms influencing estimates of vibratory ground motion that can occur at a site given the occurrence of an earthquake of a certain magnitude at certain location. Both the aleatory and epistemic uncertainties SHALL be considered in characterizing the ground motion propagation.

REQUIREMENT	COMMENTARY
(REQ. HA-D1) The PSHA SHALL account for all credible mechanisms governing estimates of vibratory ground motion that can occur at a site, and SHALL take into account regional and site-specific geological, geophysical, and geotechnical data and historical and instrumental seismicity data (including strong motion data).	**NOTE HA-D1:** It is important to note that in the guideline documents (Budnitz et.al., 1997, NRC, 1997a, and ANS, 1997), the probabilistic seismic hazard estimates are first performed for the real or assumed rock conditions in the free-field. For the non-rock sites, the site-specific estimates are performed taking into account the local site conditions and properties including aleatory and epistemic uncertainties as discussed under HLR-HA-E.
(REQ. HA-D2) The expert elicitation process to characterize the ground motion SHALL be compatible with the level of analysis discussed in Req. HA-A4, and SHALL follow a structured approach.	**NOTE-HA-D2:** The structured approach given in (Budnitz et.al., 1997) is one acceptable way to meet this requirement.
(REQ. HA-D3) The aleatory and epistemic uncertainties	**NOTE HA-D3:** The characterization of ground motion includes: the equation (attenuation relationship) that predicts the median level of ground motion

in the ground motion characterization SHALL be addressed in accordance with the level of analysis identified for REQ. HA-A4.	parameter of engineering interest (spectral acceleration, displacements, pga, etc.) as a function of magnitude and distance; an estimate of the aleatory variability in ground motion which quantifies the unexplained scatter in ground motion and the event-to-event variability of earthquakes of the same magnitude; and epistemic uncertainty in the ground motion model. As discussed in HA-D3, it is necessary to properly characterize uncertainties in the hazard estimates. (Budnitz et.al., 1997) gives guidance on an acceptable process to be used for determination and quantification of uncertainties.
(REQ. HA-D4) If an existing PSHA study is used, any ground motion models or new information that were previously unused or unknown SHALL be shown to be not significant or SHALL be included in the update of the hazard estimates.	**NOTE HA-D4:** (NRC, 1997a) gives detailed guidance on how to assess the significance of the new information including new interpretations.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT E: LOCAL SITE EFFECTS

(HLR-HA-E): The PSHA SHALL account for the effects of local site response.

REQUIREMENT	COMMENTARY
(REQ. HA-E1) The PSHA SHALL account for the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site.	**NOTE HA-E1:** The purpose of a local site response analysis is to quantify the influence of surficial geologic conditions on site ground motions. Two approaches are generally used to account for surficial conditions at a site as part of the estimation of ground motion. The first is to utilize ground motion attenuation relationships appropriate for the site conditions, i. e., relationships that have been developed for the type of subsurface conditions that exist at a site. The second is to develop site-specific transfer functions that can be used to modify the rock ground motions for the site characteristic (ANS, 1997, NRC, 1997b). The existing PSHA studies should be shown to account for the local site effects or should be updated.
(REQ. HA-E2) Both the aleatory and epistemic uncertainties SHALL be considered in the local site response analysis.	**NOTE HA-E2:** Consistent with the source characterization and ground motion estimates, it is essential that the uncertainties are properly characterized and propagated in this step.

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT F: AGGREGATION AND QUANTIFICATION

(HLR-HA-F): Uncertainties in each step of the hazard analysis SHALL be propagated and displayed in the final quantification of hazard estimates for the site. The results SHALL include fractile hazard curves, median and mean hazard curves, and uniform hazard response spectra (UHS). For certain applications, the PSHA SHALL include seismic source deaggregation and magnitude-distance deaggregation.

REQUIREMENT	COMMENTARY
(REQ. HA-F1) The final quantification of the seismic hazard SHALL include and display propagation of both aleatory and epistemic uncertainties.	**NOTE HA-F1:** The seismic hazard quantification involves the combination of seismic source and ground motion inputs to compute the frequency of exceedance of ground motions at a site (i.e., the seismic hazard). Thus, the principal result of the PSHA is a set of seismic hazard curves that quantify the aleatory and epistemic uncertainties in the site hazard. This is typically presented in terms of a set of fractile seismic hazard curves, defined at specified fractile levels, and the mean hazard. Two acceptable approaches have been used to propagate epistemic uncertainties: logic tree enumeration and Monte Carlo simulation (EPRI, 1989 and Bernreuter et.al., 1989).
(REQ. HA-F2) The PSHA SHALL include appropriate sensitivity studies and intermediate results to identify factors that are important to the site hazard and that make the analysis traceable and reviewable.	**NOTE HA-F2:** Sensitivity studies and intermediate results provide important information to reviewers about how some of the key assumptions affect the final results of this complex seismic-hazard process. Examples of useful sensitivity studies include an evaluation of alternate schemes used to assign weights to the individual expert models, and an evaluation of the way different experts make different assignments of the regional seismicity to different zonation maps.
(REQ. HA-F3) The following results SHALL be developed as a part of the quantification process, compatible with needs for the level of analysis determined in HLR-HA-A: o Fractile hazard curves for each ground motion parameter considered in the PSHA; o Median and mean hazard curves for peak ground	**NOTE HA-F3:** (ANS, 1997) is the basis for this requirement. The magnitude-distance deaggregation and seismic source deaggregation (McGuire, 1995) are useful when the application of the SPRA depends on the quantitative results and full understanding of sources of uncertainties is essential. These aspects become important when relative comparisons are to be made among risks resulting from different initiators. The magnitude-distance deaggregation helps in identifying the earthquake events (magnitude and distance) which dominate the hazard. This in turn, allows the analyst to properly characterize the nature of ground motion for use in the response and fragility analyses. Fractile curves are generally plotted for the 5, 15, 50, 85, and 95 percentiles. The UHS provides hazard information (probability of exceedance) for spectral acceleration at several discrete frequencies.

acceleration andspectral accelerations; o Fractile and mean UHS; o Magnitude-distance deaggregation for the median and mean hazard; o Seismic source deaggregation; o Mean magnitude and distance.	

SEISMIC HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT G: SPECTRAL SHAPE

(HLR-HA-G): For further use in the SPRA, the spectral shape SHALL BE based on a site-specific evaluation taking into account the contributions of deaggregated magnitude-distance results of the PSHA. Broad-band, smooth spectral shapes, such as those presented in NUREG/CR-0098 (Newmark and Hall, 1978) (for lower-seismicity sites such as most of those east of the U.S. Rocky Mountains) may also be used taking into account the site conditions. The use of UHS may also be appropriate if it reflects the site-specific shape.

REQUIREMENT	COMMENTARY
(REQ. HA-G1) The response spectral shape used in the SPRA SHALL be based on site-specific evaluations performed for the PSHA, and, at a minimum, SHALL reflect or bound the characteristics of spectral shapes associated with the mean magnitude and distance pairs determined in the PSHA for the important ground motion levels.	**NOTE HA-G1:** The issue of which spectral shape should be used in the screening of structures, systems, and components (SSCs) and in quantification of SPRA results requires careful consideration. For screening purposes, the spectral shape used should have amplification factors such that the demand resulting from the use of this shape is higher than that based on the design spectra. This will preclude premature screening of components and will avoid anomalies such as the screened components (e.g., surrogate elements) being the dominant risk contributing components. Additional discussion on this issue can be found in (Kennedy, 1999). In the quantification of fragilities and of final risk results, it is important to use as realistic a shape as possible, and specifically a shape which reflects dominant magnitude-distance events taking into account the site-specific conditions. Other semi-site specific shapes, such as those given in NUREG-0098, have been used in the past and are considered adequate for this purpose. The UHS may also be appropriate for this purpose if they reflect the spectral shape of dominating events.

SPRA HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT H: USE OF EXISTING STUDIES

(HLR-HA-H): When use is made of an existing study for PSHA purposes, it SHALL be confirmed that the basic data and interpretations are still valid in light of current information, the study meets the requirements outlined in A through G above, and the study is suitable for the intended application.

[There are no Supporting Requirements here.]	**NOTE HA-H:** When using the LLNL/NRC (NRC, 1993) and/or EPRI (EPRI, 1989) hazard studies, or another study done to a comparable technical level, the intent of this requirement is not to repeat the entire hazard exercise or calculations, unless new information and interpretations affect the usefulness of the SPRA for the intended application.

SPRA HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT I: OTHER SEISMIC HAZARDS

(HLR-HA-I): A screening analysis SHALL be performed to assess whether, in addition to the vibratory ground motion, other seismic hazards, such as fault displacement, landslide, soil liquefaction, or soil settlement need to be included in the SPRA for specific application. If so, the SPRA SHALL address the effect of these hazards through assessment of probability or frequency of either hazard occurrence and/or hazard consequences.

[There are no Supporting Requirements here.]

SPRA HAZARD ANALYSIS HIGH-LEVEL REQUIREMENT J: DOCUMENTATION

(HLR-HA-J): The PSHA SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

REQUIREMENT	COMMENTARY
(REQ. HA-J1) The documentation SHALL meet the general documentation requirements in Section 7.	

(REQ. HA-J2) The documentation SHALL, in general, meet the guidelines of (NRC, 1997a) for PSHA, including a description of the specific methods used for source characterization and ground-motion characterization, and of the scientific interpretations that are the basis for the inputs and results. If an existing PSHA is used, its documentation SHOULD be checked to assure that it is adequate to meet the spirit of the requirement here.

3.4.2 SEISMIC PRA TECHNICAL REQUIREMENTS FOR SYSTEMS ANALYSIS

3.4.2.1 INTRODUCTION

It is assumed in the systems-analysis requirements contained herein that the seismic-PRA analysis team possesses a full-scope internal-events full-power Level 1 and Level-2-LERF PRA, developed either prior to or concurrently with the seismic PRA. It is further assumed that this internal-events PRA is then used as the basis for the seismic-PRA systems analysis. If these assumptions are not valid, then such a PRA must be developed before the seismic-PRA systems-analysis work can proceed.

It is also assumed that the technical basis for the internal-events full-power PRA is the ASME PRA standard (ASME, 2000).

Systems analysis for seismic PRA generally consists of both adding some earthquake-related basic events to the internal-events systems model, and also "trimming" some aspects of that model that do not apply or can be screened out on a sound basis. Examples of trimming include eliminating the part of the model covering recovery from loss of offsite power, which is usually not feasible after a large earthquake; eliminating event trees that start with very unlikely events unrelated to earthquakes; and screening out of low-probability non-seismic failures and human-error events. Thus the seismic-PRA systems model is generally substantially simpler than the corresponding model for internal events, even though it also contains some added complexity related to earthquake-caused failures.

In special circumstances, it is acceptable to develop an ad-hoc systems model tailored especially to the seismic-PRA situation being modeled, instead of starting with the internal-events model and adapting it. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects. Whichever approach is used, either adapting the internal-events systems model or building an ad-hoc systems model, it is important that the systems model includes all important failures, including both failures caused by the earthquake and non-seismic failures and human errors.

3.4.2.2 HIGH-LEVEL REQUIREMENTS -- SYSTEMS ANALYSIS

There are six High Level Requirements for Systems Analysis, as follows:

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT A -- COMPLETENESS (HLR-SA-A): The seismic-PRA systems models SHALL include all important seismic-caused initiating events that can lead to core damage or large early release, and

SHALL include all other important failures that can contribute significantly to CDF or LERF, including seismic-induced SSC failures, non-seismic-induced unavailabilities, and human errors.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT B -- ADAPTATIONS BASED ON THE INTERNAL-EVENTS PRA SYSTEMS MODEL
(HLR-SA-B): The seismic-PRA systems model SHALL be adapted to incorporate seismic-analysis aspects that are different from corresponding aspects found in the full-power, internal-events PRA systems model.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT C -- PLANT FIDELITY
(HLR-SA-C): The seismic-PRA systems models SHALL reflect the as-built and as-operated plant being analyzed.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT D -- SEISMIC EQUIPMENT LIST
(HLR-SA-D): The list of SSCs selected for seismic-fragility analysis SHALL include all SSCs that participate in accident sequences included in the seismic-PRA systems model.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT E -- INTEGRATION AND QUANTIFICATION
(HLR-SA-E): The analysis to quantify CDF and LERF frequencies SHALL appropriately INTEGRATE the seismic hazard, the seismic fragilities, and the systems-analysis aspects.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT F -- DOCUMENTATION
(HLR-SA-F): The seismic-PRA analysis be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

3.4.2.3 SUPPORTING TECHNICAL REQUIREMENTS – SYSTEMS ANALYSIS

The Supporting Technical Requirements for "SPRA Systems Analysis" follow.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT A: COMPLETENESS

(HLR-SA-A): The seismic-PRA systems models SHALL include all important seismic-caused initiating events that can lead to core damage or large early release, and SHALL include all other important failures that can contribute significantly to CDF or LERF, including seismic-induced SSC failures, non-seismic-induced unavailabilities, and human errors.

REQUIREMENT	COMMENTARY
(REQ. SA-A1) The systems analysis SHALL assure that all important earthquake-caused initiating events are included in the seismic-PRA systems model.	**NOTE SA-A1:** It is very important that site-specific failure events, usually earthquake-caused structural, mechanical, and electrical failures, be thoroughly investigated. The usual list of seismic-caused initiating events considered in seismic PRAs includes, for example, (i) failure of the RPV or of another very large component such as a steam generator, a recirculation pump, or the pressurizer; (ii) LOCAs of various sizes and in all relevant locations; and (iii) transients, of which loss of offsite power (LOSP) is usually the most important. There are two general types of transients that should be considered, those in which the power conversion system (PCS) or heat-transport system has failed as a direct consequence of the earthquake (for example, following LOSP), and those in which the PCS is initially available. Other types of transient initiating events include, for example, losses of key support systems such as service water or DC power. Also, multiple-unit impacts and dependencies should be considered, as appropriate, including recovery resources that could be affected by a large earthquake. Attention to both the core-damage-frequency (CDF) endpoint and the large-early-release-frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement.
(REQ. SA-A2) In the initiating-event selection process, a hierarchy SHALL be developed to assure that every earthquake greater than a certain defined size produces a plant shutdown within the systems model.	**NOTE SA-A2:** It is generally a requirement at all nuclear reactor stations that any earthquake larger than a certain size -- usually defined as the operating-basis earthquake – will require the plant to shut down (terminate the chain reaction and move toward a safe, stable shutdown state) to reduce energies that may cause LOCAs and to enable inspection for possible earthquake-caused damage. The purpose of the IE hierarchy is to assure that, given an earthquake that exceeds this threshold, the sum total of all of the initiating-event conditional probabilities adds to unity (100%). If this means that a manual-shutdown sequence must be added to account for those circumstances when no automatic post-earthquake shutdown will occur, then such manual actions must be added to the systems model. Usually, this involves adding these manual-shutdown sequences to the group of transients in which the PCS is initially available. The order of the hierarchy is usually defined so that, if one earthquake-caused IE occurs, the occurrence of other IEs down the hierarchy is of no significance in terms of the systems model. Thus, for example, if the earthquake causes a large LOCA, there is no concern in the systems model for the simultaneous occurrence of a small LOCA. Implicit in the IE hierarchy is the notion that basic failure events which define an IE cannot occur in the accident sequences corresponding to IEs lower in the hierarchy, so as to avoid duplication within the sequence modeling. For example, a failure of the reactivity-control function (control rod failure) usually is modeled so that it can occur as a basic event in sequences in which a large-LOCA is modeled as the IE, but not vice-versa -- when seismic-caused control-rod failure is modeled as the IE, large-LOCAs are not included there. If the seismically-caused-IE hierarchy is constructed logically, the various types of sequences will automatically conform to this hierarchy.
(REQ. SA-A3) The systems analysis SHALL assure that the PRA systems models reflect all important	**NOTE SA-A3:** The event trees and fault trees from the internal-events full-power PRA model are generally used as the basis for the seismic event trees. This is done both to capture the thinking that has gone into their development, and to assist in allowing comparisons between the internal-events PRA and the seismic PRA to be made on a common basis. [As mentioned in the text in Section 3.4.2.1, considerable screening out and "trimming" of the internal-

earthquake-caused failures and all important non-seismic-induced unavailabilities and human errors.	events PRA systems model is also common where appropriate. The lumping of certain groups of individual components into so-called "supercomponents" in the systems model is also a valid approximation in many situations.] In special circumstances, it is acceptable to develop an ad-hoc systems model tailored especially to the seismic-PRA situation being modeled, instead of starting with the internal-events model and adapting it. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects. Earthquakes can cause failures that are not explicitly represented in the internal-events models, primarily (but not exclusively) due to damage to structures and other passive items such as distribution systems (electrical raceways, piping runs, ductwork, instrument tubing, etc.), vessels, large tanks, and all supports and anchorage and spatial interactions, that can then affect safety functions. The principal challenge in meeting this requirement is assuring that these passive-failure events are included. Other categories of seismic-induced failures that are typically not modeled in the internal-events PRA are seismic-induced relay-chatter and related events (see REQ. SA-B5 below), and seismic-caused damage that can block personnel access to safety equipment or controls, thereby inhibiting manual operability actions, in either the control room or another location that might otherwise be credited (see REQ. SA-B7 below). The principal way in which the seismic-PRA trees differ from those used in internal-events PRA analysis, besides adding in the passive SSCs, is the need to consider the physical locations and proximity of SSCs. This need exists both because secondary failures such as spatial interactions must be considered – this aspect is usually taken into account in the seismic walkdowns – and because response correlations can be important and are related to co-location of similar items. After the seismic-capacity-engineering work has been accomplished, the systems analysis needs to introduce response correlations into the models where appropriate. Introducing these aspects into the systems analysis can be done in any of several different ways: basic events can be added directly to the fault trees and the "gates" appropriately modified; or an event (such as liquefaction or building failure) that globally affects an entire safety function or accident sequence can be added directly to the Boolean expression; or linked event trees can be used along with a "seismic pre-tree" with associated conditional split fractions in the plant-response part of the model; or the (stronger) fragility definition of an SSC can be redefined in terms of the (weaker) fragility of another SSC whose failure can cause the undesired failure of the stronger SSC. Sometimes, the knowledge that a given SSC is very rugged to resist earthquakes can save the systems analysis team the work of developing a model that includes that SSC's failure. This may be true, for example, of certain structures, pressure-retaining components, or piping and duct runs. Thus a round of iteration with the seismic-capacity-engineering aspect of the seismic PRA can be useful when the systems-analysis work is underway. The SSCs to be considered in this aspect include both SSCs that can act as (or contribute to) seismic initiating events, and SSCs that appear as nodes in event trees or as basic events in fault trees. Attention to both the core-damage-frequency (CDF) endpoint and the large-early-release-frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT B: ADAPTATIONS BASED ON THE INTERNAL-EVENTS PRA SYSTEMS MODEL

(HLR-SA-B): The seismic-PRA systems model SHALL be adapted to incorporate seismic-analysis aspects that are different from corresponding aspects found in the full-power, internal-events PRA systems model.

NOTE: While the most common procedure for developing the seismic-PRA systems model is to start with the internal-events systems model and adapt it by adding and trimming, in special circumstances it is acceptable instead to develop an ad-hoc seismic-PRA systems model tailored especially to the situation being modeled. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects. See Section 3.4.2.1 and also the NOTE at REQ. SA-A3 for further commentary.

REQUIREMENT	COMMENTARY
(REQ. SA-B1) In each of the following aspects of the seismic-PRA systems-analysis work, the corresponding requirements in the ASME internal-events, full-power PRA standard (ASME, 2000) SHALL be satisfied, except where they are not applicable, or where this Standard includes additional requirements. A defined basis SHALL be developed to support the claimed non-applicability of any exceptions. The aspects governed by this requirement are: (1) Initiating-event analysis (2) Accident-sequence analysis (3) Success-criteria analysis (4) Systems analysis (5) Data analysis (6) Human-reliability analysis (7) Use of expert judgment.	**NOTE SA-B1:** These sections of the ASME standard are effectively incorporated here by reference. A few aspects, however, do not apply in detail. Whenever an exception is taken, the PRA analyst team needs to be cognizant of the underlying rationale for the specific ASME requirement, so as to assure that this rationale is considered when the exception is taken.
(REQ. SA-B2) In the HRA	**NOTE SA-B2:** In many seismic PRAs, the human error probabilities are increased for some post-earthquake actions, compared to the probabilities

(human reliability analysis) aspect, the analysis SHALL consider that additional post-earthquake stresses can increase the likelihood of human errors or inattention, compared to the likelihood assigned in the internal-events HRA when the same activities are undertaken in non-earthquake accident sequences. If increases in error probabilities are not used, the basis for this decision SHALL be justified.	assigned in analogous internal-events-initiated sequences. The rationale is usually that strong seismic motions can adversely affect human performance shortly after a very large earthquake. However, the basis for determining these increases is not well developed in the seismic-PRA literature, and several different seismic-HRA models are in use. (Of course, this factor has reduced importance to the extent that most modern nuclear-power plants have designs and procedures that do not require operator intervention for the first half-hour or more after a postulated earthquake.) This aspect can represent an important source of uncertainty in the numerical results of a seismic PRA.
(REQ. SA-B3) The analysis SHALL be performed so that any screening of SSCs appropriately accounts for seismic-caused dependencies and correlations.	**NOTE SA-B3:** It is vital that the analysis capture the important correlations among seismic-caused failures. Of course, this is generally true in all PRAs, but because the earthquake will affect all SSCs at the same time with the same incoming motion, special care must be taken on this subject when performing a seismic PRA. [See (REQ. SA-E6) where the requirement to deal with dependencies and correlations in the integration/quantification is covered, and (REQ. SA-E7) where appropriate sensitivity analyses are required to explore these issues.] Some papers at the OECD/NEA Workshop in Tokyo (OECD/NEA, 1999) provide useful discussion and guidance on this issue. One reasonable approach to take is to assume 100% response correlation as an initial assumption. If the issue of correlation then seems to make a difference to the overall results or insights, one can do a sensitivity analysis by assuming zero response correlation, to ascertain how important the correlation might be. If there is a major difference, the analyst must then attempt to determine just what the best assumption really is for treating the correlation. The screening-out step must be done conservatively, because once an SSC is screened out it is "lost" from the rest of the analysis. Before SSCs are screened out on what is an otherwise-well-defined basis, it is important to check that possible correlations do not invalidate the screening-out step. This requirement is intended to capture this practice. An acceptable method for this screening is found in (Bohn and Lambright, 1988), which provides more detail for an approach similar to that described above. REQ. SA-E1, SA-E6, and SA-E7 have additional requirements and commentary about dependencies and correlations.
(REQ. SA-B4) The analysis SHALL be performed so that any screening of human-error basic events and non-seismic-failure basic events does not significantly affect the PRA's results.	**NOTE SA-B4:** To make the systems-analysis models more manageable, it is common practice to screen out some of the non-seismic failures and human errors from the model if their contribution to the results is demonstrably very small. One acceptable approach to accomplish this screening is given in NUREG/CR-5679 (Budnitz, Moore, and Julius, 1992).

(REQ. SA-B5) In the systems analysis, the effects of the chatter of relays and similar devices SHALL be considered.	**NOTE SA-B5:** The analysis of relay and contactor chatter has become a standardized part of seismic PRA, and several reports and guidance documents exist (Budnitz, Lambert, and Hill, 1987; Hardy and Ravindra, 1990; MPR, 1990; Merz, 1991). After the list of relays and contactors that are involved in key safety functions has been developed, it is usually more efficient to screen out those with very high seismic capacities, or whose chatter will not affect the proper execution of a safety function, before including the others in the systems model. Typically, only a small subset of the relays and contactors survive these screening-out steps. (Hardy and Ravindra, 1990) provides an acceptable methodology for performing this aspect of the analysis. One acceptable method for meeting this requirement is to demonstrate that a relay evaluation has fully followed the NRC's IPEEE guidance (NRC, 1991; NRC, 1991a), applicable to the specific plant and site.
(REQ. SA-B6) In the systems-analysis models, each basic event that represents a seismically-caused failure SHALL include the complementary "success" state where applicable to a particular SSC.	**NOTE SA-B6:** At intermediate earthquake levels, many SSCs whose earthquake-caused failure is important to safety at higher levels will not fail, or will fail with only modest probability. The modeling of the non-failure (that is, the "success") of such SSCs is an important aspect of the systems model, and excluding these "success" states can lead to erroneous PRA results.
(REQ. SA-B7) The systems analysis SHALL consider the possibility that a large earthquake can cause damage that blocks personnel access to safety equipment or controls, thereby inhibiting operator actions that might otherwise be credited.	**NOTE SA-B7:** This information is most effectively gathered during the walkdown, which must be structured to search for access issues. Coordination with the human-reliability analysis aspect of the PRA is important. If access problems are identified, the systems model needs to be modified, so as to assign the (weaker) seismic fragility of the failure causing the access problem to each (presumably stronger) SSC to which access in thereby impaired. In making these evaluations, it MAY be assumed that portable lighting is available and that breathing devices are available for confined spaces, if in fact the plant configuration includes them.
(REQ. SA-B8) The systems analysis SHALL consider the likelihood that system recoveries modeled in the internal-events PRA may be more complex or even not possible after a large earthquake, and SHALL adjust the recovery models accordingly.	**NOTE SA-B8:** The restoration of safety functions can be inhibited by any of several types of causes; these include damage or failure, access problems, confusion, loss of supporting staff to other post-earthquake-recovery functions, and so on. Careful consideration of these must be given before recoveries are credited in the initial period after a large earthquake. This is especially true for earthquake-caused loss of offsite power, given that the damage could be to switchyard components or to the offsite grid towers, which are generally difficult to fix quickly. While this Standard does not require the analyst to assume an unrecoverable loss of offsite power after a large earthquake, the general practice in seismic PRAs has been to make such an assumption.
(REQ. SA-B9) The systems analysis SHALL consider including an earthquake-caused "small-small LOCA" as an additional fault within	**NOTE SA-B9:** It is almost never feasible in a seismic-PRA walkdown to evaluate every small impulse line connected to the primary circuit, whose failure in an earthquake could cause a so-called "small-small LOCA" (a leak with an area from one to a few square-centimeters) in the primary circuit. Furthermore, breaks in one or a very few such lines cannot otherwise be precluded, given the large number of such lines and their unusual

each sequence in the seismic-PRA model.	configurations in many cases. Therefore, it is a common (although not a universal) practice in seismic PRAs to include such a small-small LOCA as an additional assumed fault in every accident sequence, in addition to whatever other failures are modeled. This has the effect of making "success" (that is, reaching a safe stable state) in those sequences dependent on the availability of at least enough make-up water to the primary system to replace the inventory loss at high pressure from such a break. This requirement is intended to assure that adding such a small-small-LOCA basic event to each relevant accident sequence is considered, and is done unless a justification for omitting such can be supported.
(REQ. SA-B10) The SPRA walkdown SHALL include the potential for seismic-induced fires and flooding following the guidance given in NUREG-1407.	**NOTE SA-B10**: Normally, if the walkdown team identifies a potential seismic-induced-fire issue or seismic-induced-flooding issue, the issue should be reviewed carefully by the power-plant staff, and either dismissed on a defined basis or remedied if appropriate. Extensive experience with seismic PRAs at U.S. nuclear plants indicates that only rarely is the PRA analysis team faced with the task of quantifying a CDF or LERF for these types of scenarios using a full seismic-fire-PRA analysis, but if so then this analysis must quantify the hazard, the fragilities, and the systems-analysis aspect as in any other aspect of the SPRA. The walkdown that supports this aspect should be linked with the walkdown that examines seismic spatial interactions. (See both the High Level Requirement and the Supporting Requirements under HLR-FR-E.) NUREG-1407 (NRC, 1991a) contains acceptable guidance on how to do this evaluation.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT C: PLANT FIDELITY

(HLR-SA-C): The seismic-PRA systems models SHALL reflect the as-built and as-operated plant being analyzed.

REQUIREMENT	COMMENTARY
(REQ. SA-C1) To assure that the systems-analysis models reflect the as-built, as-operated plant, any important conservatisms or other distortions introduced SHALL be justified by demonstrating that they do not significantly alter the seismic-PRA's validity for applications.	

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT D: SEISMIC EQUIPMENT LIST

(HLR-SA-D): The list of SSCs selected for seismic-fragility analysis SHALL include all SSCs that participate in accident sequences included in the seismic-PRA systems model.

REQUIREMENT	COMMENTARY
(REQ. SA-D1) The PRA systems model SHALL be used as the basis for developing the SEL ("Seismic Equipment List"), which is the list of all SSCs to be considered by the subsequent seismic-fragility engineering task.	**NOTE SA-D1:** The SEL is the basic starting point for the work of the seismic-fragility task. As such, its development is usually a product of interactive thinking among the systems-analysis and seismic-fragility-evaluation members of the PRA team. Its development is also heavily dependent upon the scope and quality of the seismic-walkdown activity, the requirements for which are covered elsewhere in this standard. (See both the High Level Requirement and the Supporting Requirements under HLR-FR-E.) The starting point for constructing the SEL is the internal-events PRA model, to which must be added a number of SSCs with earthquake-specific issues. Attention to both the core-damage-frequency (CDF) endpoint and the large-early-release-frequency (LERF) endpoint is necessary to meet this requirement. It is advisable to compare the SEL for reasonableness with comparable SEL lists compiled for seismic PRAs at other similar nuclear power plants.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT E: INTEGRATION AND QUANTIFICATION

(HLR-SA-E): The analysis to quantify core-damage frequency (CDF) and large-early-release frequency (LERF) SHALL appropriately integrate the seismic hazard, the seismic fragilities, and the systems-analysis aspects.

REQUIREMENT	COMMENTARY
(REQ.SA-E1) In the quantification of CDF and LERF, the integration aspect SHALL be performed using an established methodology.	**NOTE SA-E1:** The integration step is where the various earlier and supporting parts of the seismic PRA are brought together and integrated to produce and quantify the final results, in terms of CDF and LERF, and in terms of identifying the "important contributors." Seismic-PRA practitioners possess different tools to accomplish this integration and quantification. Analysts usually use an iterative process, in which an interim and approximate quantification is done, after which certain parts of the overall systems model are screened out on the basis that they do not contribute importantly to the results. The quantification is then finalized. Seismic screening of an SSC can be done on the basis that its seismic capacity is very strong, so that it does not contribute importantly to any seismic-induced accident sequences, above some defined cutoff level. Screening of a non-seismic failure or of a human-error basic event in the model can be done on the basis that its contribution to any seismic-induced accident sequences is below a defined cutoff. Whatever the basis for the screening (see the Supporting Requirements below on this subject), that basis must be defined, and the selection of a cutoff should be done very carefully. While details vary, the typical systems-analysis approach is to add seismic-related basic events (or sometimes entire new "branches") to the internal-events fault tree models that are adapted from the internal-events-PRA Level 1 and Level 2-LERF analysis. Considerable screening out or "trimming" of the event trees and fault trees is also a common practice. The quantification then typically consists of a series of hazard-specific quantifications: the model is quantified several times for a range of different hazard intervals, and these quantifications are then summed. In this approach, for each hazard interval and for each SSC/basic event, the hazard, response, and fragility analyses are integrated to produce a "probability of seismic-induced failure" -- actually a distribution of the analyst's state-of-knowledge of that probability, taking into

	account the uncertainties in hazard, response, and fragility. This probability is then inserted into the relevant fault tree, which is solved. Typically, each fault tree is solved separately, and then these are integrated into the relevant event tree(s) to produce a set of accident-sequence-specific values for "core damage frequency" conditional on the hazard interval being evaluated. (Other methods are also in use in which the integration over the hazard is not done on a fault-tree-specific basis, but rather at the event-tree level; logically the outcome should be the same.) The one issue that requires great care is the treatment of seismic-related dependencies/correlations among the seismic failures: in particular (i) the linking of the various basic events to capture their correlated failures, and (ii) the screening out of SSCs and other non-seismic basic events in light of these correlations/dependencies (see the Supporting Requirements SA-B3, SA-E6, and SA-E7on these subjects.) The relevant seismic correlations/dependencies arise, of course, because in a given earthquake event every SSC in the plant is exposed to the exact same earthquake input motion (although modified -- amplified, damped, frequency-shifted, etc. -- as the earthquake energy propagates from the earth below the site to the location of the SSC at issue.). There are a number of different approaches in use to treat these correlations/dependencies, and this standard does not single out any one of them. Acceptable methods can be found in (Bohn and Lambright, 1988) and (PG&E, 1988).
(REQ. SA-E2) In quantifying CDF and LERF frequencies, the quantification SHALL be performed on a cut-set-by-cut-set or accident-sequence-by-accident-sequence basis (or for defined groups of these), as well as on a comprehensive/integrated basis.	**NOTE SA-E2:** The intent of this requirement is to assure that key information about each accident sequence (or cut set) is retained, rather than simply "lost" in the production of overall integrated values for CDF and LERF. Of course, it is common to group cut-sets of accident sequences when they are so similar that phenomenologically they cannot be distinguished very well; such grouping is entirely acceptable, if its basis is defined.
(REQ. SA-E3) In the quantification, the integration over the seismic hazard SHALL extend to a sufficiently large hazard range to capture the principal contributions to the overall results. The integration SHALL utilize hazard intervals that are fine enough to avoid distorting the results.	**NOTE SA-E3:** The postulated earthquakes that can affect any given site range over a wide range of "size," from more frequent smaller earthquakes to very infrequent larger ones. The intent of the first requirement here is to assure that the quantification does not "cut off" at an earthquake size so low that important contributions to CDF or LERF are not captured. (Here "important" is to be interpreted in light of the existing uncertainties in the analysis, and thus needs to be defined for each specific analysis; no general rules exist.) The intent of the second requirement is to assure that the earthquake-hazard "bins" used in the integration process are also selected to assure that important insights are not lost or distorted -- again, this needs to be evaluated for each specific analysis; no general rules can be given. This is an issue that deserves special attention from the peer-review team.
(REQ. SA-E4) The analysis SHALL use the quantification process to assure that any screening of	**NOTE SA-E4:** SSC screening -- the elimination from the model of SSCs -- is done throughout the process of performing any PRA. A defined set of criteria must be developed and used to assure that this screening does not eliminate elements of the model that should have been retained. The intent of this requirement is to assure that the quantification process is used to check that

SSCs does not affect the results, taking into account the various uncertainties.	the screening has not erroneously eliminated important SSCs. It is recognized that this type of work is an iterative process, in which approximate interim quantifications are done during which the screening decisions are checked, and only then is a final quantification done. There are many different approaches in current use among seismic-PRA analysts to accomplish this step. (Bohn and Lambright, 1988) contains a useful discussion on this aspect.
(REQ. SA-E5) The integration/quantification analysis SHALL account for all important dependencies and correlations that affect the results.	**NOTE SA-E5:** As discussed earlier, treating earthquake-specific correlations and dependencies properly is vital to achieving a successful seismic-PRA. This requirement is intended to assure that this issue is covered. A discussion of this type of correlation/dependency analysis is found in (Bohn and Lambright, 1988). See (REQ. SA-B3) where the requirement to deal with dependencies and correlations in initial screening is covered, and (REQ. SA-E7) where appropriate sensitivity analyses are required to explore these issues.
(REQ. SA-E6) The integration/quantification analysis SHALL account for the uncertainties in CDF and LERF results that arise from each of the several inputs (the seismic hazard, the seismic fragilities, and the systems-analysis aspects.)	**NOTE SA-E6:** All seismic-PRA is characterized by large numerical uncertainties, not only in the seismic-hazard aspect but in the seismic-capacity and systems-analysis aspects as well. Examples of other analysis areas where uncertainties arise in seismic PRA that are different from those encountered in internal-events PRA are the human-reliability analysis aspect, the issue of earthquake-caused correlations/dependencies, relay chatter, and the recovery analysis. It is essential that estimates of the uncertainties in the analysis team's state-of-knowledge about each aspect be developed, and that these be carried through to be incorporated quantitatively into the integration/quantification step. Experience has shown that to do otherwise can produce "results" that may be not be relied on in terms of both overall insights and the details. Also note that the requirement to "account for" the various uncertainties recognizes that not all of them must necessarily be quantified explicitly, especially if they are small. (See also the comment below at NOTE SA-F3.) There are numerous methods in current use to accomplish this requirement, ranging from numerical-integration schemes to schemes that approximate the various empirical distributions by well-defined analytical forms (such as log-normal forms) which are more amenable to numerical integration.
(REQ. SA-E7) Appropriate sensitivity studies SHALL be performed to illuminate the sensitivity of the CDF and LERF results to the assumptions used about dependencies and correlations.	**NOTE SA-E7:** A concern with seismic PRA today is that the overall state-of-knowledge about the amount of dependency/correlation among earthquake-induced SSC failures is limited. Specifically, when similar items are co-located (for example, adjacent), the analyst typically will assume full response correlation, whereas if SSCs are quite different or found in very different locations then the typical assumption is to assign small or zero correlation. Due to the broad range of variables in the types of SSCs, and the available test or experience data, there may not be high confidence in estimating correlation. Thus it is standard practice among seismic PRA analysts to perform sensitivity analyses to test how much difference emerges in the final PRA "results" when different amounts of correlation are assigned. This requirement is intended to capture this practice. See (REQ. SA-B3) where the requirement to deal with dependencies and correlations in the initial screening is covered, along with a discussion of sensitivity analyses; and (REQ. SA-E6), covering the integration/quantification aspect. This is an issue that deserves special attention from the peer-review team.

SYSTEMS-ANALYSIS HIGH-LEVEL REQUIREMENT F: DOCUMENTATION

(HLR-SA-F): The seismic-PRA analysis SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

REQUIREMENT	COMMENTARY
(REQ. SA-F1) The documentation SHALL meet the general documentation requirements in Section 7.	
(REQ. SA-F2) The documentation SHALL describe the specific adaptations made in the internal-events PRA model to produce the seismic-PRA model, and their motivation.	
(REQ. SA-F3) The documentation SHALL describe the major contributors to the uncertainties in each of the important final results and insights of the systems analysis.	**NOTE SA-F3:** While many of these uncertainties must necessarily be expressed in terms of numerical distributions of the analysis team's state-of-knowledge about a numerical result, not all of them must be expressed in such numerical terms. (Also see the comment under NOTE SA-E6. As in SA-E6, , which uses the words "SHALL account for", the words "SHALL describe" here imply a recognition that not all of the various uncertainties must necessarily be quantified explicitly, especially if they are small. But this requirement does ask for a description of each of the important uncertainties.)

3.4.3 SEISMIC PRA TECHNICAL REQUIREMENTS FOR SEISMIC FRAGILITY ANALYSIS

The seismic fragility of a structure, system or component is defined as the conditional probability of its failure at a given value of seismic motion parameter (e.g., peak ground acceleration, peak spectral acceleration at different frequencies, or floor spectral acceleration at the equipment frequency). The methodology for evaluating seismic fragilities of SSC is documented in the PRA Procedures Guide (NRC, 1983) and is more specifically described for application to nuclear power plants in (Kennedy and Ravindra, 1984) and (Reed and Kennedy,1994). Appendix A provides a brief description of how seismic fragility curves are developed for any SSC. Seismic fragilities used in a seismic PRA should be realistic and plant specific based on actual conditions of the SSCs in the plant, as confirmed through a detailed walkdown of the plant. Seismic fragility evaluation has been conducted for over 40 nuclear power plants in the United States and other countries. Based on the experience and insights gained in these studies, certain methodological improvements and simplifications have been proposed in (Kennedy, 1999).

Note that in performing a seismic PRA, the seismic fragility evaluation is performed before the integration and quantification that are the subjects of HLR-SA-E. Thus the order of the Requirements herein is different than the order in which the analysis work must be performed.

3.4.3.1 HIGH-LEVEL REQUIREMENTS – SEISMIC FRAGILITY EVALUATION

There are seven High Level Requirements under Seismic Fragility Evaluation, as follows:

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT A -- REALISM (HLR-FR-A): The seismic fragility evaluation SHALL be performed to estimate plant-specific, realistic seismic fragilities of structures, systems and components whose failure may contribute to core damage and/or large early release.

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT B -- SCREENING (HLR-FR-B): If screening of high-seismic-capacity components is performed, the basis for the screening SHALL be fully described.

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT C -- RESPONSE (HLR-FR-C): The seismic fragility evaluation SHALL be based on realistic seismic response that the SSCs experience at their failure levels. Depending on the site conditions and response analysis methods used in the plant design, realistic seismic response MAY be obtained by an appropriate combination of scaling, new analysis and new structural models.

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT D – FAILURE MODES (HLR-FR-D): The seismic fragility evaluation SHALL be performed for critical failure modes of structures, systems and components such as structural failure modes and functional failure modes identified through the review of plant design documents, supplemented as needed by earthquake experience data, fragility test data, generic qualification test data, and a walkdown.

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT E -- WALKDOWN (HLR-FR-E): The seismic fragility evaluation SHALL incorporate the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support, and potential systems interactions.

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT F -- DATA SOURCES (HLR-FR-F): The calculation of seismic fragility parameters such as median capacity and variabilities SHALL be based on plant specific data supplemented as needed by earthquake experience data, fragility test data and generic qualification test data. Use of such generic data SHALL be justified.

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT G -- DOCUMENTATION (HLR-FR-G): The seismic fragility evaluation SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

3.4.3.2 SUPPORTING TECHNICAL REQUIREMENTS – SEISMIC FRAGILITY EVALUATION

The Supporting Technical Requirements for SPRA Seismic Fragility Evaluation follow.

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT A:

REALISM

(HLR-FR-A): The seismic fragility evaluation SHALL be performed to estimate plant-specific, realistic seismic fragilities of structures, systems and components whose failure may contribute to core damage and/or large early release.

REQUIREMENT	COMMENTARY
(REQ. FR-A1) Seismic fragilities SHALL be developed for all those structures, systems and components identified by	**NOTE FR-A1:** Seismic fragilities are needed for all those structures, systems and components (SSC) identified by the systems analysis that are modeled in the event trees and fault trees. Failure of one or more of these may contribute to core damage and/or large early release. Requirements for developing this list of SSC are given under the systems analysis section (see REQ. SA-D1).

the systems analysis (see REQ SA-D1).	
(REQ. FR-A2) Seismic fragilities SHALL be based on plant-specific data and SHALL be realistic (median with uncertainties). Generic data (e.g., fragility test data, generic seismic qualification test data and earthquake experience data) MAY be used for screening of certain SSC. However, any use of such generic data SHALL be demonstrated to be conservative.	**NOTE FR-A2:** The objective of a seismic PRA is to obtain a realistic seismic risk profile for the plant using plant specific and site-specific data. It has been demonstrated in several seismic PRAs that the risk estimates and insights on seismic vulnerabilities are very plant specific, even varying between supposedly identical units at a multi-unit plant. In order to minimize the effort on non-significant items and to focus the resources on the more critical aspects of the seismic PRA, certain-high-seismic capacity components are screened out using generic data (e.g., fragility test data, generic seismic qualification test data and earthquake experience data). It is important to be conservative in the use of such generic data.

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT B: SCREENING

(HLR-FR-B): If screening of high seismic capacity components is performed, the basis for the screening SHALL be fully described.

REQUIREMENT	COMMENTARY
(REQ. FR-B1) If screening of high-seismic-capacity components is performed, the basis for screening and supporting documents SHALL be fully described. For example, guidance given in EPRI NP-6041 and NUREG/CR-4334 MAY be used to screen out components with high seismic capacity. However, the screening level SHALL be chosen high enough that the contribution to CDF and LERF from the screened-out components is demonstrably not significant.	**NOTE FR-B1:** When screening of high seismic capacity components is performed, the basis for screening and supporting documents are to be fully described. Guidance given in EPRI NP-6041 (EPRI, 1991) and NUREG/CR-4334 (Budnitz et al., 1985) may be used to screen out high seismic capacity components after satisfying the caveats. Note that the screening guidance in these documents has been developed generally for US-vendored equipment and based on US seismic design practice. Care should be used in applying the screening criteria for other situations. The use of generic fragility information is acceptable for screening if the specific SSC can be shown to fall within the envelope of the generic fragility caveats. The screening level chosen should be based on the seismic hazard at the site and the plant seismic design basis, and should be high enough that the contribution to CDF and LERF from the screened out components is not significant. For a discussion of possible approaches to the selection of the screening level, the reader is referred to (Reed and Kennedy, 1994) and (Kennedy, 1999).

(REQ. FR-B2) The applicability of the screening criteria given in EPRI NP-6041 (EPRI, 1991) and NUREG/CR-4334 (Budnitz et al., 1985) SHALL be assessed and documented for the specific plant and specific equipment. Note that the screening criteria do not apply to nuclear power plants in high seismic regions such as coastal California.

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT C: RESPONSE

(HLR-FR-C): The seismic fragility evaluation SHALL be based on realistic seismic response that the SSCs experience at their failure levels. Depending on the site conditions and response analysis methods used in the plant design, realistic seismic response MAY be obtained by an appropriate combination of scaling, new analysis and new structural models.

REQUIREMENT	COMMENTARY
(REQ. FR-C1) Seismic responses that the components experience at their failure levels SHALL be estimated on a realistic basis using site-specific earthquake response spectra in three orthogonal directions, anchored to a ground motion parameter such as peak ground acceleration or average spectral acceleration. If site-specific spectra are not available, the use of generic spectral shapes such as NUREG/CR-0098 median spectrum SHALL be justified.	**NOTE FR-C1:** NUREG/CR-0098 is (Newmark and Hall, 1978). NUREG-1407 (NRC, 1991a) recommends the use of 10,000-year return period median spectral shapes provided in (Bernreuter et al., 1989) along with variability estimates, if site-specific spectral shapes are not available. However, such UHS SHOULD be used cautiously making sure that the spectral shape is sufficiently rich in low frequencies. See NOTE HA-G1 for further discussion on this topic.

(REQ. FR-C2) If probabilistic response analysis is performed to obtain realistic structural loads and floor response spectra, the number of simulations done (e.g., Monte Carlo simulation and Latin Hypercube Sampling) SHALL be large enough to obtain stable median and 85% non-exceedance responses. The response analysis SHALL appropriately take into account the entire spectrum of input ground motion levels displayed in the seismic hazard curves.

(REQ. FR-C3) If scaling of	**NOTE FR-C3:** The scaling procedures given in (EPRI, 1991) may be used.

existing design response analysis is used, it SHALL be justified based on the adequacy of structural models, foundation characteristics, and similarity of input ground motion.	
(REQ. FR-C4) For soil sites, or when the design response analysis models are judged not to be realistic and state-of-the-art, or when the design input ground motion is significantly different from the site-specific input motion, new analysis SHALL be performed to obtain realistic structural loads and floor response spectra.	
(REQ. FR-C5) If median-centered response analysis is performed, the median response (i.e., structural loads and floor response spectra) and variability in the response SHALL be estimated using established methods.	**NOTE FR-C5:** (Reed and Kennedy, 1994) gives an acceptable method.
(REQ. FR-C6) When soil structure interaction (SSI) analysis is conducted, it SHALL be median centered using median properties, at soil strain levels corresponding to the input ground motions that dominate the seismically induced core damage frequency. The uncertainties in the SSI analysis SHALL be considered by varying the low strain soil shear modulus between the median value times (1+C_v) and the median value divided by (1+C_v), where C_v is a factor that accounts for uncertainties in the SSI analysis and soil properties. If adequate soil investigation data are available, the mean and standard deviation of the low strain shear modulus SHALL	**NOTE FR-C6:** Further details about the basis of this requirement can be found in (ASCE, 1998).

be established for every soil layer. The value of C_v SHALL then be established so that it will cover the mean plus or minus one standard deviation for every layer. The minimum value of C_v SHALL be 0.5. When insufficient data are available to address uncertainties in soil properties, C_v SHALL be taken as no less than 1.0.	

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT D: FAILURE MODES

(HLR-FR-D): The seismic fragility evaluation SHALL be performed for critical failure modes of structures, systems and components such as structural failure modes and functional failure modes identified through the review of plant design documents, supplemented as needed by earthquake experience data, fragility test data, generic qualification test data, and a walkdown.

REQUIREMENT	COMMENTARY
(REQ. FR-D1) Realistic failure modes of structures and equipment that interfere with the operability of equipment during or after the earthquake SHALL be identified through review of the plant design documents and the walkdown.	**NOTE FR-D1:** Note that certain structural failures (for example, partial or complete collapse) MAY not be of much interest in the seismic PRA; the lower failure modes such as drift and yielding MAY be more relevant for the functionality of attached equipment.
(REQ. FR-D2) All relevant failure modes of structures (e.g., sliding, overturning, yielding, and excessive drift), equipment (e.g., anchorage failure, impact with adjacent equipment or structures, bracing failure, and functional failure) and soil (i.e., liquefaction, slope	**NOTE FR-D2:** Published references and past seismic PRAs MAY be used as guidance. Examples include (Reed and Kennedy, 1994); (EPRI, 1991); (PG&E, 1988).

instability, excessive differential settlement) SHALL be considered, and fragilities for critical failure modes SHALL be evaluated.	

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT E:

WALKDOWN

(HLR-FR-E): The seismic fragility evaluation SHALL incorporate the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support, and potential systems interactions.

REQUIREMENT	COMMENTARY
(REQ. FR-E1) A detailed walkdown of the plant SHALL be conducted focusing on equipment anchorage, lateral seismic support and potential systems interactions. The purposes of such a walkdown are to find as-designed, as-built, and as-operated seismic weaknesses in the plant and to ensure that the seismic fragilities are realistic and plant-specific.	**NOTE FR-E1:** The seismic walkdown is an important activity in the seismic PRA. It should be done in sufficient detail and documented in a sufficiently complete fashion so that the subsequent screening or fragility evaluation is transparent.
(REQ. FR-E2) The walkdown SHALL be conducted following the guidance consistent with that given in (EPRI, 1991) and (Budnitz et al., 1985). The walkdown procedures, walkdown team composition, walkdown observations and conclusions SHALL be documented.	
(REQ. FR-E3) If components are screened out during or following the walkdown, anchorage calculations or some other basis justifying such a screening SHALL be documented.	
(REQ. FR-E4) The walkdown SHALL focus on the potential for seismic induced fire and flooding following the guidance consistent with that given in NUREG-1407 (NRC, 1991a).	
(REQ. FR-E5) The walkdown SHALL examine potential sources of interaction (e.g., II/I issues,	**NOTE FR-E5:** A "II/I issue" refers to situations where a non-seismically qualified component could fall on and damage a seismically qualified component.

impact between cabinets, flooding and spray) and consequences of such interactions on equipment contained in the systems model.	

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT F:

DATA SOURCES

(HLR-FR-F): The calculation of seismic fragility parameters such as median capacity and variabilities SHALL be based on plant specific data supplemented as needed by earthquake experience data, fragility test data and generic qualification test data. Use of such generic data SHALL be justified.

REQUIREMENT	COMMENTARY
(REQ. FR-F1) Component seismic fragility parameters such as median capacity, and variabilities (logarithmic standard deviations reflecting randomness and uncertainty) SHALL be based on plant specific data supplemented as appropriate by earthquake experience data, fragility test data and generic qualification test data.	**NOTE FR-F1:** Typically, the seismic fragility of a component is characterized by a double lognormal model whose parameters are median capacity, β_R and β_U. β_R is the logarithmic standard deviation of the capacity and represents the variability due to the randomness of the earthquake characteristics for the same acceleration and to the structural response parameters which relate to these characteristics. β_U is the logarithmic standard deviation of the median capacity and represents the uncertainties in models and model parameters. For some applications, it MAY be sufficient to develop a mean fragility curve characterized by a lognormal probability distribution with parameters of A_m and β_c where $\beta_c = (\beta_R^2 + \beta_U^2)^{1/2}$ is the logarithmic standard deviation of composite variability. An approach suggested in (Kennedy, 1999) is to first calculate the High Confidence of Low Probability of Failure (HCLPF) capacity based on the Conservative Deterministic Failure Margin (CDFM) method. This HCLPF capacity is taken as the 1% conditional–probability-of-failure value and a generic β_c is estimated for typical SSC. Using these, the median capacity and hence the mean fragility curve are approximated. For further discussion on the uses and limitations of these approximations, refer to (Reed and Kennedy, 1994) and (Kennedy, 1999).
(REQ. FR-F2) All SSCs that appear in the dominant accident cutsets SHALL have site-specific fragility parameters which are derived based on plant-specific information, such as anchoring and installation of the component or structure and plant-specific material test data. Exception: The use of generic fragility for any SSC SHALL be justified as being realistic for the	**NOTE FR-F2:** The objective of the fragility analysis is to derive fragility parameters that are as realistic as possible. They SHOULD reflect the as-built conditions of the equipment and should use plant-specific information. Use of conservative fragilities would distort the contribution of the seismic events to CDF and LERF. Note that the use of conservative fragilities may underestimate the frequencies of some accident sequences involving "success' terms. Therefore, generic fragilities, if used, SHOULD NOT BE overly conservative and SHOULD BE realistic for the specific SSC.

plant.	
(REQ. FR-F3) Seismic fragilities SHALL be developed for relays identified to be essential and which are included in the systems analysis model (see REQ. SA-B5).	**NOTE FR-F3:** Guidance on evaluation of relay chatter effects is given in (EPRI, 1991), (NRC, 1991a), and (Hardy and Ravindra, 1991).
(REQ. FR-F4) Seismic fragilities SHALL be developed for SSCs that are identified in the systems model as playing a role in the LERF part of the SPRA analysis. (See REQ. SA-A1 and REQ. SA-A3.)	**NOTE FR-F4:** Generally the concern is the seismically-induced early failure of containment functions. NUREG-1407 (NRC, 1991a) describes these functions as containment integrity, containment isolation, prevention of bypass functions, and some specific systems depending on the containment design (e.g., igniters, suppression pools, or ice baskets).

SEISMIC FRAGILITY EVALUATION HIGH LEVEL REQUIREMENT G: DOCUMENTATION

(HLR-FR-G): The seismic fragility evaluation SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

REQUIREMENT	COMMENTARY
(REQ. FR-G1) The documentation SHALL meet the general documentation requirements in Section 7.	
(REQ. FR-G2) The documentation SHALL describe the methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions.	
(REQ. FR-G3) The documentation SHALL provide a detailed list of SSC fragility values that includes the method of seismic qualification, the dominant failure mode(s), source of information, and the location of the component. The fragility parameter values (i.e., median acceleration capacity, β_R and β_U) and the technical bases for them SHALL be provided for each analyzed SSC.	
(REQ. FR-G4) The documentation SHALL cover the different aspects of seismic fragility analysis, such as the seismic response analysis, the screening steps, the walkdown, the review of	**NOTE FR-G4:** The documentation requirements given in NUREG-1407 (NRC, 1991a) and followed in the Diablo Canyon Long Term Seismic Program (PG&E, 1988) and (Bohn and Lambright, 1988) studies MAY be used as guidance.

design documents, the identification of critical failure modes for each SSC, and the calculation of fragility parameter values for each SSC modeled.	

3.5 Seismic Margin Assessment – Technical Requirements

In the mid-1980s, two different methodologies for the seismic margin assessment of nuclear power plants were developed. These are the so-called "NRC method" (Budnitz et al, 1985; Prassinos, Ravindra, and Savy, 1986) and the so-called "EPRI method" (EPRI, 1991). The Requirements herein are explicitly directed toward an analysis using the EPRI method, which employs success-path-type systems-analysis methods.

If a seismic-margin assessment uses the NRC method, which employs fault-space-type systems-analysis methods similar to those used in seismic PRA, then only some of the Requirements in this Section 3.5 are applicable. Specifically, all of the High Level Requirements (and Supporting Requirements) apply except HLR-SM-B and HLR-SM-G. Instead, the Requirements in Section 3.4.2, covering the systems-analysis part of seismic PRA, must to be used. An NRC-method SMA meets this Standard by meeting the above combination of Requirements.

The technical requirements for seismic margin assessment have been developed based on the SMA methodology guidance developed by both EPRI (EPRI, 1991) and NRC (Budnitz et al., 1985; Prassinos, Ravindra, and Savy, 1986), plus the experience gained in performing several dozen SMA reviews for nuclear power plants. Other useful references include (Kennedy et al., 1989), (Reed and Kennedy, 1994), (NRC, 1991), (NRC, 1991a) (Budnitz, Murray, and Ravindra, 1992), (ERI, 1997), and (Kennedy, 1999).

3.5.1 HIGH LEVEL REQUIREMENTS - SEISMIC MARGIN ASSESSMENT

There are eight High Level Requirements under Seismic Margin Assessment, as follows:

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT A -- REVIEW LEVEL EARTHQUAKE (HLR-SM-A): A review level earthquake characterized by a ground motion spectrum SHALL be selected to facilitate screening of structures, systems and components and performance of seismic margin calculations.

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT B -- SUCCESS PATHS (HLR-SM-B): A minimum of two diverse success paths SHALL be developed consisting of structures and equipment that can be used to bring the plant to a safe stable state and maintain this condition for at least 72 hours following an earthquake larger than the RLE.

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT C -- RESPONSES (HLR-SM-C): Seismic responses calculated for the Review Level Earthquake SHALL be median centered, SHALL be based on current state-of-the-art methods of structural modeling, and SHALL include the effects of soil-structure-interaction where applicable.

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT D -- SCREENING (HLR-SM-D): The screening of components and subsequent seismic margin calculations SHALL incorporate the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support and potential spatial interactions

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT E -- FAILURE MODES (HLR-SM-E): Seismic margin calculations SHALL be performed for critical failure modes of structures, systems and components such as structural failure modes and functional failure modes identified through the review of plant design documents, including analysis and test reports supplemented by earthquake experience data, fragility test data, generic qualification test data, and by a walkdown.

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT F CALCULATIONS (HLR-SM-F): The calculation of seismic margins (or so-called HCLPF capacities) SHALL be based on plant specific-data supplemented by earthquake experience data, fragility test data and generic qualification test data. Use of such generic data SHALL be justified.

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT G --SUCCESS PATH MARGINS (HLR-SM-G): The plant seismic margin SHALL be reported based on the margins calculated for the success paths.

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT H -- DOCUMENTATION (HLR-SM-H): The Seismic Margin Assessment SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

3.5.2 SUPPORTING TECHNICAL REQUIREMENTS – SEISMIC MARGIN ASSESSMENT

The Supporting Technical Requirements for Seismic Margin Assessment follow:

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT A
REVIEW LEVEL EARTHQUAKE

(HLR-SM-A): A review level earthquake characterized by a ground motion spectrum SHALL be selected to facilitate screening of structures, systems and components and performance of seismic margin calculations.

REQUIREMENT	COMMENTARY

(REQ. SM-A1) A Review Level Earthquake (RLE) SHALL be selected as an earthquake larger than the Safe Shutdown Earthquake (SSE) for the plant.	**NOTE SM-A1:** The seismic margins methodology is designed to demonstrate sufficient margin over the SSE to ensure plant safety and to find any "weak links" that might limit the plant capability to safely withstand a seismic event larger than the SSE. The review level earthquake is used to screen components based on generic seismic capacity. Screening is done in an SMA to optimize the resources needed and to focus attention on more critical and potentially seismically weak components. (EPRI, 1991) contains useful guidance on the selection of the RLE. The seismic margin method typically utilizes two review or screening levels geared to peak ground accelerations of 0.3g and 0.5g. Based on the guidance given in NUREG-1407 (NRC, 1991a), most plants in the Central and Eastern United States have selected 0.3g peak ground acceleration as the RLE for their SMAs. For some sites where the seismic hazard is judged to be low (i.e., less than 10^{-4} per year at SSE), a reduced-scope margin assessment relying mainly on a walkdown has been considered acceptable. NUREG-1407 further states that an RLE of 0.5g should be used for sites in the Western United States except for the California coastal sites, for which the seismic-margin methodology is not acceptable.
(REQ. SM-A2) The Review Level Earthquake SHALL be characterized by a ground motion spectrum appropriate for the site conditions.	**NOTE SM-A2:** Based on the guidance in NUREG-1407 (NRC, 1991a), seismic margin assessments have been done using the 5% damped NUREG/CR-0098 (Newmark and Hall, 1978) median rock or soil spectrum anchored at 0.3g or 0.5g (depending on the RLE for the site). Alternative approaches for selecting the RLE spectrum are described in (EPRI, 1991). The shape of the RLE ground motion spectrum is needed to develop seismic responses of structures and equipment for the calculation of seismic margins.

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT B SUCCESS PATHS

(HLR-SM-B): A minimum of two diverse success paths SHALL be developed consisting of structures and equipment that can be used to bring the plant to a safe stable state and maintain this condition for at least 72 hours following an earthquake larger than the RLE.

REQUIREMENT	COMMENTARY
(REQ. SM-B1) A primary success path and an alternative success path SHALL be selected; one of the paths SHALL be capable of mitigating a small LOCA. Success paths SHALL include systems whose function is to prevent severe core damage and their support systems.	**NOTE SM-B1:** A set of components that can be used to bring the plant to a stable hot or cold condition and maintain this condition for at least 72 hours is known as a "success path." Based on the selected success paths, a SSEL (Safe Shutdown Equipment List) is then developed for subsequent screening, walkdown, and margin evaluation. It is advisable to compare the SSEL for reasonableness with comparable SSEL lists compiled for seismic margin assessments at other similar nuclear power plants.

(REQ. SM-B2) Success paths SHALL be those for which there is a high likelihood of an adequate seismic margin, SHALL be compatible with plant operating procedures, and SHALL have acceptable operational reliability.	**NOTE SM-B2:** It is desirable that, to the maximum extent possible, the alternative path involves operational sequences, systems, distribution systems (i.e., piping, raceways, duct and tubing), and components different from those used in the primary path. (EPRI, 1991) contains useful guidance on the selection of success paths, on the use of Success Path Logic Diagrams in their selection, and on how "acceptable operational reliability" is defined for the SMA review.
(REQ. SM-B3) Offsite power SHALL be assumed to have failed and to be unrecoverable during the 72-hour period of interest following the RLE.	**NOTE-SM-B3:** Earthquake experience has shown that offsite power is almost always lost after any earthquake larger than the SSE. Because of the potential damage to the electric grid and the region surrounding the plant, it is judged that the offsite power may not be recovered for up to 72 hours. Therefore the selected success paths should be able to provide core cooling and decay heat removal for at least 72 hours following the earthquake, without recourse to offsite power. Although no credit for offsite power is taken in the SMA, one also must be aware of possible adverse effects if offsite power remains available or is restored. In the internal event PRA, the analyst assumes that there would be a successful scram given the loss of offsite power. The probability of mechanical binding of control rods is deemed low, hence there is no need to examine if the reactor protection system will function. However, in the case of SMA, the analyst should verify if the reactor protection system works and if the control rod could drop given the potential for seismic induced deformation of the reactor internals and failure of the control rod drive mechanism. Further, the power conversion system (e.g., the main condenser) SHOULD be assumed as not available for heat sink function and any equipment powered by non-vital AC is also considered unavailable.
(REQ. SM-B4) The SMA SHALL analyze at least seismically initiated transient events and small seismically induced primary coolant leakage events (referred to as "small LOCA").	**NOTE SM-B4:** A detailed walkdown within the containment, to verify that all small instrumentation or impulse lines can withstand the RLE and that there are no potential spatial interactions resulting in their failure to add up to an area of 25 mm diameter, would lead to excessive radiation exposure of the walkdown team. Therefore, it is considered prudent and expedient to concede that a small LOCA will occur after an RLE, and to include the required mitigation systems in the success path (see REQ. SA-B9).
(REQ. SM-B5) If one element in the Success Path Logic Diagram (SPLD) represents a multi-train system, safety function success SHALL be measured at the system level, not at the train level.	**NOTE SM-B5:** If one train of a system is judged to be seismically rugged (exclusive of a train-specific spatial interaction failure), then all trains of that system are considered rugged. (EPRI, 1991) states further that this assumption is valid if the train-wise layout is similar, although train-specific systems interaction problems may invalidate this assumption.
(REQ. SM-B6) Non-seismic failure modes and human actions identified on the success paths SHALL have low enough probabilities so as not to affect the seismic margin evaluation.	**NOTE SM-B6:** Non-seismic-caused component system unavailability is not explicitly addressed in a SMA. This should be reasonable for systems that have multiple and redundant trains but should be treated with caution for a single-train with recognized high unavailability. The screening criteria cited in the NRC's IPEEE guidance,NUREG/CR-5679 (Budnitz, Moore, and Julius, 1992), addressing both single-train and multi-train systems, MAY be used as guidance.

(REQ. SM-B7) The potential effects of seismically induced relay and contactor chatter SHALL be evaluated as well as the operator actions that may be required to recover from any such effects.	**NOTE SM-B7:** Guidance on evaluation of relay chatter effects is given in (EPRI, 1991), NUREG-1407 (NRC, 1991a) and (Hardy and Ravindra, 1991).
(REQ. SM-B8) Systems, structures and components needed to prevent early containment failure following core damage SHALL be examined as part of the SMA.	**NOTE SM-B8:** NUREG-1407 (NRC, 1991a) identifies these functions

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT C RESPONSES

(HLR-SM-C): Seismic responses calculated for the Review Level Earthquake SHALL be median centered, SHALL be based on current state-of-the-art methods of structural modeling and SHALL include the effects of soil-structure-interaction where applicable.

REQUIREMENT	COMMENTARY
(REQ. SM-C1) Seismic responses calculated for the Review Level Earthquake SHALL be median centered, SHALL be based on current state-of-the-art methods of structural modeling, and SHALL include the effects of soil-structure-interactions where applicable.	
(REQ. SM-C2) Depending on the site conditions and response analysis methods used in the plant design, realistic seismic responses MAY be obtained by a judicious combination of scaling, new analysis and new structural models.	
(REQ. SM-C3) For soil sites or when the design response analysis models are judged not to be realistic and state-of-the-art, or when the design input ground motion is significantly different from the site-specific input motion, new analysis SHALL be performed to obtain realistic structural loads and floor response spectra.	**NOTE SM-C3:** Further details about the basis for this requirement can be found in (ASCE, 1998).

(REQ. SM-C4) Soil structure interaction analysis SHALL be median centered using median properties at soil strain levels corresponding to the RLE input ground motion. At least three SSI analyses SHALL be conducted to investigate the effects on response due to uncertainty in soil properties. One analysis SHALL be at the median low strain soil shear modulus and additional analyses at the median value times $(1+C_v)$ and the median value divided by $(1+C_v)$, where C_v is a factor that accounts for uncertainties in the SSI analysis and soil properties. If adequate soil investigation data are available, the mean and standard deviation of the low strain shear modulus SHALL be established for every soil layer. The value of C_v SHALL then be established so that it will cover the mean plus or minus one standard deviation for every layer. The minimum value of C_v SHALL be 0.5. When insufficient data are available to address uncertainty in soil properties, C_v SHALL be taken as no less than 1.0.	

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT D SCREENING

(HLR-SM-D): The screening of components and subsequent seismic margin calculations SHALL incorporate the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support and potential spatial interactions.

REQUIREMENT	COMMENTARY
(REQ. SM-D1) If SSCs on the SSEL are screened out on the basis of their generic high seismic capacity exceeding the RLE the basis for such screening SHALL be confirmed through a walkdown.	
(REQ. SM-D2) A detailed walkdown of the plant SHALL be conducted focusing on equipment anchorage, lateral seismic support and potential systems spatial interactions. The purposes of such a walkdown are to find as-designed, as-built, and as-operated seismic weaknesses in the plant and to ensure that the seismic margins are realistic and plant-specific.	
(REQ. SM-D3) The walkdown SHALL be conducted consistent with the guidance given in (EPRI, 1991) and (Budnitz et al., 1985).	
(REQ. SM-D4) If components are screened out during or following the walkdown, anchorage evaluation justifying such a screening SHALL be provided.	**NOTE SM-D4:** Normally an anchorage calculation is required to support the screening. In some cases, the analyst MAY use judgment in deciding the adequacy of anchorage. Such judgments SHOULD be documented. For details and scope of anchorage evaluation, the reader is referred to (EPRI, 1991) and (Czarnecki, 1991).
(REQ. SM-D5) The walkdown SHALL focus on the potential for seismic induced fire and flooding following the guidance given in NUREG-1407 (NRC, 1991a).	**NOTE SM-D5:** Normally, if the walkdown team identifies a potential seismic-induced fire issue or a seismic-induced-flooding issue, it should be reviewed by the plant personnel, and is either dismissed on a defined basis or remedied if necessary. Only rarely is the SMA analysis team faced with the task of quantifying a seismic margin for seismic induced fire and/or flooding issues. However, if this is needed, the assessment must quantify the relevant HCLPF capacities and integrate these with the systems-analysis aspect as in any other aspect of SMA.

(REQ. SM-D6) The walkdown SHALL examine potential sources of spatial interaction (e.g., II/I issues, impact between cabinets, flooding and spray) and consequences of such interactions on SSCs contained in the SSEL, and SHALL incorporate them into the analysis as appropriate.	**NOTE SM-D6:** A "II/I issue" refers to the condition wherein a non-seismically qualified item could fall on and damage a seismically qualified equipment.

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT E
FAILURE MODES

(HLR-SM-E): Seismic margin calculations SHALL be performed for critical failure modes of structures, systems and components such as structural failure modes and functional failure modes identified through the review of plant design documents including analysis and test reports supplemented by earthquake experience data, fragility test data, generic qualification test data, and by a walkdown.

REQUIREMENT	COMMENTARY
(REQ. SM-E1) Realistic failure modes of screened-in structures, distribution systems and components that interfere with the operability of equipment during or after the earthquake SHALL be identified through review of plant design documents and the walkdown.	
(REQ. SM-E2) All relevant failure modes of structures (e.g., sliding, overturning, yielding, and excessive drift), equipment (e.g., anchorage failure, impact with adjacent equipment or structures, bracing failure, and functional failure) and soil (i.e., liquefaction, slope instability, excessive differential settlement) SHALL be considered, and HCLPF capacities for critical failure modes evaluated.	**NOTE SM-E2:** The concept of HCLPF capacity as an indicator of seismic margin was introduced in (Budnitz et al., 1985). Examples of calculations of HCLPF capacities for a selected set of SSCs can be found in (Kennedy et al., 1989). Detailed and more prescriptive guidance on methods for calculating HCLPF capacities of SSCs under different critical failure modes can be found in (EPRI,1991) and (Reed and Kennedy. 1994). Past seismic SMA reviews and seismic PRAs MAY also be used as guidance.

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT F CALCULATIONS

(HLR-SM-F): The calculation of seismic margins (or so-called HCLPF capacities) SHALL be based on plant-specific data supplemented by earthquake experience data, fragility test data and generic qualification test data. Use of such generic data SHALL be justified.

REQUIREMENT	COMMENTARY
(REQ SM-F1) Component seismic HCLPF capacities SHALL be based on plant specific data supplemented as needed by earthquake experience data, fragility test data and generic qualification test data.	
(REQ. SM-F2) All components and structures that are screened in SHALL have HCLPF capacities derived based on plant-specific information, such as site-specific seismic input, anchoring and installation of the component or structure, spatial interaction and plant-specific material test data.	
(REQ. SM-F3) Seismic HCLPF capacities SHALL be developed for SSCs that are identified in the systems model as playing a role in the LERF part of the Seismic PRA analysis (see REQ. SA-A1).	**NOTE SM-F3:** Generally the concern is the seismically induced early failure of containment functions. NUREG-1407 (NRC, 1991a) describes these functions as containment integrity, containment isolation, prevention of bypass functions, and some specific systems depending on the containment design (e.g., igniters, suppression pools, or ice baskets).

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT G SUCCESS PATH MARGINS

(HLR-SM-G): The plant seismic margin SHALL be reported based on the margins calculated for the success paths.

REQUIREMENT	COMMENTARY
(REQ. SM-G1) Plant seismic margin SHALL be reported based on the margins calculated for the SSCs on the success paths.	
(REQ. SM-G2) Plant seismic margin SHALL be reported for the plant after all SMA-related seismic upgrades have been done.	

SEISMIC MARGIN ASSESSMENT HIGH LEVEL REQUIREMENT H DOCUMENTATION

(HLR-SM-H): The Seismic Margin Assessment SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

REQUIREMENT	COMMENTARY
(REQ. SM-H1) The documentation SHALL meet the general documentation requirements in Section 7.	
(REQ. SM-H2) The documentation SHALL describe the methodologies used to quantify the seismic margins or HCLPF capacities of SSCs, together with key assumptions.	
(REQ. SM-H3) The documentation SHALL provide a detailed list of SSC margin values that includes the method of seismic qualification, the dominant failure modes(s), the source of information, and the location of each SSC. The parameter values defining the seismic margin (i.e., the HCLPF capacity and any other parameter values such as the median acceleration capacity and the beta values), and the technical bases for them, SHALL be provided for each analyzed SSC.	
(REQ. SM-H4) If SSCs on the SSEL are screened out on the basis of their generic high seismic capacity exceeding the RLE (see (REQ. SM-D1)), the basis for such screening SHALL be documented.	
(REQ. SM-H5) Different aspects of the SMA such as the selection of the RLE, the development of success paths and SSEL, the seismic response analysis, the screening, the walkdown, the review of design documents, the identification of critical failure modes for each SSC, and the calculation of HCLPF capacities for each screened-in SSC SHALL be documented.	**NOTE SM-H5:** The documentation requirements given in (NRC, 1991a) and (EPRI, 1991) MAY be used as guidance.

3.6 PRA for "Other" External Events – Requirements for Screening and Conservative Analysis

3.6.1 INTRODUCTION - SCOPE

The term "other external event" refers to external events other than earthquakes.

The term "screening out" is used here for the process whereby an external event is excluded from further consideration in the PRA analysis.

For the purposes of this section, which deals with screening out of one or more entire categories of external events, the term "external event" in the singular is used for a single and entire category of similar events; and the category is intended to include all "sizes" of such events within the category, so that if an "external event" is screened out, the implication is that the entire category is considered screened out.

For example, the external event "extremely cold weather" includes all extreme-cold conditions, no matter how extreme or how infrequent; the external event "nearby surface-transportation accidents" includes all such accidents arising from nearby surface transport modes; the external event "aircraft impact" includes crashes of all aircraft, of all sizes; and so on.

This set of requirements is concerned with screening-out. Even though as written it contemplates the screening-out of an entire "external event" category, it is not intended to restrict the analyst from screening out a sub-category, if the screening can be done on a defined basis and if the differentiation of the sub-category from the rest of the broad category is clear. For example, suppose that for a given site the only important risk potential from "aircraft impact" arises from military jet overflights. Suppose that large commercial jets can be screened out on the basis of a very low annual frequency, and that small cropduster planes can be screened out on the basis of not being able to cause enough damage. It is completely acceptable to sub-divide the external event "aircraft impact" into sub-categories, to screen the large jets and cropdusters on a defined basis, and then to subject only the military-jet subcategory to detailed PRA analysis using the requirements in Section 3.7.

3.6.2 UNDERLYING RATIONALE FOR SUCESSIVE SCREENING

There is a three-part underlying rationale for the requirements in this section:

(i) All potential external events (both natural hazards and man-made events) that may affect the facility must be considered, and each of them must be either screened out on a defined basis (following the requirements in this Section) or subjected to analysis using a

detailed PRA (following the requirements in Sections 3.7 to 3.9).

(ii) A set of screening criteria is provided, which provide a defensible basis for screening out an event.

(iii) If an external event cannot be screened out using these screening criteria, then a demonstrably conservative or bounding analysis, when used together with quantitative screening criteria, can also provide a defensible basis for screening out the event, without the need for detailed analysis. (Herein, the phrases "bounding analysis" and "demonstrably conservative analysis" are used interchangeably.)

The burden of demonstrating that a given bounding analysis is "demonstrably conservative" falls on the analyst; different circumstances will require different approaches. The general notion is that the conservatism is demonstrated in part by accounting for all uncertainties, approximations, or simplifications that might invalidate the demonstration if not accounted for appropriately.

There are three fundamental screening criteria embedded in the requirements here, as follows: An event can be screened out either (i) if it meets the criteria in the NRC's 1975 Standard Review Plan (NRC, 1975) or a later revision, or (ii) if it can be shown using a demonstrably conservative analysis that the mean value of the design-basis hazard used in the plant design is less than about 10^{-5}/year, and that the conditional core-damage probability is less than 10^{-1}, given the occurrence of the design-basis hazard; or (iii) if it can be shown using a demonstrably conservative analysis that the CDF (core damage frequency) is less than 10^{-6} per year.

Note that there is an implicit assumption that if an external event is screened out using one or another of the screening criteria herein, then neither the CDF nor the LERF arising due to that event is of concern. This assumption is made even though no explicit consideration is given in the screening to LERF issues.

An external event that cannot be screened out using any of these criteria must be subjected to the detailed-analysis requirements in Sections 3.7 to 3.9.

3.6.3 HIGH LEVEL REQUIREMENTS

<u>HLR-OTH-A</u> All potential external events (i.e., all natural hazards and man-made events) that may affect the site SHALL be considered, and SHALL be subjected to either screening, bounding analysis (demonstrably conservative analysis), or detailed analysis.

It should be understood that the remaining High Level Requirements below are applicable when an external event is selected for screening rather than for detailed analysis. At any time during the screening process, a decision can be made to bypass that process and

go directly to the detailed-analysis requirements in Sections 3.7 to 3.9. Appendix B contains a list of external events to be considered, and using this list is one acceptable approach to meeting this requirement. (See REQ. OTH-A1 below).

<u>HLR-OTH-B</u> Preliminary screening, if used, SHALL be performed using a defined set of screening criteria.

<u>HLR-OTH-C</u> A bounding (demonstrably conservative) analysis, if used for screening, SHALL be performed using defined quantitative screening criteria.

(If an external event cannot be screened out using either the qualitative criteria under HLR-OTH-B or the quantitative criteria under HLR-OTH-C, then it SHALL be subjected to detailed analysis under Sections 3.7 - 3.9).

<u>HLR-OTH-D:</u> The basis for the screening out of an external event SHALL be confirmed through a walkdown of the plant and its surroundings.

<u>HLR-OTH-E:</u> The screening out of an external event SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

3.6.4 SUPPORTING TECHNICAL REQUIREMENTS

<u>HIGH-LEVEL REQUIREMENT A – SCOPE</u>

(HLR-OTH-A): All potential external events (i.e., all natural hazards and man-made events) that may affect the site SHALL be considered, and SHALL be subjected to either screening, bounding analysis (demonstrably conservative analysis), or detailed analysis.

REQUIREMENT	COMMENTARY
(REQ. OTH-A1) The list of external events SHALL as a minimum include those that are enumerated in the PRA Procedures Guide, NUREG/CR-2300 (NRC, 1983) and NUREG-1407 (NRC, 1991a) and examined in past studies such as the NUREG-1150 analyses (Lambright et al, 1990). Appendix B contains the list adapted from NUREG/CR-2300, and the use of this list is one acceptable way to meet this requirement.	
(REQ. OTH-A2) The list considered in (REQ. OTH-A1) SHALL be supplemented with any site-specific and plant-unique external events.	**NOTE OTH-A2:** The purpose of this requirement is to assure that an unusual type of event is not inadvertently omitted simply because it does not fit neatly into any of the list of events commonly considered and listed in the standard references in REQ. OTH-A1. Examples are possible detritus or zebra mussels growth in the river affecting the intake (although they may be considered to have been included in the category "biological events"), or possible shoreline-slump effects (although they may be considered to have been included under "landslide" or "seiche.")
HIGH-LEVEL REQUIREMENT B – PRELIMINARY SCREENING **(HLR-OTH-B): Preliminary screening, if used, SHALL be performed using a defined set of screening criteria.**	
(REQ. OTH-B1) Initial Preliminary Screening: Meeting any one of the following five screening criteria SHALL be an acceptable basis for screening out an external event: <u>Criterion 1:</u> The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate the resistance of plant structures and systems to a particular external event. <u>Criterion 2:</u> The event has a significantly lower mean frequency of occurrence than another event, taking into account the uncertainties in the estimates of both frequencies, and the event could not result in worse consequences than the consequences from the other event. <u>Criterion 3:</u> The event cannot occur close enough to the plant to affect it.	**NOTE OTHER-B1:** These criteria are based on those found in the PRA Procedures Guide (NRC, 1983). The use of these criteria minimizes the likelihood of omitting any significant risk contributors while at the same time reducing the amount of detailed analysis required. In its guidance for the IPEEE procedures and submittals (NRC, 1991 and 1991a), the NRC staff applied these criteria for the population of operating nuclear power plants in the US, and concluded that only earthquakes, high winds, floods, transportation accidents, and nearby-facility accidents requiredevaluation in the IPEEE. However, the NRC staff required that each licensee confirm that no plant-unique external events with the potential to cause severe-accidents were being excluded from the IPEEE. In NUREG-1407 (NRC, 1991a), a progressive screening approach is recommended for evaluating high winds, floods, transportation accidents, and nearby facility accidents in the IPEEE. This IPEEE guidance required all licensees to review the information obtained on the plant design bases and any identified significant changes since the operating license for conformance with the 1975 Standard Review Plan criteria. It also required a confirmatory walkdown.

This criterion must be applied taking into account the range of magnitudes of the event for the recurrence frequencies of interest. <u>Criterion 4:</u> The event is included in the definition of another event. <u>Criterion 5:</u> The event is slow in developing and it can be demonstrated that there is sufficient time to eliminate the source of the threat or to provide an adequate response.	
(REQ. OTH-B2) Second Preliminary Screening: Meeting the following screening criterion SHALL be an acceptable basis for screening out an external event. The criterion is that the design basis for the event meets the criteria in the NRC's 1975 Standard Review Plan (NRC, 1975).	**NOTE OTH-B2:** If an external event meets the criteria in the NRC's Standard Review Plan (NRC, 1975), the contribution to core damage frequency is judged to be less than 10^{-6} per year based on various considerations. For certain external events, the SRP requires the selection of the design basis event at annual frequencies of occurrence between 10^{-7} and 10^{-6} (e.g., design basis explosions on transportation routes near the plant following Regulatory Guide 1.91 and turbine missile protection per Regulatory Guide 1.112). For some other events, conservative maximum sizes or intensities are specified (e.g., Design Basis Flooding per Regulatory Guide 1.59). In a study on wind risk, Ravindra and Nafday (1990) showed that the mean core damage frequency of plants meeting the 1975 SRP criteria is less than 10^{-6} per year. Based on a review of these and other supporting documents, the NRC staff recommended this screening criterion in NUREG-1407 (NRC, 1991a).
(REQ. OTH-B3) Application of the screening criteria for a given external event SHALL be based on a review of information on the plant's design hazard and the plant's NRC licensing basis relevant to that event.	**NOTE OTH-B3:** In the siting and plant-design stage, most site-specific natural and man-made external events will have been addressed and included in the design basis, unless they were screened out using the licensing criteria described in the NRC Standard Review Plan and Regulatory Guides.
(REQ. OTH-B4) Any significant changes since the NRC operating license was issued SHALL be reviewed. In particular, the review SHALL consider: (1) military and industrial facilities within 8 km of the site, (2) onsite storage or other activities involving hazardous materials, (3) nearby transportation, and (4) any other developments that could affect the original design conditions.	**NOTE OTH-B4:** This short list (1, 2, and 3) is specifically identified because it represents the most common areas where a significant change might have occurred since the issuance of the operating license.

HIGH-LEVEL REQUIREMENT C – DEMONSTRABLY CONSERVATIVE ANALYSIS

(HLR-OTH-C): A bounding (demonstrably conservative) analysis, if used for screening, SHALL be performed using defined quantitative screening criteria.

NOTE HLR-OTH-C: Herein, the phrases "bounding analysis" and "demonstrably conservative analysis" are used interchangeably.

(REQ. OTH-C1) If, by using a bounding (demonstrably conservative) analysis, any one of the following three screening criteria is met, this SHALL constitute an acceptable basis for screening out an external event. <u>Criterion A:</u> The current design-basis hazard cannot cause a core-damage accident. <u>Criterion B:</u> The current design basis hazard has a mean frequency less than 10^{-5} per year, and the mean value of the conditional core damage probability (CCDP) is assessed to be less than 10^{-1}. <u>Criterion C:</u> The core-damage frequency (CDF), calculated using a bounding (demonstrably conservative) analysis, has a mean frequency less than 10^{-6} per year.	**NOTE OTH-C1:** The bounding (demonstrably conservative) analysis is intended to provide a conservative calculation showing, if true, either that the hazard would not result in core damage or that the core damage frequency (CDF) is acceptably low. Some or all of the key elements of the external-event risk analysis could be used to reach and support this conclusion: hazard analysis, fragility analysis, or systems analysis (plant-systems analysis, human reliability analysis, accident-sequence analysis, etc.). In some cases, Criterion B can allow an efficient way to verify that the original design basis hazard is low and that the CDF is also acceptably low. Using Criterion B requires a refined modeling of the hazard and an approximate evaluation of CCDP. The analysis under Criterion B is a subset of the more extensive demonstrably conservative analysis of CDF under Criterion C.
(REQ. OTH-C2) Estimation of the mean frequency and the other parameters of the design basis hazard SHALL be based on state-of-the-art hazard modeling and recent data (e.g., annual maximum wind speeds at the site, aircraft activity in the vicinity, or precipitation data), or SHALL be bounding (demonstrably conservative) for the purposes of the demonstrably conservative analysis. The uncertainties in modeling and data SHALL be considered in the hazard evaluation.	**NOTE OTH-C2:** The spirit of a bounding (demonstrably conservative) or conservative analysis is such that it is acceptable to use demonstrably conservative modeling and data for the hazard evaluation here.
(REQ. OTH-C3) Estimation of the mean CCDP SHALL utilize a systems model of the plant that meets the Category II systems-analysis requirements in the ASME internal-events PRA Standard insofar as they apply (ASME, 2000). For the purposes of the demonstrably conservative analysis, a demonstrably conservative approach to the systems model SHALL be acceptable.	
(REQ. OTH-C4) The conservative estimation of the mean core damage frequency (CDF) developed here SHALL be based on models and data that are either realistic or	**NOTE OTH-C4:** Calculation of this CDF may be done using different demonstrably conservative assumptions, as explained by the following example. Example: Typically, nuclear power plants are sited such that the accidental impact of plant structures by aircraft is highly unlikely. As part of the external event PRA, the risk from aircraft accidents may be assessed at different levels. The mean annual frequency of aircraft

demonstrably conservative.	impact during takeoff, landing, or in flight may be determined. If this hazard frequency is very low (e.g., 10^{-7} per year), then the aircraft impact as an external event may be eliminated from further study. This approach assumes that the aircraft impact results in damage of the structure leading to core damage or large early release (this assumption is likely to be highly conservative). If the frequency of aircraft impacting the plant structures is estimated to be larger, the fragility of the structures may be evaluated to make a refined estimate of the frequency of core damage. Further refinements could include (1) elimination of certain structural failures as not resulting in core damage (e.g., damage of diesel generator building may not result in core damage if offsite electrical power is available); and (2) performing a plant systems and accident sequence analysis to calculate the core damage frequency. This example shows that for some external events, it may be sufficient to perform only the hazard analysis; for some others, the hazard analysis and a simple fragility analysis may be needed; in rare cases, a plant-systems and accident sequence analysis may be necessary. Other examples of bounding (demonstrably conservative) analysis can be found in (Ravindra and Bannon, 1985, 1985a), (Kimura and Budnitz, 1987), and (Lambright et al., 1990).

(REQ. OTH-C4) If none of the screening criteria in this entire Section 3.6 can be met for a given external event, then additional analysis SHALL be undertaken. (See Sections 3.7 to 3.9 of this Standard.)

HIGH LEVEL REQUIREMENT D -- WALKDOWN

(HLR-OTH-D): The basis for the screening out of an external event SHALL be confirmed through a walkdown of the plant and its surroundings.

(REQ. OTH-D1) The basis for the screening out of an external event SHALL be confirmed through a walkdown of the plant and its surroundings.	**NOTE OTH-D1:** The general external-events-screening walkdown SHOULD concentrate, although not exclusively, on outdoor facilities that could be affected by high winds and flooding, onsite storage of hazardous materials, and offsite developments such as increased usage of or new airports/airways, highways and gas pipelines.

(REQ. OTH-D2) If the screening-out of any specific external event depends on the specific plant layout, then the walkdown SHALL confirm that layout. For most external events, this typically means a walkdown that evaluates the site layout outside the plant buildings as well as inside.

HIGH-LEVEL REQUIREMENT E -- DOCUMENTATION

(HLR-OTH-E): The screening out of an external event SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

(REQ. OTH-E1) The documentation SHALL meet the general documentation requirements in Section 7.

(REQ. OTH-E2) For each external event that is screened out, the approach used for the screening

(preliminary screening or demonstrably conservative analysis) and the screening criteria used SHALL be documented.
(REQ. OTH-E3) The documentation SHALL include any engineering or other analysis performed to support the screening-out of an external event.

3.7 PRA for "Other" External Events – Technical Requirements for Analysis

3.7.1 SCOPE, APPLICABILITY, TERMINOLOGY, PEER REVIEW

Scope: The term "other external event" refers to external events other than earthquakes.

Applicability: This section applies to "other" external events that cannot be screened out (that is, cannot be excluded from further consideration in the PRA analysis) using the processes and criteria in Section 3.6, "PRA for 'Other' External Events – Requirements for Screening and Conservative Analysis." The Requirements in this section can be used for the analysis of any "other" external event. Alternatively, the Requirements in Section 3.8 ("High Winds PRA") or Section 3.9 ("External Flooding PRA") can be used for those external events. If either Section 3.8 or 3.9 is used, then all of the Requirements therein apply.

Aircraft impact PRA: For the PRA of aircraft impact, the Requirements herein apply. However, another acceptable method for meeting this Standard is to follow the methodology in the DOE Standard (DOE, 1996), which is a methodology standard for aircraft-impact PRA developed by the U.S. Department of Energy for analyzing impacts on various DOE facilities. This DOE methodology may be used as an alternative way to satisfy in full the intent of High Level Requirements HLR-ANA-A ("Hazard Analysis") and HLR-ANA-B ("Fragility Evaluation") herein, and of their Supporting Requirements. It would still be necessary to meet the Requirements under HLR-ANA-C ("Systems Analysis and Quantification") and HLR-ANA-D ("Documentation") herein.

Terminology – "external event" in the singular: For the purposes of this section, which deals with analysis of an entire category of external event, the term "external event" in the singular is used for a single and entire category of similar events; and the category is intended to include all "sizes" of such events within the category. For example, the external event "extremely cold weather" includes all extreme-cold conditions, no matter how extreme or how infrequent; the external event "nearby surface-transportation accidents" includes all such accidents arising from nearby surface transport modes; the external event "aircraft impact" includes crashes of all aircraft, of all sizes; and so on.

This set of requirements is concerned with detailed PRA analysis of an external event category. Even though as written it contemplates the analysis of an entire "external event" category, it is not intended to restrict the analyst from analyzing a sub-category, if the differentiation of the sub-category from the only important of the broad category is clear. For example, suppose that for a given site the real risk potential from "aircraft impact" arises from military jet overflights. Suppose that large commercial jets can be screened out using Section 3.6 on the basis of a very low annual frequency, and that small cropduster planes can be screened out using Section 3.6 on the basis of not being able to cause enough damage. It is completely acceptable to sub-divide the external

event "aircraft impact" into sub-categories, to screen the large jets and cropdusters using judgment and approximate analysis, and then to subject only the military-jet subcategory to detailed PRA analysis using the requirements here.

Importance of Peer Review: It should be noted that detailed analysis of external events other than earthquakes (and occasionally high winds and external flooding) is not common for US nuclear power plants, because screening analyses and demonstrably conservative analyses, using the approaches in Section 3.6, have usually shown that the contributions to CDF are insignificant. Therefore, the collective experience of the analysis community is limited. Because of this limited experience, the analyst team may need to improvise its approach for any external event requiring detailed analysis following the overall methodology requirements in this Section. Given the above, an extensive peer review is very important if an analysis under this Section is undertaken.

LERF (Large Early Release Frequency): In applying the analyses covered in this Section 3.7, it is necessary to be attentive to both core-damage frequency and LERF. In this regard, the discussion about LERF in Section 1.8.3 is applicable, and should be taken into account. Also, the analyst is urged to be especially attentive to effects of the external event that might compromise containment integrity and thereby possibly contribute to LERF-type accident sequences.

3.7.2 UNDERLYING RATIONALE FOR THE ANALYSIS REQUIREMENTS

Screening, Realistic Analysis, and Conservative Analysis: Presumably, if an external event cannot be screened out based on the criteria in Section 3.6, it is because it fails to meet those criteria -- or at least, the external event cannot be shown to meet those criteria using the screening-out methods or demonstrably-conservative analysis methods of Section 3.6. The fundamental screening-out criteria in Section 3.6 are as follows (quoting from Section 3.6.2): "An event can be screened out either (i) if it meets the criteria in the NRC's 1975 Standard Review Plan (NRC, 1975), or (ii) if it can be shown using a demonstrably conservative analysis that the mean value of the design-basis hazard used in the plant design is less than about 10^{-5}/year, and that the conditional core-damage probability is less than 10^{-1}, given the occurrence of the design-basis hazard; or (iii) if it can be shown using a demonstrably conservative analysis that the CDF (core damage frequency) is less than 10^{-6} per year."

It is recognized that for some external events, although it may be difficult or impossible to demonstrate that any of these criteria is met using screening or demonstrably conservative analysis, nevertheless the risk posed by the entire event category is quite small, as measured by the event's contribution to CDF and LERF. Given this possibility, although the detailed analysis contemplated in this Section is intended to be a realistic analysis, it is quite acceptable to introduce conservatisms in any given step, provided that at the end the overall contributions to CDF and LERF are demonstrably small. If, however, either of these contributions turns out to be "important" – presumably, important

compared to other CDF and/or LERF contributions from other initiators – then the PRA analyst team is obliged to re-visit the analysis here to make it as realistic as feasible.

Rationale and Structure of the Requirements Here: There is a three-part structure to the PRA of any external event, and hence to the requirements here: (i) hazard analysis; (ii) fragility evaluation; and (iii) systems analysis and quantification.

General Guidance: The PRA Procedures Guide (NRC, 1983) and the PSA Procedures Guide (Brookhaven, 1985) both contain detailed discussions that provide general guidance on how to approach the PRA analysis of an external event. Some of the "Commentary" herein is adapted from these guides.

3.7.3 HIGH LEVEL REQUIREMENTS

HLR-ANA-A – HAZARD ANALYSIS: The analysis of the hazard (the frequency of occurrence of different intensities of the external event) SHALL be based on a site-specific probabilistic evaluation reflecting recent available data and site-specific information. The analysis can be based on either historical data or a phenomenological model, or a mixture of the two. The models used for frequency and intensity calculations SHOULD NOT be unduly influenced by recent, short-term trends in the frequencies.

HLR-ANA-B – FRAGILITY EVALUATION: The fragility of an SSC (the conditional probability of its failure as a function of the intensity of the external event hazard) SHALL be evaluated using plant-specific, SSC-specific information and an accepted engineering method for evaluating the postulated failure.

HLR-ANA-C – SYSTEMS ANALYSIS AND QUANTIFICATION: The systems model SHALL include all important initiating events caused by the effects of the external event that can lead to core damage or large early release. The model SHALL be adapted from the internal-events, full-power PRA systems model to incorporate those aspects that are different, due to the external event's effects, from the corresponding aspects of the full-power, internal-events model.

HLR-ANA-D – DOCUMENTATION: The detailed PRA analysis of the external event SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

3.7.4 SUPPORTING TECHNICAL REQUIREMENTS

HIGH-LEVEL REQUIREMENT A -- HAZARD ANALYSIS

(HLR-ANA-A): The analysis of the hazard (the frequency of occurrence of different intensities of the external event) SHALL be based on a site-specific probabilistic evaluation reflecting recent available data and site-specific information. The analysis can be based on either historical data or a phenomenological model, or a mixture of the two. The models used for frequency and intensity calculations SHOULD NOT be unduly influenced by recent, short-term trends in the frequencies.

REQUIREMENT	COMMENTARY
REQ. ANA-A1) The hazard analysis SHALL be site-specific and plant-specific to the extent necessary for the purposes of the analysis.	**NOTE ANA-A1:** Although a site-specific and plant-specific hazard analysis is always desirable, it is often acceptable to develop a hazard on some other basis (for example, a regional or even generic basis), provided that the uncertainties introduced are acceptable for the applications contemplated.
(REQ.-ANA-A2) The hazard analysis for the external event SHALL use an accepted methodology and up-to-date databases. Uncertainties in the models and parameter values SHALL be properly accounted for and fully propagated in order to obtain a family of hazard curves from which a mean hazard curve can be derived.	**NOTE ANA-A2:** In general, the hazard posed by any external event can only be described by a multitude of variables related to the "size" of the event. Often, some of these variables are probabilistically dependent on other variables. However, for simplicity the hazard function is generally described, albeit imperfectly, in terms of a limited number of variables -- typically, one. For example, although a proper characterization of the hazard from a potential chemical explosion from a nearly railroad train carrying chemicals should include blast distance, duration, instantaneous pressure duration, shape of the pressure pulse as a function of frequency, chemical form of the explosive, and so on, the hazard would likely be characterized by only one or two of these parameters in any actual analysis. The other variables that would be needed for a "complete" description of the hazard would typically be considered in the response analysis and fragility evaluation, or may represent an irreducible variability in the hazard, or some of each. The output of the hazard analysis is a so-called "hazard curve" -- actually, a family of hazard curves accounting for uncertainties -- of exceedence frequency vs. hazard intensity. The PRA Procedures Guide (NRC, 1983) has a useful discussion of the general considerations involved in hazard analysis.
(REQ.-ANA-A3) If expert elicitation or another use-of-experts process is used in developing the hazard, it SHALL be done in accordance with established guidelines.	**NOTE ANA-A3:** The discussion in Section 3.4.1.1 (in the section that introduces the hazard requirements for seismic PRA), and the corresponding Supporting Technical Requirements and Commentary in Section 3.4.1.3 at REQ. HA-A4, HA-C2, and HA-D2, contain useful guidance on this subject. Also, the ASME PRA Standard (ASME, 2000) contains requirements on this subject. Adapting these to the situation of the "other" external event analyzed here is acceptable.

HIGH-LEVEL REQUIREMENT B -- FRAGILITY EVALUATION

(HLR-ANA-B): The fragility or vulnerability of an SSC (the conditional probability of its failure as a function of the intensity of the external event hazard) SHALL be evaluated using plant-specific, SSC-specific information and an accepted engineering method for evaluating the postulated failure.

(REQ. ANA-B1) The fragility estimates SHALL be site-specific and plant-specific to the extent necessary for the purposes of the analysis.	**NOTE ANA-B1:** Although a site-specific and plant-specific analysis of the fragilities of SSCs is always desirable, it is often acceptable to develop fragility estimates on some other basis (for example, based on generic information), provided that the uncertainties introduced are acceptable for the applications contemplated.
(REQ. ANA-B2) Fragilities of SSCs SHALL be evaluated using an accepted methodology and plant-specific data. The findings of a plant walkdown SHALL be considered in this evaluation.	**NOTE ANA-B2:** The fragility or vulnerability of an SSC is estimated from the actual capacity of the SSC for a <u>given failure mode</u>. Thus a failure-mode identification is a crucial aspect of this work. Another crucial aspect is an engineering evaluation of how the effect of the external event is transmitted to the SSC -- what force or effect leads to the specified failure mode. To make the PRA analysis tractable, the fragility should be expressed as a function of the same variable – related to the "size" of the external event -- that the hazard curves are functions of. This allows the convolution of the hazard curves and fragility curves during the quantification step to be done in a mathematically straightforward way. The PRA Procedures Guide (NRC, 1983) has a useful discussion of the general considerations involved in fragility evaluation.
(REQ. ANA-B3) The fragility analysis SHALL appropriately reflect the uncertainties in the underlying information and the models used.	**NOTE ANA-B3:** The analysis of the fragility or vulnerability of an SSC must account for the various uncertainties in both underlying data and models. The Requirements and Commentary on this subject given in Section 3.4.2 (on seismic PRA fragility analysis) contain useful guidance on this subject. Adapting these to the situation of the "other" external event analyzed here is acceptable.

HIGH-LEVEL REQUIREMENT C -- SYSTEMS ANALYSIS AND QUANTIFICATION

(HLR-ANA-C): The systems model SHALL include all important initiating events caused by the effects of the external event that can lead to core damage or large early release. The model SHALL be adapted from the internal-events, full-power PRA systems model to incorporate those aspects that are different, due to the external event's effects, from the corresponding aspects of the full-power, internal-events model.

NOTE: In special circumstances, it is acceptable to develop an ad-hoc systems model tailored especially to the external event being analyzed, instead of starting with the internal-events systems model and adapting it. If this approach is used, it is especially important that a peer review be undertaken that concentrates on this aspect.

(REQ. ANA-C1) Accident sequences initiated by the external event SHALL be assessed to estimate CDF and LERF contribution. The analysis	**NOTE ANA-C1:** The PRA systems-analysis model for any external event is almost always based on the internal-events full-power PRA systems model, to which are added basic failure events derived from the information developed in the specific external event's fragility analysis. Considerable screening out and trimming of the internal-events systems

SHALL consider the appropriate hazard curves and the fragilities of structures and equipment.	model is also common, where appropriate. The analysis consists of developing event trees and fault trees in which the initiating event can be either the external event itself or a transient or LOCA induced by the event. Various accident sequences that lead to core damage or large early release are identified, and their conditional probabilities of occurrence calculated. The frequency of core damage or large early release is obtained by a convolution of these conditional probabilities over the relevant range of hazard intensities. The procedure for determining the accident sequences is similar to that used in seismic-PRA systems analysis, and following the Requirements therein represents one acceptable approach, after they are adapted to apply to the PRA situation represented by the specific external event. (See the Requirements and Commentary in Section 3.4.2, and the discussion about seismic PRA methods in Appendix A). Other factors to be considered include non-external-event-related unavailabilities or failures of equipment; operator errors; unique aspects of common causes, correlations, and dependencies; any warning time available to take mitigating steps; the possibility of recovery actions by operators and replacement by substitutes to accomplish the needed function; and the likelihood of common-caused failures.
(REQ. ANA-C2) The integration-quantification SHALL account for all dependencies and correlations, and SHALL account for the uncertainties in each of the inputs.	**NOTE ANA-C2:** The usefulness of the "final results" of the PRA for the external hazard are dependent on performing enough assessment to understand the dependencies, correlations, and uncertainties, and to account for them quantitatively if they are important.

HIGH-LEVEL REQUIREMENT D -- DOCUMENTATION

(HLR-ANA-D): The detailed PRA analysis of the external event SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

(REQ. ANA-D1) The documentation SHALL meet the general documentation requirements in Section 7.

(REQ. ANA-D2) The documentation SHALL include a description of the specific methods used for determining the hazard curves, including the technical interpretations that are the basis for the inputs and results.

(REQ. ANA-D3) The documentation SHALL describe the specific adaptations made to the internal-events PRA model to produce the specialized external event PRA model, and their motivation.

(REQ. ANA-D4) The documentation SHALL describe the methodologies used to quantify the fragilities of SSCs, together with key assumptions.

(REQ. ANA-D5) The documentation SHALL provide a detailed list of SSC fragility values that includes the method of analysis, the dominant failure modes(s), the sources of information, and the location of each SSC.

(REQ. ANA-D6) The documentation SHALL discuss the basis for the screening-out of any generic high-capacity SSCs.

3.8 High-Winds PRA – Technical Requirements

3.8.1 INTRODUCTION

It should be noted that detailed PRA analysis of high winds has been carried out for very few US nuclear power plants, because the hazard and plant analysis carried out during the design stage provide a basis for the screening analyses and demonstrably conservative analyses using the approaches in Section 3.6. These approaches have usually shown that the contribution to CDF is insignificant. Therefore, the collective experience with PRA analysis is limited. Because of this limited experience, the analyst team may need to improvise its approach to high-winds PRA analysis following the overall methodology requirements in this Section. A peer review is very important if an analysis under this Section is undertaken.

The technical requirements for high-winds PRA are similar, with adaptations, to those for seismic PRA. The major elements are wind hazard analysis, wind fragility analysis, and systems analysis including quantification. The analyst familiar with seismic PRA but unfamiliar with high winds PRA methods should refer both to the seismic-PRA Requirements and Commentary (Section 3.4) and to Appendix A ("Seismic PRA Methodology").

There are several types of high-wind events that need to be considered, depending on the site. These include 1) tornado winds and other tornado effects, 2) tropical cyclone winds (cyclones, hurricane and typhoons), and 3) extra tropical straight winds (thunderstorms, squall lines, weather fronts, etc.). It is assumed here that the analyst team has employed screening methods (see Section 3.6) to eliminate from consideration those high wind events that are not important at the site under study, so that the requirements in this section will be used to analyze only those high-wind phenomena that have not been screened out.

It is further assumed here that the high-winds-PRA analysis team possesses an internal-events full-power Level 1 and Level-2-LERF PRA, developed either prior to or concurrently with the high-winds PRA; that this internal-events PRA is used as the basis for the high-winds-PRA systems model; and that the technical basis for the internal-events full-power PRA is the ASME PRA standard (ASME, 2000).

References that are useful in developing a high-winds PRA include (NRC, 1983), (Brookhaven, 1985), (Consolidated Edison Company et al., 1983), (Ravindra et al., 1987), (Cramond, Ericson, and Sanders, 1987), and (Reed and Ferrell, 1987). The relevant references for wind-hazard analysis are provided in the Commentary below adjacent to the relevant wind-hazard Technical Requirements.

3.8.2 HIGH-WINDS-PRA TECHNICAL REQUIREMENTS

The high-winds-PRA technical requirements consist of four High-Level Requirements, under which are organized the several Supporting Technical Requirements, as follows:

HIGH-LEVEL REQUIREMENT A: HAZARD

(HLR-WIND-A): The frequency of high winds at the site SHALL be based on a site-specific probabilistic wind hazard analysis (existing or new) that reflects recent available regional and site-specific information.

NOTE: The models used for frequency and intensity calculations SHOULD NOT be unduly influenced by recent, short-term trends in the frequencies of high-wind events. They SHOULD incorporate at least the worst weather conditions experienced historically at the site.

REQUIREMENT	COMMENTARY
(REQ.-WIND-A1) Tornado wind hazard analysis SHALL use an accepted methodology and up-to-date databases on tornado occurrences, intensities, etc. Uncertainties in the models and parameter values SHALL be properly accounted for and fully propagated in order to obtain a family of hazard curves from which a mean hazard curve can be derived.	**NOTE WIND-A1:** Acceptable methodologies are given by (Twisdale, Dunn, and Alexander, 1981) and (Reinhold and Ellingwood, 1982). Examples of tornado hazard analysis for nuclear facilities using these methodologies can be found in (Ravindra and Bannon, 1985), (Twisdale and Hardy, 1985), and (Ramsdell and Andrews, 1986). Tornado wind hazard analysis SHOULD include the following elements: Variation of tornado intensity with occurrence frequency. (The frequency of tornado occurrence decreases rapidly with increased Intensity.) Correlation of tornado width and length of damage area; longer tornadoes are usually wider. Correlation of tornado area and intensity; stronger tornadoes are usually larger than weaker tornadoes. Variation in tornado intensity along the damage path length; tornado intensity varies throughout its life cycle. Variation of tornado intensity across the tornado path width. Variation of tornado differential pressure across the tornado path width.
(REQ. WIND-A2) Risks from hurricanes SHALL be evaluated using an accepted hurricane hazard analysis methodology and up-to-date databases on hurricane occurrences, intensities, etc. Uncertainties in the models and parameter values SHALL be properly accounted for and fully propagated in order to obtain a family of hazard curves from which a mean hazard curve can be derived.	**NOTE WIND-A2:** In the U.S., hurricanes predominantly affect the Gulf of Mexico and the Atlantic coastline. Hurricanes rapidly decay during their movement over land due to friction from terrain. Hence, it is sufficient to consider their impact only up to a few hundred kilometers or so from the coastline and a hurricane risk analysis is not required further inland. However, wind hazard frequencies for a site can be generated from direct wind measurements at the site (Liu, 1991). Due to the absence of direct wind measurements at many sites of interest for significant time periods, numerical simulation techniques are commonly used to generate hurricane wind hazard frequencies for a site. A stochastic model of hurricane occurrences is used and the hazard analysis considers the occurrence rate of hurricanes for each coastal segment, distribution of central pressure, radius of maximum winds, storm decay over land, wind field

	characteristics, and coast crossing location. Available probabilistic models are discussed in (Twisdale and Vickery, 1995). Numerical simulations based on these models simulate the hurricane wind field using random variables that model the size, intensity, translation speed, direction and the location of the site with respect to the coastal line. The probability density functions of these variables are developed using hurricane data compiled by (Batts et al., 1980) and (Jarvinen et al., 1984). Such a simulation procedure was used in developing the hurricane wind hazard curves for the Indian Point site (Twisdale, et al., 1981).
(REQ. WIND-A3) The hazard from extra-tropical windstorms and other high straight wind phenomena SHALL be evaluated using recorded wind-speed data appropriate to the site.	**NOTE WIND-A3:** For inland sites in the U.S. the hazard (i.e., annual probability of exceedance) at lower wind speeds is typically higher from extra-tropical straight windstorms than from tornadoes or hurricanes. Therefore, the evaluation of risks from extra-tropical straight windstorms is needed, especially if the plant structures have not been designed to withstand tornadoes. Typically, the annual maximum wind speed data recorded at a weather station appropriate to the site are fitted by a Type I extreme value probability distribution. Since the site-specific wind speed data may be available over only a short period (e.g., less than 50 years), there is considerable uncertainty in the hazard, especially at higher wind speeds (Lui, 1981). It is customary to assume that the uncertainty in the hazard comes mainly from the sampling error due to the small number and duration of records (See (Simiu and Scanlan, 1986)). This standard deviation is taken into account to obtain a family of hazard curves with assigned subjective probabilities (e.g., (Reed and Ferrell, 1987). Other uncertainties that arise from lack of weather station data near the site, terrain differences, and so on SHOULD be accounted for properly in developing the wind hazard curves.
(REQ.WIND-A4) Risks from wind-generated missiles SHALL be evaluated using an accepted high-wind missile hazard analysis methodology. Specific features of exterior barriers (i.e., walls and roof) of safety-related structures, any weather exposed structures, systems and components and the consequences of this damage from wind-borne missile impact which may result in core damage or large early release SHALL be considered in this evaluation.	**NOTE WIND-A4:** An acceptable method for evaluating wind-borne missile risk is given in (Twisdale, 1988) and (Twisdale and Vickery, 1995). It models the tornado wind field, trajectory of missiles (injection and transportation) and impact effects of missiles onto safety-related buildings and exposed equipment. A survey of the plant buildings and its surroundings SHOULD be made to assess the number and types of objects that could be picked up by a tornado and could become potential missiles. Using the results of the detailed tornado missile risk analysis, (Reed and Ferrell, 1987) have developed missile strike probabilities per unit area of buildings. These MAY be used in a demonstrably conservative analysis. Note that tornado missile risk is judged to be acceptably small if the plant design meets the 1975 NRC Standard Review Plan Criteria (NRC, 1975). Note also that wind-generated missiles from other high-wind phenomena (hurricanes, etc.) can be analyzed using the tornado-missile method discussed here.

HIGH-LEVEL REQUIREMENT B: FRAGILITIES

(HLR-WIND-B): A wind fragility evaluation SHALL be performed to estimate plant-specific, realistic wind fragilities for those structures, systems, and components whose failure may contribute to core damage and/or large early release.

(REQ. WIND-B1) Wind fragilities of structures and components (e.g.,	**NOTE WIND-B1:** Wind fragility is evaluated using the same general methodology as for seismic fragilities (See the Requirements in Section

tanks, transformers, diesel-generator exhaust stack, piping and intake pumps) SHALL be evaluated using an accepted methodology and plant-specific data. The assessment SHALL include non-safety structures that could fall into/onto safety-related structures, thereby causing damage. The findings of a plant walkdown SHALL be considered in this evaluation.	3.4.3.1, "Seismic Fragility Evaluation," and the seismic-fragility discussion in Appendix A). Typically, the entire family of fragility curves for an SSC corresponding to a particular failure mode is expressed in terms of the median wind speed capacity, V_m, and the logarithmic standard deviations, β_R and β_U, representing randomness in capacity and uncertainty in median capacity, respectively. Such fragility parameters are estimated for the credible failure modes of the SSC. Failure of structures could be overall, such as failure of a shearwall or moment resisting frame, or local, such as out-of-plane wall failure or pull-off of metal siding. Wind pressure loading is based on the methodology contained in wind design standards (ASCE, 1998a). The effect of wind borne missiles on structures, systems and components can be found in (ASCE, 1980) and (Stevenson and Zhao, 1996). The development of fragility curves for structures is done in terms of the factor-of-safety, defined as the resistance capacity divided by the response associated with the design basis loads from extreme winds. The variability of the factor-of-safety depends on the variability of strength capacity and the response to specified loads. Wind capacity is modeled as a product of random variables and is expressed in terms of wind speed. Besides the strength characteristics, the capacity of a structure for the effects of wind pressure also depends on a number of factors affecting wind pressure/force relationship. For example, shielding effects of various structures at the site results in an increase of wind speed through a constricted space or a decrease where it may be slowed down due to obstructions. Such funneling characteristics describing the channeling of winds around structures have a very important influence on the wind forces. The actual forces are also determined by the structural shapes, because wind pressure and forces are related to the wind velocity by a shape factor. Another factor important in this regard is the vertical distribution of wind velocity, which is a function of terrain roughness. Examples of the development of wind fragilities for structures can be found in (Consolidated Edison Company et al., 1983), (Ravindra et al., 1997) and (Reed and Ferrell, 1987). Most nuclear power plant structures have excellent wind resistance. Major vulnerabilities have sometimes been identified for non-seismic-Category I structures due to their potential for collapsing on safety-related structures or equipment. This includes exhaust stacks, unprotected walls, outside wiring and cabling, etc. Similarly, many of the older plants have safety-related equipment such as tanks and equipment located outdoors that are vulnerable to wind-borne missiles. They SHOULD be identified during the walkdown. In analyzing the failure of indoor equipment (within the structures), it is conservatively assumed that the failure of a structure causes the failure of all equipment dependent on or within the structure. It is possible that the structure may not collapse but the indoor equipment may still be damaged from pressure drop due to passage of a tornado. This occurs because of inadequate venting in the structure. There is a rapid pressure drop due to passage of a tornado and this results in escape of air from the building; if the exit is not rapid enough, it causes internal pressure. This might lead to failure of block walls, which could collapse onto safety-related structures. Indoor equipment is also susceptible to damage from missiles entering through louvres, vents, etc. Damage to internal structures, systems or components may also be caused by wind-induced pressurization through openings in the structure.

HIGH-LEVEL REQUIREMENT C: SYSTEMS ANALYSIS AND QUANTIFICATION

(HLR-WIND-C): The wind-PRA systems model SHALL include all important wind-caused initiating events that can lead to core damage or large early release. The model SHALL be adapted from the internal-events, full-power PRA systems model to incorporate wind-analysis aspects that are different from the corresponding aspects in the full-power, internal –events PRA systems model.

NOTE: In special circumstances, it is acceptable to develop an ad-hoc systems model tailored especially to the high-wind phenomenon being analyzed, instead of starting with the internal-events systems model and adapting it. If this approach is used, it is especially important that a peer review be undertaken that concentrates on this aspect.

(REQ. WIND-C1) Accident sequences initiated by high winds SHALL be assessed to estimate CDF and LERF contribution. The analysis SHALL consider the appropriate wind hazard curves and the fragilities of structures and equipment.
The systems-analysis approach for wind-initiated accident sequences SHALL use the same general approach used for developing seismic-initiated sequences in seismic PRA (see the corresponding Requirements in Section 3.4.2.2). One acceptable approach is to follow those seismic-PRA Requirements, after adapting them to the wind-PRA situation.

NOTE WIND-C1: The wind-PRA systems-analysis model is almost always based on the internal-events full-power PRA systems model, to which are added basic failure events derived from the information developed in the wind fragility analysis. Considerable screening out and trimming of the internal-events systems model is also common, where appropriate. The analysis consists of developing event trees and fault trees in which the initiating event can be either the extreme wind effect itself or a transient or LOCA induced by the extreme winds. Various accident sequences that lead to core damage or large early release are identified and their conditional probabilities of occurrence calculated. The frequency of core damage or large early release is obtained by a convolution over the relevant range of hazard intensities.

The procedure for determining the accident sequences is similar to that used in seismic-PRA systems analysis, and following the Requirements therein represents one acceptable approach, after they are adapted to apply to the wind-PRA situation. (See the Requirements and Commentary in Section 3.4.2, and the discussion about seismic PRA methods in Appendix A). Other factors to be considered include non-wind-related unavailabilities or failures of equipment, operator errors, any warning time available to take mitigating steps (e.g., in the case of hurricanes), the possibility of recovery actions by operators and replacement by substitutes to accomplish the needed function; and the likelihood of common-caused failures.

Examples of systems analysis for high winds can be found in the Indian Point IPEEE report (Consolidated Edison Company et al., 1996) and the several so-called "TAP A-45" reports that Sandia performed for the NRC (Cramond, Ericson, and Sanders, 1987).

(REQ. WIND-C2) The integration-quantification SHALL account for the uncertainties in each of the inputs, and SHALL account for all identified dependencies and correlations.

NOTE WIND-C2: The usefulness of the "final results" of the PRA for high winds are dependent on performing enough assessment to understand the uncertainties, dependencies, and correlations, and to account for them quantitatively if they are important.

HIGH-LEVEL REQUIREMENT D: DOCUMENTATION

(HLR-WIND-D): The high-winds-PRA analysis SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

(REQ. WIND-D1) The documentation SHALL meet the general documentation requirements in Section 7.
(REQ. WIND-D2) The documentation SHALL include a description of the specific methods used for determining the high-wind hazard curves including associated wind pressure, pressure distributions, missile and differential pressure effects and the scientific interpretations that are the basis for the inputs and results.
(REQ. WIND-D3) The documentation SHALL describe the specific adaptations made to the internal-events PRA model to produce the high-wind-PRA model, and their motivation.
(REQ. WIND-D4) The documentation SHALL describe the methodologies used to quantify the high-wind fragilities of SSCs, together with key assumptions.
(REQ. WIND-D5) The documentation SHALL provide a detailed list of SSC fragility values that includes the method of analysis, the dominant failure modes(s), the sources of information, and the location of each SSC.
(REQ. WIND-D6) The documentation SHALL discuss the basis for the screening-out of any generic high-capacity SSCs.

3.9 External Flooding PRA – Technical Requirements

3.9.1 INTRODUCTION

Detailed PRA analysis of external flooding has been carried out for very few US nuclear power plants, because the hazard and plant analysis carried out during the design stage provide a basis for the screening analyses and demonstrably conservative analyses using the approaches in Section 3.6. These approaches, based on a combination of using of the recurrence intervals for the design-basis floods and analyzing the effectiveness of mitigation measures to prevent core damage, have usually shown that the contribution to CDF is insignificant.

The collective experience with PRA external-flooding analysis is limited. Because of this limited experience, and the unavailability of any detailed methodology guidance documents, the analyst team may need to improvise its approach to external flooding analysis following the overall methodology requirements in this Section. Given the above, an extensive peer review is very important if an analysis under this Section is undertaken.

The technical requirements for external flooding PRA including local precipitation are similar, with adaptations, to those for internal-flooding PRA and seismic PRA. The major elements of the PRA methodology are flooding hazard analysis, flooding fragility analysis (involving analysis of flooding pathways and water levels), and systems analysis including quantification. The analyst familiar with internal-flooding PRA and/or

seismic PRA but unfamiliar with external flooding PRA methods should refer both to the section on internal-flooding PRA in the ASME internal-events-PRA standard (ASME, 2000) and also to the seismic-PRA Requirements and Commentary herein (Section 3.4) and to Appendix A herein ("Seismic PRA Methodology"). Specifically, some aspects of external flooding PRA, especially concerning how flooding causes the failure of structures, systems and components, are similar to internal-flooding PRA.

There are several types of external-flooding phenomena that need to be considered, depending on the site (ASCE, 1998). These include both natural phenomena (high river or lake water, ocean flooding such as from high tides or wind-driven storm surges, extreme precipitation, tsunamis, seiches, flooding from landslides, etc.), and man-made events (principally failures of dams, levees, and dikes). It is also important to consider rational probabilistic-based combinations of the above phenomena.

It is assumed here that the analyst team has employed screening methods (see Section 3.6) to eliminate from consideration those external-flooding phenomena that are not important at the site under study, and therefore that the requirements in this section will be used to analyze only those flooding phenomena that have not been screened out. It is further assumed here that the external-flooding-PRA analysis team possesses an internal-events full-power Level 1 and Level-2-LERF PRA, developed either prior to or concurrently with the external flooding PRA; that this internal-events PRA is used as the basis for the external-flooding-PRA systems model; and that the technical basis for the internal-events full-power PRA is the ASME PRA standard (ASME, 2000).

As mentioned above, external-flooding risks are generally not found to be important contributors to overall risk at nuclear power plants. One major reason is that the siting requirements are intended to assure this outcome, and by-and-large they have been successful in that regard (NRC, 1971, 1973a, 1973b, 1976a, 1976b, 1976b, 1996). Another key reason is that most large external floods occur only after significant warning time or over a long enough duration to allow the plant operating staff to take appropriate steps to secure the plant and its safety-related structures, systems and components. The PRA analysis team is therefore urged to take as much credit for warning time and compensatory actions as the plant's planning and procedures allow (see REQ. FLOOD-C1 below).

References that are useful in developing an external flooding PRA include (NRC, 1983), (Brookhaven, 1985), (Kimura and Budnitz, 1987), and (Budnitz and Lambert, 1990).

3.9.2 EXTERNAL FLOODING-PRA: TECHNICAL REQUIREMENTS

The external flooding-PRA technical requirements consist of four High-Level Requirements, under which are organized the several Supporting Technical Requirements, as follows:

HIGH-LEVEL REQUIREMENT A: HAZARD

(HLR-FLOOD-A): The frequency of external flooding at the site SHALL be based on a site-specific probabilistic hazard analysis (existing or new) that reflects recent available site-specific information. The external-flooding hazard analysis SHALL use an accepted methodology and up-to-date databases. Uncertainties in the models and parameter values SHALL be properly accounted for and fully propagated in order to obtain a family of hazard curves from which a mean hazard curve can be derived.

REQUIREMENT	COMMENTARY
(REQ.-FLOOD-A1) For extreme local precipitation, the hazard analysis SHALL use an accepted methodology and up-to-date data for the relevant phenomena. Both site-specific and regional data MAY be utilized.	**NOTE FLOOD-A1:** The usual methodologies for analyzing extreme local precipitation depend on modeling of intense local rain over very short time periods (a few minutes up to, say, an hour), coupled with computer-based stochastic studies, such as Monte-Carlo-type analysis, to generate the likelihood of several severe rains or snows in a longer period such as an 8-hour period. The limitations on these methods are principally that not enough is known about the correlations among extreme short-duration storms. Attempts have been made to develop correlations, either spatial over short distances or temporal over a few hours, based on the proposition that one can develop an understanding of how a severe storm might move (or not) in time. Site-specific historical records of precipitation may be used to predict extreme precipitation effects in much the same manner that such statistical data are used to define wind design criteria (Liu, 1991). There is a general consensus that some limited extrapolation beyond the site-specific historical record, using data from other sites, can be justified. However, for the most extreme rainfalls, say those with frequencies below about 0.001/year, the problem is that these rare events seem to involve more than one extreme phenomenon in time correlation, and the correlations are neither understood from empirical information nor modeled satisfactorily. The technical basis for such a correlation model is not understood for most sites. See (Interagency Committee, 1986) for more discussion on these methods. The NRC's guidance in this area is in Regulatory Guide 1.59.
(REQ.-FLOOD-A2) For extreme river flooding, the hazard analysis SHALL use an accepted methodology and up-to-date data for the relevant phenomena. Both site-specific and regional data MAY be utilized.	**NOTE FLOOD-A2:** The river-flooding design basis for most nuclear power plants is based on the Army Corps of Engineers "Probable Maximum Flood" (PMF). Although the method for selecting the PMF is not directly linked to its annual frequency or return period, the PMF annual frequencies are typically in the range of from 0.01 to 0.001 per year (Kimura and Budnitz, 1987). It is difficult to develop hazard curves for much larger river floods, with annual frequencies much below about 0.001 per year. One prestigious study by a government advisory committee (Interagency Committee, 1986) was very pessimistic about the technical basis for such hazard curves, but another study (National Academy of Sciences, 1988) was more optimistic, believing that methods do exist for making estimates down to the range of 0.001/year or even lower, if appropriate watershed data can be obtained. The fundamental problem is that, when extrapolations beyond the

	historical record must be made, there is a need to understand the correlations between weather phenomena, which correlations are neither understood theoretically nor reliably known from actual data at most sites. See (Kimura and Budnitz, 1987) for a discussion of these issues. The NRC's guidance in this area is in Regulatory Guide 1.59.
(REQ.-FLOOD-A3) For extreme ocean (coastal and estuary) flooding, the hazard analysis SHALL use an accepted methodology and up-to-date data for the relevant phenomena. Both site-specific and regional data MAY be utilized.	**NOTE FLOOD-A3:** For most U.S. coastal sites, the historical record, going back perhaps a century or sometimes two or more, provides a reasonable basis for a limited extrapolation beyond the actual record. For example, data for a longer section of coastline can be used to strengthen the data base, provided that care is taken to account for the specific site topography, both beneath the adjacent sea surface and on the land. The largest coastal floods sometimes involve the coincident arrival of a large storm surge when the tides are also very high, and it is necessary to use a joint probability distribution to account for this. Unfortunately, the correlations are not well understood for the largest storms. This presents a major difficulty for analyses that attempt to extrapolate well beyond the historical record (say, beyond about one order-of-magnitude). Various extreme-value distributions have been used (see St. Lucie PRA, 1987; Kimura and Budnitz, 1987).
(REQ.-FLOOD-A4) For extreme lake flooding, the hazard analysis SHALL use an accepted methodology and up-to-date data for the relevant phenomena. High water levels, surges, and wind-wave effects SHALL be considered.	**NOTE FLOOD-A4:** In the U.S., the issue of extreme lake flooding arises mostly for the several nuclear power plants located on the Great Lakes, where the problem is principally due to the possible (but rare) combination of several effects such as storm-driven wave run-up, wind-generated waves, and an unusually high lake level. For the Great Lakes, only slightly more than 100 years of reliable data exist. (For other lakes, the record may be somewhat longer.) Effects of extreme winds, including both wind-driven waves and wind setup along the shore, are often much larger than the variations in the lake levels themselves (see Kimura and Budnitz, 1987). Theoretical analysis of wind-wave effects is reasonably well grounded, and can support modest extrapolations beyond the historical record when local subsurface topographical features are accounted for.
(REQ.-FLOOD-A5) For extreme tsunami flooding, the hazard analysis SHALL use an accepted methodology and up-to-date data for the relevant phenomena. Both site-specific and regional or ocean-wide data MAY be utilized.	**NOTE FLOOD-A5:** The historical data base for tsunamis extends for several hundred years in both the Pacific and Atlantic Ocean basins, with less reliable historical data going back somewhat further. Given a distant tsunami arriving at a specific location, it is feasible to determine how large the tsunami-induced flood will be, taking into account the local offshore subsurface topography. Usually, an engineering analysis is sufficient to assure that tsunami effects will not be troublesome at a specific U.S. site; if a probabilistic (numerical) analysis of the hazard is required, the uncertainties are often large.
(REQ.-FLOOD-A6) For flooding caused by the failure of a dam, levee, or dike, the hazard analysis SHALL use an accepted methodology and up-to-date data for the failure probabilities and effects.	**NOTE FLOOD-A6:** Several generic data bases exist on U.S. dam failures, categorized by the different dam types (earthfill dams, concrete dams, etc.) See (Vanmarke and Bohnenblust, 1982; McCann and Hatem, 1985). These data bases must be used with care, depending on how closely the specific dam fits into the data base. The mean failure rate for all U.S. dams is in the range between about 10^{-4} and 10^{-5} per year (Kimura and Budnitz, 1987). However, for some modern dams with extensive engineering, values below 10^{-5}/year have been quoted (McCann and Boissonnade, 1988), while for older, poorly constructed dams values near 10^{-3}/year could be appropriate. An accurate and useful probabilistic analysis of any specific dam would require detailed engineering evaluations.

HIGH-LEVEL REQUIREMENT B: FRAGILITIES

(HLR-FLOOD-B): A flooding fragility evaluation SHALL be performed to estimate plant-specific, realistic flooding fragilities for those structures, systems, and components whose failure may contribute to core damage and/or large early release.

(REQ. FLOOD-B1) Flood fragilities of structures and exposed equipment (low-lying equipment on the site, intake and ultimate-heat-sink equipment, etc.) SHALL be evaluated using an accepted methodology and plant-specific data. The findings of a plant walkdown SHALL be considered in this evaluation.	**NOTE FLOOD-B1:** Flood-caused failure of equipment is typically due to immersion, although in some instances, particularly applicable to structures, the failure may be due to flow-induced phenomena. The analyst needs to account for the ability to survive and to function of each equipment item susceptible to flooding. Usually, it assumed that equipment submerged by the flood waters and not specially protected will "fail," meaning that it will fail to perform its safety function. Account needs to be taken of the fact that with sufficient warning times, the plant staff can secure equipment in a safe configuration. Further, the analysis must account for whether the "failure" of an item of equipment would leave it in a fail-safe position. Also, flood waters may only partially submerge an item of equipment, so the analysis must determine how much partial submersion would be sufficient to cause the "failure." Failure of structures could be overall, such as due to a foundation failure, or local, such as failure of a wall or barrier leading to leakage or major flooding through the wall or barrier. Most nuclear power plant structures have excellent resistance to flooding, by design. Major vulnerabilities have sometimes been identified for certain structures, but usually the equipment housed therein is not crucial to overall plant safety. The walkdown SHOULD play a major role in identifying potential problems, supplemented by an evaluation of structural drawings. Fragility analysis for both capacity and demand MAY be based on standard methodology (ASCE, 1998).

HIGH-LEVEL REQUIREMENT C: SYSTEMS ANALYSIS AND QUANTIFICATION

(HLR-FLOOD-C): The external-flooding-PRA systems model SHALL include all important flood-caused initiating events that can lead to core damage or large early release. The model SHALL be adapted from the internal-events, full-power PRA systems model to incorporate flood-analysis aspects that are different from the corresponding aspects in the full-power, internal-events PRA systems model.

NOTE: In special circumstances, it is acceptable to develop an ad-hoc systems model tailored especially to the particular flooding phenomenon being analyzed, instead of starting with the internal-events systems model and adapting it. If this approach is used, it is especially important that a peer review be undertaken that concentrates on this aspect.

(REQ. FLOOD-C1) Accident sequences initiated by external flooding SHALL be assessed to estimate CDF and LERF contribution. The analysis SHALL consider the appropriate flooding hazard curves and the fragilities of structures and equipment. The systems-analysis approach for flood-initiated accident sequences SHALL	**NOTE FLOOD-C1:** The external-flooding-PRA systems-analysis model is almost always based on the internal-events full-power PRA systems model, to which are added basic failure events derived from the information developed in the flooding fragility analysis. Considerable screening out and trimming of the internal-events systems model is also common, where appropriate. The analysis consists of developing event trees and fault trees in which the initiating event can be either the extreme flood itself or a transient or LOCA induced by the extreme flood. Various accident sequences that lead to core damage or large early release are identified and their conditional probabilities of occurrence calculated. The frequency of core damage or large early release is obtained by a convolution over the relevant range of hazard intensities. The procedure for determining the accident sequences is similar to that used in seismic-PRA systems analysis, and following the Requirements therein represents one acceptable approach, after they are adapted to apply to the external-flooding-PRA situation. (See the Requirements and Commentary in Section 3.4.2, and the

use the same general approach used for developing internal-flooding initiated sequences (see the corresponding Requirements in the ASME standard, (ASME, 2000)), and/or used for developing seismic-initiated sequences in seismic PRA (see the corresponding Requirements in Section 3.4.2 herein). One acceptable approach is to follow either those internal-flooding-PRA Requirements or those seismic-PRA Requirements, after adapting them to the external-flooding-PRA situation.	discussion about seismic PRA methods in Appendix A). Other factors to be considered include non-flooding-related unavailabilities or failures of equipment, operator errors, any warning time available to take mitigating steps, the possibility of recovery actions by operators and replacement by substitutes to accomplish the needed function; and the likelihood of common-caused failures. The clogging of intake structures and other flow paths by debris related to the flooding must also be considered, and a walkdown is important to assure that this issue has been evaluated properly. One key consideration is that most large external floods occur only after significant warning time or extended duration has allowed the plant operating staff to take appropriate steps to secure the plant and its key equipment. This warning time and the typical situation in which the plant grade is well above any credible flooding phenomena are the principal reasons why external-flooding risks are not often found to be important contributors to overall risks. The analysis team is therefore urged to take as much credit for warning time and compensatory actions as the plant's planning and procedures allow.
(REQ. FLOOD-C2) The integration-quantification SHALL account for the uncertainties in each of the inputs, and SHALL account for all dependencies and correlations.	**NOTE FLOOD-C2:** The usefulness of the "final results" of the PRA for external flooding are dependent on performing enough assessment to understand the uncertainties, dependencies, and correlations, and to account for them quantitatively if they are important.

HIGH-LEVEL REQUIREMENT D: DOCUMENTATION

(HLR-FLOOD-D): The external-flooding-PRA analysis SHALL be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

(REQ. FLOOD-D1) The documentation SHALL meet the general documentation requirements in Section 7.

(REQ. FLOOD-D2) The documentation SHALL include a description of the specific methods used for determining the external flooding hazard curves, including the scientific interpretations that are the basis for the inputs and results.

(REQ. FLOOD-D3) The documentation SHALL describe the specific adaptations made to the internal-events PRA model to produce the external-flooding-PRA model, and their motivation.

(REQ. FLOOD-D4) The documentation SHALL describe the methodologies used to quantify the flooding-caused fragilities of SSCs, together with key assumptions.

(REQ. FLOOD-D4) The documentation SHALL provide a detailed list of SSC fragility values that

includes the method of analysis, the dominant failure modes(s), the sources of information, and the location of each SSC.
(REQ. FLOOD-D5) The documentation SHALL discuss the basis for the screening-out of any SSCs that is done on a basis other than the SSC being located where flooding does not occur.

SECTION 4 ---- PRA CONFIGURATION CONTROL

4.1 General Requirement

PRA configuration control SHALL be accomplished according to the requirements found in Section 5 ("PRA Configuration Control") of the ASME PRA Standard (ASME, 2000).

SECTION 5 ---- PRA PEER REVIEW

5.1 General Requirement

Peer review of a PRA, seismic margin assessment (SMA), or other external-events analysis covered under this Standard SHALL be performed according to the requirements found in Section 6 ("Peer Review") of the ASME PRA Standard (ASME, 2000), except where the specific requirements therein do not apply because the ASME Standard covers internal-initiating-events PRA whereas this ANS standard covers external events PRA and SMA.

In addition, specific additional peer-review requirements for seismic PRA, seismic-margin-assessments, and PRAs of other external events are found next, in Sections 5.2, 5.3, and 5.4 respectively.

The purpose of the peer review is fundamentally to provide an independent review of the PRA or SMA. This means reviewing the analysis vis-à-vis the applicable Requirements in the Standard. The composition and qualifications of the peer review team are important, as is its independence; these aspects are covered in the ASME Standard's requirements (ASME, 2000) that are incorporated here by reference. Other process issues, including the need for a team leader and the need for a methodology for the review, are also covered in the ASME Standard.

The fundamental task of the peer review is succinctly stated in Section 6.1 of the ASME Standard. This task is identical for the peer review required herein: "The peer review shall assess the PRA Elements contained in Section 4 to the extent necessary to determine if the methodology and its implementation meet the requirements of this Standard. The peer review need not assess all aspects of the PRA against all Section 4 requirements; however, enough aspects of the PRA shall be reviewed for the reviewers to achieve consensus on the adequacy of methodologies and their implementation for each PRA Element." [Note that ASME's Section 4 contains the PRA technical requirements.]

5.2 Peer Review Requirements for Seismic PRA

(REQ. SPRA-PR-1) The peer review team SHALL have combined experience in the areas of systems engineering, seismic hazard, seismic capability engineering, and seismic PRAs or seismic margin methodologies. The reviewer(s) focusing on the seismic fragility work SHALL have successfully completed the SQUG Walkdown Screening and Seismic Evaluation Training Course (SQUG, 1993) or equivalent, or SHALL have demonstrated equivalent experience in seismic walkdowns.
(REQ. SPRA-PR-2) The peer review team SHALL evaluate whether the seismic hazard study used in the PRA is appropriately specific to the site and has met the relevant requirements of the Standard.
(REQ. SPRA-PR-3) The peer review team SHALL evaluate whether the seismic initiating events are properly identified, the SSCs are properly modeled, and the accident sequences are properly quantified. The review team SHALL ensure that the Seismic Equipment List is reasonable for the plant considering the reactor type, design vintage, and specific design.
(REQ. SPRA-PR-4) The peer review team SHALL evaluate whether the seismic response analysis used in the development of seismic fragilities meets the relevant requirements of the Standard. Specifically, the review SHOULD focus on the input ground motion (i.e., spectrum or time history), structural modeling including SSI effects, parameters of structural response (e.g., structural damping, soil damping), and the reasonableness of the calculated seismic response.
(REQ. SPRA-PR-5) The peer review team SHALL review the seismic walkdown of the plant in order to assure the validity of the findings of the Seismic Review Team on screening, seismic spatial interactions, and the identification of critical failure modes.
(REQ. SPRA-PR-6) The peer review team SHALL evaluate whether the methods and data used in the fragility analysis of SSCs are adequate for the purpose. The review team SHOULD perform independent fragility calculations of a selected sample of components covering different categories and contributions to CDF and LERF.
(REQ. SPRA-PR-7) The peer review team SHALL evaluate whether the seismic quantification method used in the seismic PRA is appropriate and provides all the results and insights needed for risk-informed decisions. The review SHALL focus on the CDF and LERF estimates and uncertainty bounds, and on the dominant risk contributors.

5.3 Peer Review Requirements for Seismic Margin Assessment

(REQ. SMA-PR-1) The peer review team SHALL have combined experience in the areas of systems engineering, seismic hazard, seismic capability engineering, and seismic PRAs or seismic margin methodologies. The reviewer(s) focusing on the seismic capability work SHALL have successfully completed the SQUG Walkdown Screening and Seismic Evaluation Training Course (SQUG, 1993) or equivalent or SHALL have demonstrated experience in seismic walkdowns.
(REQ. SMA-PR-2): The peer review team SHALL evaluate whether the selection of the RLE used in the SMA is appropriately specific to the site and has met the relevant requirements of the Standard.
(REQ. SMA-PR-3) The peer review team SHALL evaluate whether the success paths are chosen properly and reflect the systems and operating procedures in the plant, and that the preferred and alternative paths are reasonably redundant. The review team SHALL ensure that the Safe Shutdown Equipment List is reasonable for the plant considering the reactor type, design vintage, and specific design.
(REQ. SMA-PR-4) The peer review team SHALL evaluate whether the seismic response analysis used in the development of seismic margins meets the relevant requirements of the Standard. Specifically, the review SHOULD focus on the input ground motion (i.e., spectrum or time history), structural modeling including SSI effects, parameters of structural response (e.g., structural damping, soil damping), and the reasonableness of the calculated seismic response for the RLE input.
(REQ. SMA-PR-5) The peer review team SHALL review the seismic walkdown of the plant in order to assure the validity of the findings of the Seismic Review Team on screening, seismic spatial interactions and identification of critical failure modes.
(REQ. SMA-PR-6) The peer review team SHALL evaluate whether the methods and data used in the seismic margin analysis of components are adequate for the purpose. The review team SHOULD perform independent HCLPF calculations of a selected sample of components covering different categories and contributions to plant margin.
(REQ. SMA-PR-7) The peer review team SHALL evaluate whether the seismic margin assessment method used is appropriate and provides all the results and insights needed for risk-informed decisions. The review SHOULD focus on the HCLPF capacities of components and success paths, and on the dominant contributors to seismic margins.

5.4 Peer Review Requirements for PRA of an "Other" External Event

(REQ. OTHER-PR-1) The peer review team SHALL have combined experience in the areas of systems engineering, evaluation of the hazard for the relevant external event, and evaluation of how the external event could damage the nuclear plant's SSCs.
(REQ. OTHER-PR-2) The peer review team SHALL evaluate whether the external-event hazard used in the PRA is appropriately specific to the site and has met the relevant requirements of the Standard.
(REQ. OTHER-PR-3) The peer review team SHALL evaluate whether the initiating events postulated to be caused by the external event are properly identified, the SSCs are properly modeled, and the accident sequences are properly quantified.
(REQ. OTHER-PR-4) The peer review team SHALL evaluate whether the methods and data used in the "fragility" analysis of SSCs are adequate for the purpose and meet the relevant requirements of the Standard. The review team SHOULD perform independent fragility calculations of a selected sample of SSCs covering different categories and contributions to CDF and LERF.
(REQ. OTHER-PR-5) The peer review team SHALL review the walkdown of the plant in order to assure the validity of the findings of the analysis in terms of screening, any spatial interactions, and the identification of critical failure modes.
(REQ. OTHER-PR-6) The peer review team SHALL evaluate whether the quantification method used in the PRA is appropriate and provides all of the results and insights needed for risk-informed decisions. If the analysis contains screening assumptions, or assumptions that the analysis team claims to be demonstrably conservative, the peer review team SHALL review the validity of these assumptions. The review SHALL focus on the CDF and LERF estimates and uncertainty bounds, and on the dominant risk contributors.

SECTION 6 ---- RISK ASSESSMENT APPLICATION PROCESS

6.1 General Requirement

The risk-assessment application process covered under this Standard SHALL be performed according to the requirements found in Section 3 ("Risk Assessment Application Process") of the ASME PRA Standard (ASME, 2000), except where the specific requirements therein do not apply because the ASME Standard covers internal-initiating-events PRA whereas this ANS standard covers external events PRA.

6.2 Applications Using a Seismic-Margin Assessment or a Screening/Conservative Analysis

Although Section 3 ("Risk Assessment Application Process") of the ASME Standard (ASME, 2000) was written with a PRA in mind, the requirements therein apply equally well to applications using a Seismic Margin Assessment, or a screening or demonstrably conservative analysis, that meets this Standard.

SECTION 7 – DOCUMENTATION REQUIREMENTS

7.1 General Documentation Requirements

INTRODUCTORY NOTE: In the documentation requirements below, the phrase "PRA" is intended to include a screening or bounding (demonstrably conservative) analysis, a seismic margin assessment, or any other analysis covered by this standard, as well as a full probabilistic risk assessment (PRA).

To meet this Standard, a PRA requires appropriate documentation. This section contains several general documentation requirements. In addition, under the requirements for each external event, there are a few additional documentation requirements specific to that external event.

The general documentation requirements follow.

NOTE SUPPORTING THE GENERAL DOCUMENTATION REQUIREMENTS: When developing documentation, it is important to consider the overall broad objective of this Standard, which is to facilitate risk-informed applications using the PRA or seismic-margin assessment covered by the Standard. In the context of this broad objective, there are three broad aims of the documentation requirements:

(i) It is important that the documentation be sufficient to enable the peer reviewers to understand how the various requirements have been met..

(ii) When the PRA is modified or updated, the individuals doing the work may not be the same as those who did the original work. Without adequate guidance, this updating cannot be accomplished well enough that the PRA will continue to meet the Standard.

(iii) It is important that the peer reviewers, and in fact the PRA analysis team itself, be able (based on the documentation) to reproduce the analysis and results, even though the work of reproducing the results is seldom undertaken except for narrow parts of the PRA.

In furtherance of the above aims, the documentation needs to cover, sufficiently to meet the above aims in the context of the contemplated applications, descriptions of the methodologies used; the major assumptions; the sources and limitations of the data and models used; the major final results and important intermediate results; the factors that influence these results; and the underlying technical concepts that are the basis for the inputs and results.

(REQ. DOC-1) The documentation SHALL be sufficient to enable peer review according to the peer-review requirements in Section 5. Specifically, the documentation SHALL be sufficient to enable the peer reviewers to understand how the various requirements have been met in each technical area of the PRA.

(REQ. DOC-2) The documentation SHALL be sufficient to facilitate the modification or updating of the PRA at a later date by a different group of cognizant individuals.

(REQ. DOC-3) The documentation SHALL describe the analysis performed, in sufficient detail to enable peer reviewers and analysts other than the original analysis team to understand the methodologies, assumptions, models, and data used to perform each part of the PRA.

(REQ. DOC-4) The documentation SHALL describe the important final results and insights of the PRA, along with a selection of important intermediate results.

(REQ. DOC-5) The documentation SHALL describe the major contributors to the uncertainties in each of the important final PRA results and insights.

(REQ. DOC-6) The documentation SHALL describe the motivations for and the results of important sensitivity analyses performed for the PRA.

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APPENDIX A

SEISMIC PROBABILISTIC RISK ASSESSMENT METHODOLOGY

A.1 Background

Seismic PRAs have been conducted for over 50 nuclear power plants worldwide in the last 20 years. The methodology has been well established and the necessary data on the parameters of the PRA model have been generally collected. Detailed description of the procedures used in seismic PRA is given in several published reports and technical papers: PRA Procedures Guide (NRC, 1983), PSA Procedures Guide (Brookhaven, 1985), (NRC, 1991a), (Reed and Kennedy, 1994), (Ravindra, 1995), (Budnitz et al.,1997), (Budnitz, 1998), and (Kennedy, 1999). *[See the citation list in Section 8 of the Standard.]*

Seismic PRA is different from an internal event PRA in several important ways:

- Earthquakes could cause initiating events different from those considered in the Internal Event PRA.
- All possible levels of earthquakes along with their frequencies of occurrence and consequential damage to plant systems and components should be considered.
- Earthquakes could simultaneously damage multiple redundant components. This major common cause effect should be properly accounted for in the risk quantification.

The objectives of a seismic PRA include:

- Develop an appreciation of accident behavior (i.e., consequences and role of operator),
- Understand the most likely accident sequences induced by earthquakes (useful for accident management),
- Gain an understanding of the overall likelihood of core damage induced by earthquakes,
- Identify the dominant seismic risk contributors,

- Identify the range of peak ground acceleration that contributes significantly to the plant risk (this is helpful in making judgements on seismic margins), and

- Compare seismic risk with risks from other events and establish priorities for plant upgrading.

A.2 Key Elements of Seismic PRA

The key elements of a seismic PRA can be identified as

- **Seismic Hazard Analysis:** to develop frequencies of occurrence of different levels of ground motion (e.g., peak ground acceleration) at the site.

- **Seismic Fragility Evaluation:** to estimate the conditional probability of failure of important structures and equipment whose failure may lead to unacceptable damage to the plant (e.g., core damage); plant walkdown is an important activity in conducting this task.

- **Systems/Accident Sequence Analysis:** modeling of the various combinations of structural and equipment failures that could initiate and propagate a seismic core damage sequence.

- **Risk Quantification:** Assembly of the results of the seismic hazard, fragility, and systems analyses to estimate the frequencies of core damage and plant damage states. Assessment of the impact of seismic events on the containment and consequence analyses, and integration of these results with the core damage analysis to obtain estimates of seismic risk in terms of effects on public health (e.g., early deaths and latent cancer fatalities).

The process is shown schematically in Figure A-1 and is described in detail in (NRC, 1983). Following is a brief description of the four steps utilized in the seismic PRA process.

A.2.1 Seismic Hazard Analysis

Seismic hazard is usually expressed in terms of the frequency distribution of the peak value of a ground motion parameter (e.g., peak ground acceleration) during a specified time interval. The different steps of this analysis are as follows:

1. Identification of the sources of earthquakes, such as faults and seismotectonic provinces.

2. Evaluation of the earthquake history of the region to assess the frequencies of occurrence of earthquakes of different magnitudes or epicentral intensities.
3. Development of attenuation relationships to estimate the intensity of earthquake-induced ground motion (e.g., peak ground acceleration) at the site.
4. Integration of the above information to estimate the frequency of exceedance for selected ground motion parameters

The hazard estimate depends on uncertain estimates of attenuation, upperbound magnitudes, and the geometry of the postulated seismic sources. Such uncertainties are included in the hazard analysis by assigning probabilities to alternative hypotheses about these parameters. A probability distribution for the frequency of occurrence is thereby developed. The annual frequencies for exceeding specified values of the ground motion parameter are displayed as a family of curves with different probabilities or with different fractiles (Figure A-2).

A mean estimate of the frequency of exceedance at any peak ground acceleration is obtained as the weighted sum of the frequencies of exceedance at this acceleration given by the different hazard curves; the weighting factor is the probability assigned to each hazard curve. Thus, the PSHA embeds uncertainties in the core of the methodology and results are expressed in terms of likelihood – estimated probabilities in a given time period or estimated frequencies - that earthquakes producing various sizes of ground motion will occur at a given site. These results reflect two different classes of uncertainties. Lack-of knowledge uncertainties or *epistemic* uncertainties arise from imperfect scientific understanding which can, in principle, be further reduced through additional research and acquisition of data. The *aleatory* or random uncertainties are those uncertainties that, for all practical purposes, can not be known in detail or can not be reduced. Although, in some applications, it may not be necessary to display this distinction in the nature of uncertainties (e.g., NUREG-1407 (NRC, 1991a) allowed the use of the mean hazard curve which includes combined uncertainties instead of the full family of hazard curves for identification of vulnerabilities and ranking dominants sequences and contributors), it is crucial that in the development of a PSHA this distinction is maintained to understand and communicate the sources of uncertainties.

For further details on seismic hazard analysis methods, the reader is referred to (Budnitz et.al., 1997) and (Reiter, 1990). Typical results of a PSHA include families of seismic hazard curves in terms of peak ground acceleration or spectral acceleration values at different frequencies, and site-specific ground motion response spectra.

A.2.2 Seismic Fragility Evaluation

The methodology for evaluating seismic fragilities of structures and equipment is documented in Kennedy and Ravindra (1984) and Reed and Kennedy (1994). Seismic fragility of a structure or equipment item is defined as the conditional probability of its failure at a given value of the seismic input or response parameter (e.g., peak ground acceleration, stress, moment, or spectral acceleration). Seismic fragilities are needed in a PRA to estimate the conditional probabilities of occurrence of initiating events (i.e., loss of emergency AC power, loss of forced circulation cooling systems) and the conditional failure probabilities of different mitigating systems (e.g., auxiliary feedwater system).

The objective of fragility evaluation is to estimate the ground motion capacity of a given component and its uncertainty. This capacity is defined either in terms of average spectral acceleration value or peak ground acceleration (PGA) value for which the seismic response of a given component located at a specified point in the structure exceeds the component's resistance capacity, resulting in its failure. Although the average spectral acceleration is preferable, PGA has been used in many seismic PRAs and is acceptable provided that the uncertainties in the spectral shape are not too large. The ground acceleration capacity of the component is estimated using information on plant design bases, responses calculated at the design analysis stage, as-built dimensions, and material properties. Because there are many variables in the estimation of this ground acceleration capacity, component fragility is described by a family of fragility curves; a probability value is assigned to each curve to reflect the uncertainty in the fragility estimation. This family of fragility curves may be described by three parameters; the median acceleration capacity A_m, and logarithmic standard deviations, β_R and β_U, for randomness and uncertainty.

In seismic margin assessments, the HCLPF capacity is used as a measure of seismic margin. "HCLPF" is an acronym for high-confidence-of-low-probability-of-failure. HCLPF capacity is a ground motion value at which there is 95% confidence that the probability of failure is less than 5%. If the fragility curve is described by the median, A_m, the randomness, β_R, and uncertainty, β_U, where the βs are logarithmic standard deviations, the HCLPF may be computed from:

$$\text{HCLPF} = A_m \exp\left[-1.65\left(\beta_R + \beta_U\right)\right] \qquad \text{(A-1)}$$

An example family of seismic fragility curves is shown in Figure A-3. The component is designed for a Safe Shutdown Earthquake of 0.17g. Its median capacity for overturning (resulting in failure of attached piping) is calculated as 0.87g, the logarithmic standard deviations β_R and β_U are estimated as 0.25 and 0.35, respectively. The HCLPF capacity of the component is calculated from Equation A-1 as 0.32g. Figure A-3 shows the median, 5% confidence and 95% confidence fragility curves. The mean fragility curve is also shown which is obtained from the lognormal probability distribution with A_m and $\beta_c = (\beta_R^2 + \beta_U^2)^{1/2}$. For some applications, exclusive use of mean fragility curves is judged to be sufficient.

Seismic fragilities of structures and equipment are calculated using many sources: plant specific seismic design and qualification data, fragility test data, generic seismic qualification test data and earthquake experience data. In a typical seismic PRA, over 500 components are identified as requiring evaluations. A plant walkdown is performed to screen out a large number of these components based on their generically high seismic capacities and on lack of obvious seismic deficiencies (such as poor anchorage and inadequate lateral support) and spatial interactions (e.g., a non-seismically qualified component failing and falling on a component modeled in the seismic PRA). For the remaining components, seismic fragilities are calculated using one or more of the data sources.

A.2.3 Analysis of Plant Systems and Accident Sequences

Frequencies of severe core damage and radioactive release to the environment are calculated by combining plant logic with component fragilities and seismic hazard

estimates. Event and fault trees are constructed to identify the accident sequences that may lead to severe core damage and radioactive release.

The plant systems and sequence analyses used in seismic PRAs are based on the PRA Procedures Guide (NRC, 1983) and can generally be summarized as follows:

1. The analyst constructs fault trees reflecting (a) failures of key system components or structures that could initiate an accident sequence and (b) failures of key system components or structures that would be called on to stop the accident sequence.

2. The fragility of each such component (initiators and mitigators) is estimated.

3. Fault trees are used to develop Boolean expressions for severe core damage that lead to each distinct plant damage state sequences.

4. Considering possible severe core damage sequences and containment mitigation systems (e.g., fan coolers, containment sprays, and containment), Boolean expressions are developed for each release category.

As an example, the Boolean expression for severe core damage in the Zion Probabilistic Safety Study is:

$$M_S = 4+8+10+14+17+21+(12+22+26)*9 \qquad \text{(A-2)}$$

The numbers represent components for which seismic fragilities have been developed. The symbols "+" and "*" indicate "OR" and "AND" operations, respectively. Plant level fragility curves are obtained by combining the fragilities of individual components according to Equation A-2, using either Monte Carlo simulation or numerical integration. The plant level fragility is defined as the conditional probability of severe core damage as a function of the peak ground acceleration at the site. The uncertainty in plant level fragility is displayed by developing a family of fragility curves; the weight (probability) assigned to each curve is derived from the fragility curves of components appearing in the specific plant damage state accident sequence.

A.2.4 Evaluation of Core Damage Frequency

Plant level fragilities are convolved with the seismic hazard curves to obtain a set of doublets for the plant damage state frequency,

$$\{<p_{ij}, f_{ij}>\} \tag{A-3}$$

where f_{ij} is the seismically-induced plant damage state frequency and p_{ij} is the discrete probability of this frequency.

$$p_{ij} = q_i p_j \tag{A-4}$$

$$f_{ij} = \int_0^\infty f_i(a) \frac{dH_j}{da} da \tag{A-5}$$

Here, H_j represents the j^{th} hazard curve, f_i the i^{th} plant damage fragility curve; q_i is the probability associated with the i^{th} fragility curve and p_j is the probability associated with the j^{th} hazard curve.

The above equations state that the convolution between the seismic hazard and plant level fragility is carried out by selecting hazard curve j and fragility curve i; the probability assigned to the plant damage frequency resulting from the convolution is the product of the probabilities p_j and q_i assigned to these two curves. The convolution operation given by Equation A-5 consists of multiplying the occurrence frequency of an earthquake peak ground acceleration between a and a + da (obtained as the derivative of H_j with respect to a) with the conditional probability of the plant damage state, and integrating such products over the entire range of peak ground accelerations from 0 to ∞. In this manner, a probabilistic distribution on the frequency of a plant damage state can be obtained.

Severe core damage occurs if any one of the plant damage states occurs. By probabilistically combining the plant damage states, the plant level fragility curves for severe core damage are obtained. Integration of the family of fragility curves over the family of seismic hazard curves yields the probability distribution function of the

occurrence frequency of severe core damage. By extending this procedure, probability distribution functions of the occurrence of different release categories are obtained.

A.3 Outputs of Seismic PRA

The outputs of a seismic PRA are:

- Seismic fragilities of components and seismic margins
- Seismic fragilities of accident sequences and seismic margins
- Seismic accident sequence frequencies and uncertainty distributions
- Impact of non-seismic unavailabilities on seismic risk
- Identification of dominant risk contributors: components, systems, sequences and procedures.
- Distribution on range of accelerations contributing to seismic risk
- Risk reduction as a function of seismic upgrading to aid in backfit decisions.

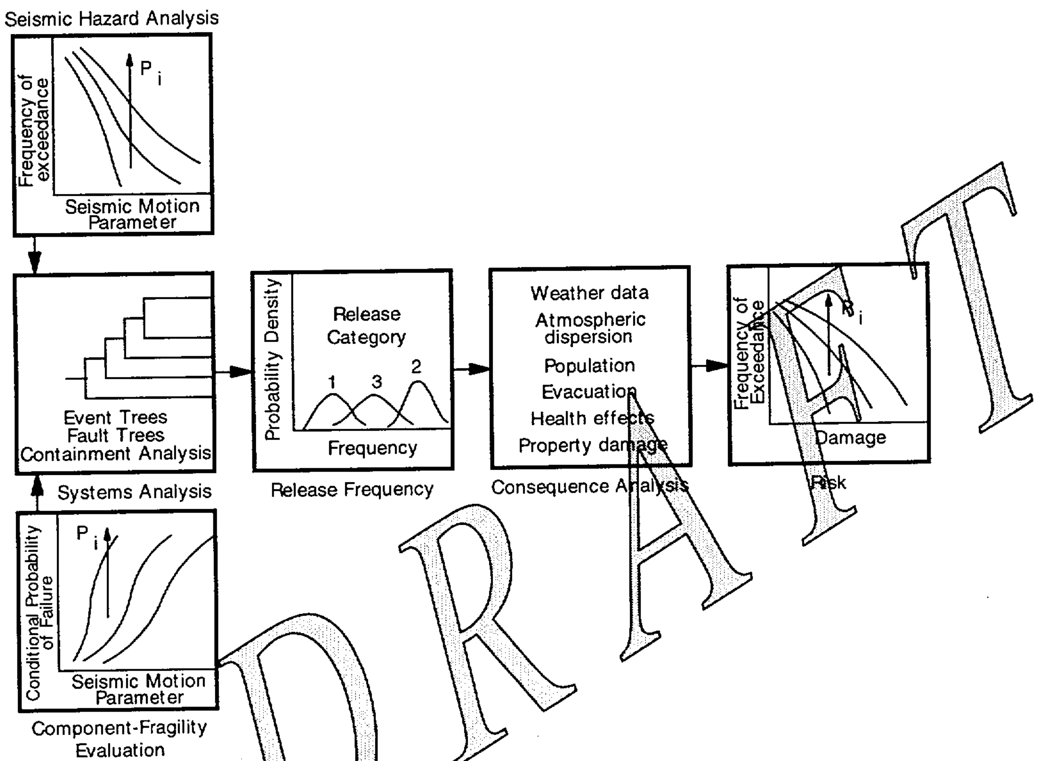

Fig. A-1: Schematic Overview of a Seismic PRA
(P_i indicates the subjective probability weight assigned to each curve $_i$)

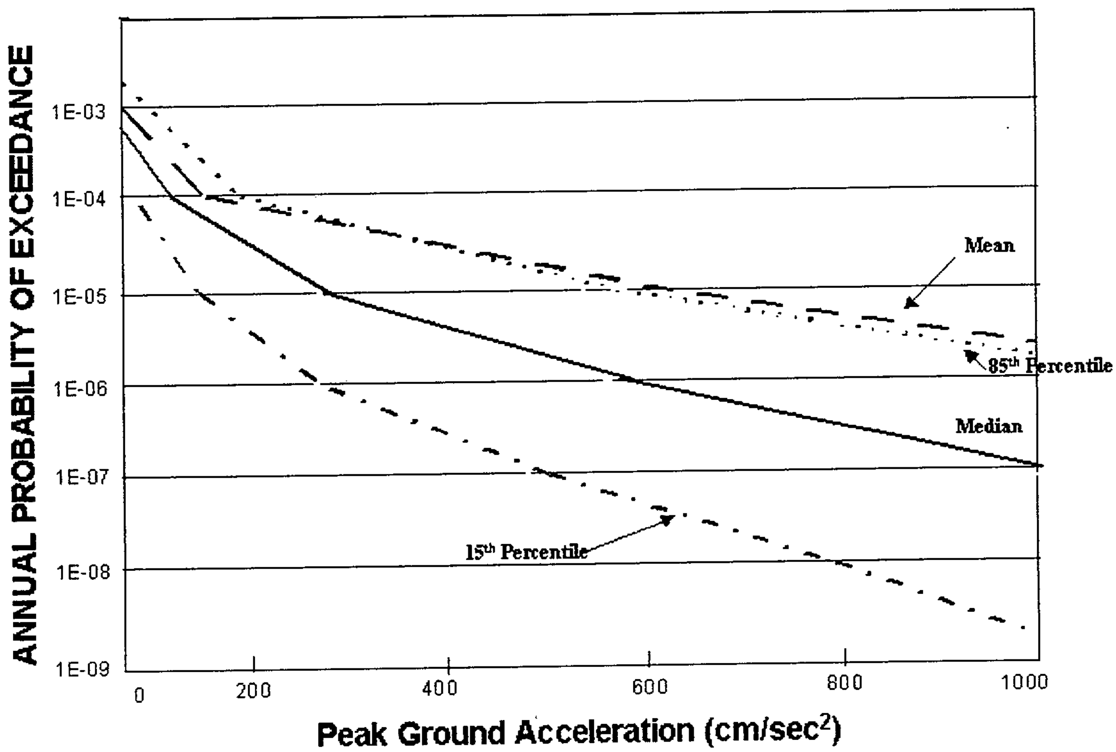

Fig. A-2: Typical Seismic Hazard Curves for a Nuclear Power Plant Site

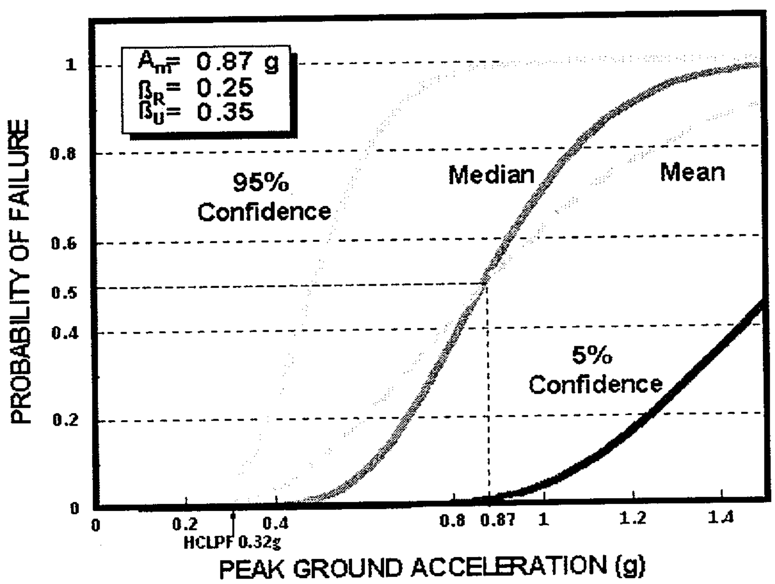

Fig. A-3: **Typical Family of Fragility Curves for a Component**

APPENDIX B
LIST OF EXTERNAL EVENTS REQUIRING CONSIDERATION
(see REQ. OTH-A1)
Adapted From NUREG/CR-2300, PRA Procedures Guide (Ref. NRC, 1983)

External Event	Applicable Screening Criteria: (REQ.OTH-B1 describes these five Criteria)	Remarks
Aircraft Impacts	--	Site specific; requires detailed study
Avalanche	3	Can be excluded for most sites in the United States
Biological Events	1, 5	Includes events such as □detritus□ and zebra mussels
Coastal Erosion	4, 5	Included in the effects of external flooding
Drought	1, 3	Can often be excluded where there are multiple sources of ultimate heat sink or where the ultimate heat sink is not affected by drought (e.g., cooling tower with adequately sized basin)
External Flooding	------	Site specific; requires detailed study
Extreme Winds and Tornadoes	------	Site specific; requires detailed study
Fog	1	Could, however, increase the frequency of man-made hazard involving surface vehicles or aircraft; accident data include the effects of fog
Forest Fire	1, 3	Fire cannot propagate to the site because the site is cleared; plant design and fire-protection provisions are adequate to mitigate the effects
Frost	1	Snow and ice govern
Hail	1	Other missiles govern
High Tide	4	Included under external flooding

High Summer Temperature	1	Can often be excluded where the Uultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms. Evaluation is needed of possible loss of air-cooling due to high temperatures.
Hurricane	4	Included under external flooding; wind forces are covered under extreme winds and tornadoes
Ice Cover	1, 4	Ice blockage of river included in flood; loss of cooling-water flow is considered in plant design
Industrial or Military Facility Accident	-----	Site specific; requires detailed study
Internal Flooding	---	Plant specific; requires detailed study
Landslide	3	Can be excluded for most sites in the United States; confirm through walkdown
Lightning	1	Considered in plant design
Low Lake or River Water Level	1, 5	Can often be excluded where the ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other waste-loss mechanisms
Low Winter Temperature	1, 5	Thermal stresses and embrittlement are usually insignificant or covered by design codes and standards for plant design; generally, there is adequate warning of icing on the ultimate heat sink so that remedial action can be taken.
Meteorite/Satellite Strikes	2	All sites have approximately the same frequency of occurrence
Pipeline Accident	----	Site specific; requires detailed study
Precipitation, Intense	4	Included under external and internal flooding. Roof loading nd its effect on building integrity must be checked.
Release of Chemicals from On-Site Storage	----	Plant specific; requires detailed study
River Diversion	1, 4	Considered in the evaluation of the ultimate heat sink; should diversion become a hazard, adequate storage is usually provided. Requires

		detailed site/plant study.
Sandstorm	1, 4	Included under tornadoes and winds; potential blockage of air intakes with particulate matter is generally considered in plant design
Seiche	4	Included under external flooding
Seismic Activity	--	Site specific; requires detailed study
Snow	1, 4	Plant designed for higher loading; snow melt causing river flooding is included under external flooding
Soil Shrink-Swell	1, 5	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard
Storm Surge	4	Included under external flooding
Transportation Accidents	--	Site specific; requires detailed study
Tsunami	4	Included under external flooding and seismic events
Toxic Gas	--	Site specific; requires detailed study
Turbine-Generated Missiles	1,2	Plant specific; requires detailed study
Volcanic Activity	3	Can be excluded for most sites in the United States
Waves	4	Included under external flooding

APPENDIX C

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Attachment 2

Principles/Objectives for the ASME Standard

In the risk-informed environment in which NRC and industry are currently operating, PRA results are used as one, but not the only input to a decision-making process. Depending on the specific nature of the application, PRA results can play a more or less significant role. The extent to which the PRA results influence the decision will be impacted by the confidence the decision-makers have in those results. Accordingly, development of a Standard that promotes a consistent determination of the strengths and weaknesses of a PRA will directly impact the ability of decision-makers to efficiently establish a level of confidence in the results. The requirements of such a Standard provide a reference point for determining the strengths and weaknesses and also for evaluating alternative PRA approaches. The Standard should also recognize that in some areas methodology and data enhancements will occur over the next several years.

1. The PRA Standard needs to provide well-defined criteria against which to judge the strengths and weaknesses of the PRA so that decision-makers can determine the degree of reliance that can be placed on the PRA results of interest.

2. The Standard needs to be based on current good practices as reflected in publicly available documents. The need for the documentation to be publicly available follows from the fact that the Standard may be used to support safety decisions.

3. To facilitate the use of the Standard for a wide range of applications, categories can be defined to aid in determining the applicability of the PRA for various types of applications.

4. The Standard needs to be thorough and complete in defining what is technically required and should, where appropriate, identify one or more acceptable methods.

5. The Standard needs to require a peer review process that identifies and assesses where the technical requirements of the Standard are not met. The Standard needs to assure that the peer review process:
   1. determines whether methods identified in the Standard have been used appropriately;
   2. determines that, when acceptable methods are not specified in the Standard, or when alternative methods are used in lieu of those identified in the Standard, the methods used are adequate to meet the requirements of the Standard;
   3. assesses the significance on the results and insights gained from the PRA of not meeting the technical requirements in the Standard;
   4. highlights assumptions that may significantly impact the results and provides an assessment of the reasonableness of the assumptions;
   5. is flexible and accommodates alternate peer review approaches; and
   6. includes a peer review team that is comprised of members who are knowledgeable in the technical elements of a PRA, are familiar with the plant design and operation, and are independent with no conflicts of interest.

6. The Standard needs to address the maintenance and update of the PRA to incorporate changes that can substantially impact the risk profile, so that the PRA adequately represents the current as-built and as-operated plant.

7. The Standard needs to be viewed as a living document. Consequently, it should not impede research but needs to be structured such that when improvements in our state of knowledge occur, the Standard can easily be updated.

Attachment # 3

SECY-00-0162 Attachment 1
PRA SCOPE AND TECHNICAL ATTRIBUTES

1. INTRODUCTION

Over the past 25 years a number of probabilistic risk assessments (PRAs) have been performed by both the U. S. Nuclear Regulatory Commission (NRC) and the nuclear industry. The scope, depth, and technical content of the PRAs have varied along with their purposes and uses. Results from PRAs have increasingly been used in the regulatory process, starting with generic safety issue prioritization and progressing to regulatory analysis in support of rulemaking and backfits and currently risk-informed regulation, which opens up the possibility of using PRA information in many ways not previously done.

The NRC issued a Policy Statement on the use of PRA in 1995, encouraging its use in all regulatory matters. Since that time, many uses have been implemented or undertaken, including the initiation of work to modify the reactor regulations and inspection program. As a result PRA is becoming a mainstream regulatory tool and, as such, is providing valuable input into the decision-making process regarding the design, operation and maintenance of plants. Consequently, confidence in the information derived from a PRA is an important issue. That is, the scope of the analysis must be sufficiently broad and the accuracy of the technical content must be of sufficient rigor to justify the specific results and insights from the PRA that are used to support the decision under consideration.

Each application may impose somewhat different requirements on the supporting PRA. Therefore, it is important to note what are the different risk-informed activities for which defining PRA technical acceptability is needed. Recent activities include the following:

- Risk-Inform 10 CFR Part 50: The NRC is evaluating the scope of the special treatment requirements and the technical requirements of 10CFR Part 50 and is considering revisions to them, as appropriate, based in part on risk insights obtained from PRAs.

- Reactor Oversight Process: The NRC is increasing the focus of inspection on those activities with the greatest potential impact on safety. Inspection results will routinely be evaluated to determine the risk importance of the findings. Likewise, enforcement sanctions for violations of regulatory requirements will be better linked to the safety significance of inspection findings.

- Operating Events Assessment: The NRC is continuing to evaluate the risk significance of operational events and trends in data in conjunction with risk assessments so that safety vulnerabilities can be identified, prioritized, communicated, and resolved on a timely basis.

- License Amendments: The NRC has developed Regulatory Guide 1.174 that provides guidance on an acceptable analysis approach to support changes to a plant's licensing basis using plant-specific risk information. Application specific regulatory guides have also been developed in the areas of inservice testing, inservice inspection, graded quality assurance and technical specifications. The staff is continuing its reviews of license amendments in these and other areas.

- Risk-informed technical specifications: The NRC is continuing to work with industry on several initiatives to further develop risk-informed improvements to the technical specifications. Examples of these initiatives include the replacement of fixed allowed outage times with a PRA-based configuration risk management program, and a definition of preferred end-states for technical specification actions.

- Maintenance rule: The NRC has required licensees to monitor the effectiveness of maintenance actions via the maintenance rule (50.65). A new section (a)(4) is being implemented (11-28-00) to help in controlling configuration-specific risks.

For each of the above activities, PRA results are used to determine the risk significance of structures, systems, and components (SSCs), the design and operational features critical to risk, and the events or scenarios important to risk. To make these determinations, the following are needed:

- an evaluation of the core damage frequency (CDF), large early release frequency (LERF) and potential for late containment failure of the as-operated and as-built plant

- an evaluation of the change in CDF and LERF

- an identification and understanding of the major core damage sequences and their contributors

- an identification and understanding of the core damage states and phenomena contributing to the large early release of radionuclides and late containment failure

- an understanding of the sources of uncertainty and their impact on the results.

The PRA scope needed to provide these results, and the minimal functional technical attributes necessary to ensure the risk analysis is capable of providing the above information are discussed in the following sections.

2. PRA SCOPE

The scope of a PRA plays an important role in determining the role PRA results can have in the decision-making regulatory activity. The scope of a PRA is defined by the following characteristics:

- Degree of coverage of plant operating states (POSs) that define the plant's operating mode of concern: from full-power, to low-power, to shutdown modes of operation.

- Degree of coverage of initiating events, either internal or external to the plant boundary, that cause off-normal conditions.

- Level of characterization of risk:

    — Level 1 PRA that estimates the CDF (given an event that challenges plant operation occurs).

— Level 2 PRA that estimates the containment failure and radionuclide release frequencies (given a core damage state occurs).

— Level 3 PRA that estimates the offsite consequences from a release, e.g., early and latent cancer fatalities (given a radionuclide release occurs).

For PRAs used in risk-informed activities (as outlined above), the scope and level of risk analysis are summarized in Table 1.

Table 1 List of Items Defining PRA Scope and Level of Risk Analysis

Item		**Desired Scope and Level of Risk**
POS		full and low power, hot and cold shutdown
Initiating Events	internal	• transients • LOCAs • floods • fires
	external	• seismic • high wind • others
Risk Characterization		Level 1: core damage frequency
		Level 2: large early release frequency and late containment failure
		Level 3: not required

Plant operating states (POSs) are used to subdivide the plant operating cycle into unique states such that the plant response can be assumed to be the same for all subsequent accident initiating events. Operational characteristics (such as reactor power level; in-vessel temperature, pressure, and coolant level; equipment operability; and changes in decay heat load or plant conditions that allow new success criteria) are examined to identify those important to defining plant operational states. The important characteristics are used to define the states and the fraction of time spent in each state is estimated using plant specific information. The risk perspective should be based on the total risk connected with the operation of the reactor which includes not only full power operation, but low power and shutdown conditions. Therefore, to gain the maximum benefit from a PRA, the model should address all modes of operation.

Initiating events are the events that have the ability to challenge the condition of the plant. These events include failure of equipment from either "internal plant causes" such as hardware faults, operator actions, floods or fires, or "external plant causes" such as seismic or high winds. The risk perspective should be based on the total risk connected with the operation of the reactor which includes events from both internal and external sources. Therefore, to gain the maximum benefit from a PRA, the model should address both internal and external initiating events.

The risk characterization used in risk-informed applications are CDF, LERF (as a surrogate for early fatalities), and the consideration of late containment failure; therefore, to provide the risk perspective for use in decision-making, a Level 1 PRA is required. A Level 2 PRA may be needed (i.e., estimation of the other release beyond a large early release is not needed) if the estimation of LERF for the level 1 PRA is not sufficient to provide insights on application-specific issues, or if late releases can become important for the application. A Level 3 PRA will not be required.

3. PRA ELEMENTS AND TECHNICAL ATTRIBUTES

The technical elements of a PRA that provide acceptable results are summarized below in Table 2. A PRA that is missing one or more of these elements would not be acceptable and, in fact, would not be considered a PRA.

Table 2 Technical Elements of an Acceptable PRA

Scope/Level of Analysis	Technical Element*	
Level 1	• Initiating event analysis • Success criteria analysis • Accident sequence analysis • Systems analysis	• Parameter estimation analysis • Human reliability analysis • Quantification analysis • Interpretation of results
Level 2	• Plant damage state analysis • Accident progression analysis	• Quantification analysis • Interpretation of results
*Note: documentation is not a "technical" element, however, it is an essential aspect of a PRA, and therefore, needs to be included; it is not listed as an element under Level 1 or Level 2 because it is common to each technical element		

Each of the elements in Table 2 has associated with it technical attributes needed to ensure that the results are technically correct. These technical attributes are listed in Table 3.

Table 3 Summary of Characteristics and Attributes of an Acceptable PRA

Element	Desired Characteristics and Attributes
PRA Full Power, Low Power and Shutdown	
Level 1 PRA (internal events -- transients and loss of coolant accidents (LOCAs))	
Initiating Event Analysis	• sufficiently detailed identification and characterization of initiators • grouping of individual events according to plant response and mitigating requirements
Success Criteria Analysis	• based on best-estimate engineering analyses applicable to the actual plant design and operation • codes developed, validated, and verified in sufficient detail – analyze the phenomena of interest – be applicable in the pressure, temperature, and flow range of interest – run by qualified and trained personnel
Accident Sequence Development Analysis	• defined in terms of hardware, operator action, and timing requirements • includes necessary and sufficient equipment (safety and non-safety) reasonably expected to be used to mitigate initiators • includes functional, phenomenological, and operational dependencies and interfaces

Table 3 Summary of Characteristics and Attributes of an Acceptable PRA

Element	Desired Characteristics and Attributes
Systems Analysis	models developed in sufficient detail to: • reflect the as build as operated plant • capture impact of dependencies • include failure modes that impact the function of the system, including common cause failures, human errors, etc.
Parameter Estimation Analysis	• estimation of parameters associated with basic event probability models that account for plant-specific and generic data • estimation includes a characterization of the uncertainty
Human Reliability Analysis	• identification and definition of the human failure events that would result in initiating events or would impact the mitigation of initiating events • quantification of the associated human error probabilities taking into account scenario (where applicable) and plant-specific factors and including appropriate dependencies
Quantification Analysis	• estimation of the CDF for modeled sequences that are not screened due to truncation, given as a mean value • estimation of the accident sequences CDFs for each initiating event group • truncation values set relative to the total plant CDF such that the frequency in not significantly impacted
Interpretation of Results	• identification of the key contributors to CDF: initiating events, accident sequences, equipment failures and human errors • identification of sources of uncertainty and their impact on the results • understanding of the impact of the key assumptions* on the CDF and the identification of the accident sequence and their contributors
Level 2 PRA	
Plant Damage State Analysis	• identification of the attributes of the core damage scenarios that influence severe accident progression, containment performance, and any subsequent radionuclide releases • grouping of core damage scenarios with similar attributes into plant damage states
Severe Accident Progression Analysis	• use of verified, validated codes by qualified trained users • assessment of the credible severe accident phenomena • assessment of containment system performance • establishment of the capacity of the containment to withstand severe accident environments • assessment of accident progression timing, including timing of containment failure • use of verified and validated codes run by qualified and trained personnel
Quantification Analysis	• estimation of the frequency of different containment failure modes and resulting radionuclide source terms

Table 3 Summary of Characteristics and Attributes of an Acceptable PRA

Element	Desired Characteristics and Attributes
Interpretation of Results	• identification of the contributors to containment failure and resulting source terms • identification of sources of uncertainty and their impact on the results • understanding of the impact of the key assumptions* on Level 2 results
Documentation	
Traceability and defensibility	• The documentation is sufficient to facilitate independent peer reviews • The documentation describes all of the important interim and final results, insights, and important sources of uncertainties • Walkdown process and results are fully described
*Assumptions include those decisions and judgments that were made in the course of the analysis.	

In addressing the above elements, because of the nature and impact of internal flood and fire and external hazards, their attributes need to be discussed separately. This is because flood, fire and external hazards analyses have the ability to cause initiating events but also have the capability to impact the availability of mitigating systems. Therefore, in developing the PRA model, the impact of flood, fire and external hazards needs to be considered in each of the above technical elements. Table 4 provides a summary of the desired attributes of an acceptable internal flood and fire and external hazards analyses.

Table 4 Summary of Characteristics and Attributes of an Acceptable Internal Flood and Fire Analysis and External Hazards Analysis

Areas of Analysis	Desired Characteristics and Attributes**
Internal Flood Analysis	
Flood Identification Analysis	• sufficiently detailed identification and characterization of: – flood areas and SSCs located within each area – flood sources and flood mechanisms – the type of water release and capacity – the structures functioning as drains and sumps • verification of the information through plant walkdowns
Flood Evaluation Analysis	• identification and evaluation of – flood propagation paths – flood mitigating plant design features and operator actions – the susceptibility of SSCs in each flood area to the different types of floods • elimination of flood scenarios uses well defined and justified screening criteria
Quantification Analysis	• Identification of flooding induced initiating events on the basis of a structured and systematic process • Estimation of flooding initiating event frequencies • Modification of the Level 1 models to account for flooding effects including uncertainties
Internal Fire Analysis	
Screening Analysis	• all potentially risk-significant fire areas are identified and addressed • screening criteria are defined and justified • necessary walkdowns are performed to confirm the screening decisions • screening process and results are documented • unscreened events are subjected to appropriate level of evaluations (including detailed fire PRA evaluations as described below) as needed
Fire Initiation Analysis	• all potentially significant fire scenarios in each unscreened area are addressed • fire scenario frequencies reflect plant-specific features • fire scenario physical characteristics are defined
Fire Damage Analysis	• all potentially significant components are addressed • all potentially significant damage mechanisms are addressed • analysis addresses scenario-specific factors affecting fire growth, suppression, and component damage • models and data are consistent with experience from actual fire experience as well as experiments

Table 4 Summary of Characteristics and Attributes of an Acceptable Internal Flood and Fire Analysis and External Hazards Analysis

Areas of Analysis	Desired Characteristics and Attributes**
Plant Response Analysis	• all potentially significant fire-induced initiating events are addressed • analysis reflects plant-specific safe shutdown strategy • potential circuit interactions which can interfere with safe shutdown are addressed • human reliability analysis addresses effect of fire scenario-specific conditions on operator performance • identification of sources of uncertainty and their impact on the results • understanding of the impact of the key assumptions* on the CDF
External Hazards Analysis	
Screening and Bounding Analysis	• credible external events (natural and man-made) that may affect the site are addressed • screening and bounding criteria are defined and results are documented • necessary walkdowns are performed • non-screened events are subjected to appropriate level of evaluations
Hazard Analysis	• the hazard analysis is site and plant-specific • the hazard analysis addresses uncertainties
Fragility Analysis	• fragility estimates be plant-specific for important SSCs • walkdowns are conducted to identify plant-unique conditions, failure modes, and as-built conditions.
Level 1 Model Modification	• important external event caused initiating events that can lead to core damage and large early release are included • external event related unique failures and failure modes are incorporated • equipment failures from other causes and human errors are included. When necessary, human error data is modified to reflect unique circumstances related to the external event under consideration • unique aspects of common causes, correlations, and dependencies are included • the systems model reflects as-built, as-operated plant conditions • the integration/quantification accounts for the uncertainties in each of the inputs (i.e., hazard, fragility, system modeling) and final quantitative results such as CDF and LERF • the integration/quantification accounts for all dependencies and correlations that affect the results

*Assumptions include those decisions and judgments that were made in the course of the analysis.
**Documentation also applies to flood, fire and external hazards.

The following provide additional description of the characteristics and attributes in Tables 3 and 4.

<u>Level 1 PRA</u> —

Initiating event analysis identifies and characterizes those random internal events that both challenge normal plant operation during power or shutdown conditions and require successful mitigation by plant equipment and personnel to prevent core damage from occurring. Events that have occurred at the plant and those that have a reasonable probability of occurring are identified and characterized. An understanding of the nature of the events is performed such that a grouping of the events into event classes, with the classes defined by similarity of system and plant responses (based on the success criteria), may be performed to manage the large number of potential events that can challenge the plant.

Success criteria analysis determines the minimum requirements for each function (and ultimately the systems used to perform the functions) needed to prevent core damage (or to mitigate a release) given an initiating event occurs. The requirements defining the success criteria are based on acceptable engineering analyses that represent the design and operation of the plant under consideration. The criteria needed for a function to be successful is dependent on the initiator and the conditions created by the initiator. The code(s) used to perform the analyses for developing the success criteria are validated and verified for both technical integrity and suitability to assess plant conditions for the reactor pressure, temperature and flow range of interest, and accurately analyze the phenomena of interest. Calculations are performed by personnel qualified to perform the types of analyses of interest and are well trained in the use of the code(s).

Accident sequence development analysis models, chronologically, the different possible progression of events (i.e., accident sequences) that can occur from the start of the initiating event to either successful mitigation or to core damage. The accident sequences account for those systems and operator actions that are used (and available) to mitigate the initiator based on the defined success criteria and plant operating procedures (e.g., plant emergency and abnormal operating procedures and as practiced in simulator exercises). The availability of a system includes consideration of the functional, phenomenological and operational dependencies and interfaces between and among the different systems and operator actions during the course of the accident progression.

Systems analysis identifies the different combinations of failures that can preclude the ability of the system to perform its function as defined by the success criteria. The model representing the various failure combinations includes, from an as-built and as-operated perspective, the system hardware and instrumentation (and their associated failure modes) and the human failure events that would prevent the system from performing its defined function. The basic events representing equipment and human failures are developed in sufficient detail in the model to account for dependencies between and among the different systems, and to distinguish the specific equipment or human event (and its failure mechanism) that has a major impact on the system's ability to perform its function.

Parameter estimation analysis quantifies the frequencies of the identified initiators and quantifies the equipment failure probabilities and equipment unavailabilities of the modeled systems. The estimation process includes a mechanism for addressing uncertainties, has the ability to combine different sources of data in a coherent manner, and represents the actual operating history and experience of the plant and applicable generic experience as applicable.

Human reliability analysis identifies and quantifies the human failure events that can negatively impact normal or emergency plant operations. The human failure events associated with normal plant operation include those events that leave the system (as defined by the success criteria) in an unrevealed, unavailable state. The human failure events associated with

emergency plant operation include those events that, if not performed, do not allow the needed system to function. Quantification of the probabilities of these human failure events are based on plant and accident specific conditions, where applicable, including any dependencies among actions and conditions.

Quantification analysis provides an estimation of the CDF given the design, operation and maintenance of the plant. This CDF is based on the summation of the estimated CDF from each initiator class. If truncation of accident sequences and cutsets is applied, truncation limits are set so that the overall model results are not impacted significantly and that important accident sequences are not eliminated. Therefore, the truncation limit can vary for each accident sequence. Consequently, the truncation value is selected so that the accident sequence CDF before and after truncation only differs by less than one significant figure.

Interpretation of results analysis entails examining and understanding the results of the PRA and identifying the important contributors sorted by initiating events, accident sequences, equipment failures and human errors. Methods such as importance measure calculations (e.g., Fussel-Vesely, risk achievement, risk reduction, and Birnbaum) are used to identify the contributions of various events to the model estimation of core damage frequency for both individual sequences and the model as a total. Sources of uncertainty are identified and their impact on the results analyzed. The sensitivity of the model results to model boundary conditions and other key assumptions is evaluated using sensitivity analyses to look at key assumptions both individually or in logical combinations. The combinations analyzed are chosen to fully account for interactions among the variables.

Level 2 PRA —

Plant damage state analysis groups similar core damage scenarios together to allow a practical assessment of the severe accident progression and containment response resulting from the full spectrum of core damage accidents identified in the Level 1 analysis. The plant damage state analysis defines the attributes of the core damage scenarios that represent important boundary conditions to the assessment of severe accidents progression and containment response that ultimately affect the resulting source term. The attributes address the dependencies between the containment systems modeled in the Level 2 analysis with the core damage accident sequence models to fully account for mutual dependencies. Core damage scenarios with similar attributes are grouped together to allow for efficient evaluation of the Level 2 response.

Severe accident progression analysis models the different series of events that challenge containment integrity for the core damage scenarios represented in the plant damage states. The accident progressions account for interactions among severe accident phenomena and system and human responses to identify credible containment failure modes including failure to isolate the containment. The timing of major accident events and the subsequent loadings produced on the containment are evaluated against the capacity of the containment to withstand the potential challenges. The containment performance during the severe accident is characterized by the timing (e.g., early versus late), size (e.g., catastrophic versus bypass), and location of any containment failures. The code(s) used to perform the analysis are validated and verified for both technical integrity and suitability. Calculations are performed by personnel qualified to perform the types of analyses of interest and well trained in the use of the code(s).

Source term analysis characterizes the radiological release to the environment resulting from each severe accident sequence leading to containment failure or bypass. The characterization

includes the time, elevation, and energy of the release and the amount, form, and size of the radioactive material that is released to the environment. The source term analysis is sufficient to determine whether a large early release (significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects) or large late release occurs (significant, unmitigated release from containment in a time frame that allows effective evacuation of the close-in population such that early fatalities are unlikely).

Quantification integrates the accident progression models and source term evaluation to provide estimates of the frequency of radionuclide releases that could be expected following the identified core damage accidents. This quantitative evaluation reflects the different magnitudes and timing of radionuclide releases and specifically allows for identification of the LERF and the probability of a large late release.

Interpretation of results analysis entails examining results from importance measure calculations (e.g., Fussel-Vesely, risk achievement, risk reduction, and Birnbaum) to identify the contributions of various events to the model estimation of LERF and large late release probability for both individual sequences and the model as a total. Sources of uncertainty are identified and their impact o the results analyzed. The sensitivity of the model results to model boundary conditions and other key assumptions is evaluated using sensitivity analyses to look at key assumptions both individually or in logical combinations. The combinations analyzed are chosen to fully account for interactions among the variables.

<u>Internal Floods</u> —

Flood identification analysis identifies those plant areas where flooding could pose significant risk. Flooding areas are defined on the basis of physical barriers, mitigation features, and propagation pathways. For each flooding area, flood sources due to equipment (e.g., piping, valves, pumps), internal (e;.g., tanks) and external (e.g., rivers) water sources are identified along with the affected SSCs. Flooding mechanisms are examined which include failure modes of components, human induced mechanisms, and other water releasing events. Flooding types (e.g., leak, rupture, spray) and flood sizes are determined. Plant walkdowns are performed to verify the accuracy of the information.

Flood evaluation analysis identifies the potential flooding scenarios for each flood source by identifying flood propagation paths of water from the flood source to its accumulation point (e.g., pipe and cable penetrations, doors, stairwells, failure of doors or walls). Plant design features or operator actions that have the ability to terminate the flood are identified. Credit given for flood isolation is justified. The susceptibility of each SSC in a flood area to flood-induced mechanisms is examined (e.g., submerge, spray, pipe whip, and jet impingement). Flood scenarios are developed by examining the potential for propagation and giving credit for flood mitigation. Flood scenarios can be eliminated on the basis of screening criteria. The screening criteria used are well defined and justified.

Quantification analysis provides an estimation of the CDF of the plant due to internal floods. Flooding induced initiating events that represent the design, operation and experience of the plant are identified and their frequencies quantified. The Level 1 models are modified and the internal flood accident sequences quantified: (1) modify accident sequence models to address flooding phenomena, (2) perform necessary calculations to determine success criteria for flooding mitigation, (3) perform parameter estimation analysis to include flooding as a failure mode, (4) perform human reliability analysis to account for PSFs due to flooding, and (5)

quantify internal flood accident sequence CDF. Modification of the Level 1 models are performed consistent with the characteristics for Level 1 elements for transients and LOCAs. In addition, sources of uncertainty are identified and their impact o the results analyzed. The sensitivity of the model results to model boundary conditions and other key assumptions is evaluated using sensitivity analyses to look at key assumptions both individually or in logical combinations. The combinations analyzed are chosen to fully account for interactions among the variables.

Internal Fire —

Screening analysis identifies fire areas where fires could pose a significant risk. Fire areas which are not risk significant can be "screened out" from further consideration in the PRA analysis. Both qualitative and quantitative screening criteria can be used. The former address whether an unsuppressed fire in the area poses a nuclear safety challenge; the latter are compared against a bounding assessment of the fire-induced core damage frequency for the area. The potential for fires involving multiple areas should be addressed. Assumptions used in the screening analysis should be verified through appropriate plant walkdowns. Key screening analysis assumptions and results, e.g., the area-specific conditional core damage probabilities (assuming fire-induced loss of all equipment in the area), should be documented.

Fire initiation analysis determines the frequency and physical characteristics of the detailed (within-area) fire scenarios analyzed for the unscreened fire areas. The analysis needs to identify a range of scenarios which will be used to represent all possible scenarios in the area. The possibility of seismically-induced fires should be considered. The scenario frequencies should reflect plant-specific experience, and should be quantified in a manner that is consistent with their use in the subsequent fire damage analysis (discussed below). The physical characterization of each scenario should also be in terms that will support the fire damage analysis (especially with respect to fire modeling).

Fire damage analysis determines the conditional probability that sets of potentially risk-significant components (including cables) will be damaged in a particular mode, given a specified fire scenario. The analysis needs to address components whose failure will cause an initiating event, affect the plant's ability to mitigate an initiating event, or affect potentially risk significant equipment (e.g., through suppression system actuation). Damage from heat, smoke, and exposure to suppressants should be considered. If fire models are used to predict fire-induced damage, compartment-specific features (e.g., ventilation, geometry) and target-specific features (e.g., cable location relative to the fire) should be addressed. The fire suppression analysis should account for the scenario-specific time required to detect, respond to, and extinguish the fire. The models and data used to analyze fire growth, fire suppression, and fire-induced component damage should be consistent with experience from actual nuclear power plant fire experience as well as experiments.

Plant response analysis involves the modification of appropriate plant transient and LOCA PRA models to determine the conditional core damage probability, given damage to the set(s) of components defined in the fire damage analysis. All potentially significant fire-induced initiating events, including such "special" events as loss of plant support systems, and interactions between multiple nuclear units during a fire event, should be addressed. The analysis should address the availability of non-fire affected equipment (including control) and any required manual actions. For fire scenarios involving control room abandonment, the analysis should address the circuit interactions raised in NUREG/CR-5088, including the possibility of fire-induced damage prior to transfer to the alternate shutdown panel(s). The

human reliability analysis of operator actions should address fire effects on operators (e.g., heat, smoke, loss of lighting, effect on instrumentation) and fire-specific operational issues (e.g., fire response operating procedures, training on these procedures, potential complications in coordinating activities). In addition, sources of uncertainty are identified and their impact o the results analyzed. The sensitivity of the model results to model boundary conditions and other key assumptions is evaluated using sensitivity analyses to look at key assumptions both individually or in logical combinations. The combinations analyzed are chosen to fully account for interactions among the variables.

External Hazards —

Screening and bounding analysis identifies external events other than earthquake that may challenge plant operations and require successful mitigation by plant equipment and personnel to prevent core damage from occurring. The term "screening out" is used here for the process whereby an external event is excluded from further consideration in the PRA analysis. There are two fundamental screening criteria embedded in the requirements here, as follows: An event can be screened out either (i) if it meets the certain design criteria, or (ii) if it can be shown using an analysis that the mean value of the design-basis hazard used in the plant design is less than 10^{-5}/year, and that the conditional core-damage probability is less than 10^{-1}, given the occurrence of the design-basis hazard. An external event that cannot be screened out using either of these criteria is subjected to the detailed-analysis.

Hazard Analysis characterizes non-screened external events and seismic events, generally, as frequencies of occurrence of different sizes of events (e.g., earthquakes with various peak ground accelerations, hurricanes with various maximum wind speeds) at the site. The external events are site specific and include both aleatory and epistemic uncertainties.

Fragility Analysis characterizes conditional probability of failure of important structures, components, and systems whose failure may lead to unacceptable damage to the plant (e.g., core damage) given occurrence of an external event. For important SSCs, the fragility analysis is realistic and plant-specific. The fragility analysis is based on extensive plant-walkdowns reflecting as-built, as-operated conditions.

Level 1 Model Modification assures that the system models include all important external-event caused initiating events that can lead to core damage or large early release. The system model includes external-event induced SSC failures, non-external-event induced failures (random failures), and human errors. The system analysis is well coordinated with the fragility analysis and is based on plant walkdowns. The results of the external event hazard analysis, fragility analysis, and system models are assembled to estimate frequencies of core damage and large early release. Uncertainties in each step are propagated through the process and displayed in the final results. The quantification process is capable of conducting necessary sensitivity analysis and to identify dominant sequences and contributors.

Documentation

Traceability and defensibility provides the necessary information such that the results can easily be reproduced and justified. The sources of information used in the PRA are both referenced and retrievable. The methodology used to perform each aspect of the work is described either through documenting the actual process or through reference to existing methodology documents. Assumptions[1] made in performing the analyses are identified and documented along with their justification to the extent that the context of the assumption is understood. The results (e.g., products and outcomes) from the various analyses are documented.

4. PEER REVIEW PROCESS

A peer review process can be used to identify weaknesses in the PRA and the importance of the weaknesses to the confidence in the PRA results. An acceptable peer review needs to be performed by qualified personnel, needs to be performed according to an established process that compares the PRA against desired characteristics and attributes, and needs to document the results including both strengths and weaknesses of the PRA.

The desired characteristics and attributes for an acceptable peer review of a PRA are described below and summarized in Table 5.

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[1]Assumptions include those decisions and judgments that were made in the course of the analysis.

Table 5 Summary of Desired Characteristics and Attributes of a Peer Review

Element	Desired Characteristics and Attributes
Team Qualifications	• independent with no conflicts of interest • expertise in all the technical elements of a PRA including integration • knowledge of the plant design and operation • knowledge of the peer review process
Peer Review Process	• documented process • utilize a set of desired PRA characteristics and attributes • review PRA methods • review application of methods • review key assumptions • determine if PRA represents as-built and as-operated plant • review results of each PRA technical element for reasonableness • review PRA maintenance and update process
Documentation	• describe the peer review team qualifications • describe the peer review process • document where PRA does not meet desired characteristics and attributes • assess and document significance of deficiencies

The team qualifications determines the credibility and acceptability of the peer reviewers. The peer reviewers can not give any perception of a conflict of interest, therefore, they are independent of the utility whose PRA is being reviewed and have not performed any technical work on the PRA. The members of the peer review team have technical expertise in the PRA elements they review including experience in the specific methods that are utilized to perform the PRA elements. This technical expertise includes experience in performing (not just reviewing) the work in the element assigned for review. In addition, knowledge of the specific plant design and operation is essential. Finally, each member of the peer review team is knowledgeable of the peer review process including the desired characteristics and attributes used to assess the acceptability of the PRA.

The peer review process includes a documented procedure to direct the team in evaluating the acceptability of a PRA. The review process compares the PRA against the desired PRA characteristics and attributes. In addition to reviewing the methods utilized in the PRA, the peer review also determines if the application of those methods were done correctly. The PRA models are compared against the plant design and procedures to validate that they reflect the as-built and as-operated plant. Key assumptions are reviewed to determine if they are appropriate and if they have a significant impact on the PRA results. The PRA results are checked for fidelity with the model structure and also for consistency with the results from PRAs for similar plants. Finally, the peer review process examines the procedures or guidelines in place for updating the PRA to reflect changes in plant design, operation, or experience.

Documentation provides the necessary information such that the peer review process and the findings are both traceable and defensible. A description of the qualifications of the peer review team members and the peer review process are documented. The results of the peer review for each technical element and the PRA update process are described including those areas where the PRA do not meet or exceed the desired characteristics and attributes used in the review process. This includes an assessment of the importance of any identified deficiencies on the PRA results and potential uses and how these deficiencies were addressed and resolved.

5. PRA TECHNICAL ACCEPTABILITY

The technical acceptability of the PRA can be determined by performing a peer review against a defined set of elements and characteristics specifying the scope and risk characterization. Applications can differ in the weight given to PRA results in the decision-making process. The weight given will depend on the scope of the PRA as well as its technical quality. For a given scope, the technical quality will determine the degree of confidence the decision-maker can have in the results and their role in the decision-making.

This role of the PRA is determined initially by its ability to produce the results required of the decision, and secondly by the degree of coverage of the risk contributors included in the risk metrics used in the decision. Given the role has been defined, the next step is to determine the technical acceptability of the PRA to support the results used, identify the differences, determine the importance of the differences, and determine an acceptable resolution for the important differences. The characteristics and attributes of this process are described below and summarized in Table 6.

Table 6 Summary of Characteristics and Attributes of an Acceptable Use of a PRA in Risk-Informed Applications

Element	Desired Characteristics and Attributes
Definition of the Application	Identification of: • SSCs, operator actions and plant operational characteristics affecting the decision for the application • cause-effect relationships between the change and the above SSCs, operator actions and plant operational characteristics • PRA results that can be used in the decision-making • scope of risk contributors needed to support the decision • level of analysis needed to support the decision • elements of the PRA affected by the application, • PRA characteristics and attributes needed to fully support the decision-making process
Determination of the Adequacy of PRA	• determination of whether the existing PRA scope is sufficient to address the risk contributors that impact the decision • determination of whether the existing PRA attributes, including modeled SSCs is sufficient to provide the results necessary to support the decision • identification of differences between PRA and the defined needed characteristics and attributes
Resolution of Differences	• Expand PRA to address insufficiencies and differences, or • Perform analyses with input from expert panel

Table 6 Summary of Characteristics and Attributes of an Acceptable Use of a PRA in Risk-Informed Applications

Element	Desired Characteristics and Attributes
*Note: documentation is not a "technical" element, however, it is an essential aspect of a PRA, and therefore, needs to be included; it is not listed as an element because it is common to each technical element	

The definition of the application identifies the SSCs and plant activities that are the subject of the application. When the application involves a decision on changes to the plant, the cause-effect relationship between the plant change and risk is assessed to identify how the plant change impacts the elements of the PRA model. The results from the PRA to be used in the decision-making process are identified. Therefore, to have confidence in the technical basis of the PRA for a given application, the scope and level of analysis that are needed to produce these results are identified. In addition, the technical elements for generating these results along with their associated attributes are also identified.

Determination of the adequacy of PRA identifies differences between the existing PRA and the above defined PRA scope, elements, and technical attributes and the significance of these differences. It may be determined that the scope of the existing PRA does not provide the required risk information, (for example because it only addresses internal events at full power, and the decision algorithm involves risk from all modes of operation and all initiating events); or it does not have the needed elements and technical attributes for the specific application. For the important differences, a process for resolution is determined (as discussed below).

Resolution of Differences identifies the process for resolution of identified important differences between the standard and the PRA. The resolution process either includes updating the PRA to include the important missing scope, elements and attributes, or performing compensatory measures. These measures involve accounting for deficiencies by an expert panel (see below).

6. EXPERT PANEL

As discussed above, not meeting specific attributes of an element that is important to the decision under consideration does not necessarily invalidate the use of the PRA model. The results will either have to be supplemented by engineering judgement, or compensated for by including conservatisms, or limitations in the implementation of the decision. This process can be performed with the use of an expert panel.

If an expert panel approach is elected, then there are certain characteristics and attributes that the expert panel needs to meet to be an acceptable alternative. With respect to the PRA, the primary responsibility of the expert panel is to establish the role that PRA results play in the decision, commensurate with the level of confidence in those PRA results. This requires establishing an appreciation of, and compensation for, the limitations of the model, which can be identified by comparison with the desired requirements for technical acceptability. PRA technical acceptability, as discussed above, may be achieved by performing a PRA that meets the desired characteristics and attributes defined for each technical element for the defined scope and level of analysis.

The desired characteristics and attributes to define an acceptable expert panel that are needed to support the identified applications are described below and summarized in Table 7.

Table 7 Summary of Desired Characteristics and Attributes of an Expert Panel to Use PRA Results

Element		Desired Characteristics and Attributes
Panel Member Qualifications		• diverse membership including PRA, engineering, operations, etc • wide knowledge of plant • broad understanding of how changes in requirements and issues could affect SSC response • training
Expert panel process	Decision-making Process	• decision-making process appropriate • appropriate information available • evaluation of risk significance represents appropriate consideration of issues
	Technical Information Bases	• adequate for the scope of the analysis
	Incorporation of non-PRA Modeled Items	• evaluate in a systematic manner the safety significance of items not modeled in the PRA but affected by a proposed application (e.g., SSCs, modes of operation)
	Identification of Limitations	• process applied by the licensee to overcome limitations of PRA is appropriate • decisions made that do not follow straightforwardly from the PRA need a technical basis that shows how the PRA information and the supplementary information validly combine to support the finding, and • no findings contradict the PRA in a fundamental way
Documentation		• written procedure of the expert panel process • report of the decision concluded by the panel and the basis for the conclusion

Panel member qualifications identifies the needed credentials of the panel such that decisions reached by the panel are technically defensible. The panel involves diverse membership such as PRA, engineering, operations. Plant members have a wide knowledge of plant, and a broad understanding of how changes in requirements and issues could affect SSC

response. Training is provided to the members for the activities they are required to perform. This training is of sufficient depth such that the member can make informed decisions by combining multiple, diverse knowledge sets.

The decision-making process is based on a written, systematic approach and shown to be appropriate for the decisions the panel is needed to render. The necessary technical information is made available to the panel and is examined to allow the applicable issues to be raised. The issues are disposed of using a systematic and defensible process, and documentation of findings made by the panel are traceable and reviewable. Any evaluation of the risk significance of issues appropriately consider probabilistic information, traditional engineering evaluations, sensitivity studies, operational experience, engineering judgment, and current regulatory requirements.

The technical information bases provides the necessary information for the panel to arrive at a defensible decision. This information is derived from various sources, including, for example, simplified or detailed engineering analyses, specific plant-operational expertise, and expert opinion, and shown to be adequate for the scope of the analysis. Therefore, the used technical information is sufficient to allow analysis (e.g., quantification) of both success and failure scenarios to (1) identify the roles played by the SSCs, and (2) establish the safety significance of the SSCs; and to identify causal models to be used to establish the effects of any proposed changes.

Incorporation of non-PRA modeled items involves evaluating the safety significance items not modeled in the PRA but affected by a proposed application. This systematic evaluation consists of searching for items that might contribute to initiating event occurrence, identifying mitigating system items that were not modeled in the PRA because their failure was not expected to dominate system failure in the baseline configuration, and recognizing items in systems that do not play a direct role in accident mitigation but do interface with accident mitigating systems.

Identification of limitations specifies those aspects in the PRA that decrease the level of confidence in the results, and consequently, to be addressed by the expert panel process. These deficiencies may exist because (1) an item was not modeled in the PRA, (2) an item was inappropriately modeled, or (3) lack of technology to adequately model in the PRA. The process used by the expert panel to resolve the deficiency is based the type of deficiency identified and includes (1) modeling the item in the PRA or accounting for the effects of the item by other means (e.g., using surrogate components), (2) revising the PRA model to appropriately model the item, or (3) soliciting and using expert opinion to resolve items involving a lack of technology. When a decision made by the panel that does not follow straightforwardly from the PRA, a technical basis is provided that shows how the PRA information and the supplementary information validly combine to support the finding. Further, no findings by the panel can contradict the PRA in a fundamental way.

Documentation provides the necessary information such that the expert panel process and its findings are both traceable and defensible. The documentation includes a description of the qualifications of each expert panel member, the written procedures employed by the panel, and a report of any decisions made by the panel including the basis for the conclusions.