

# **INDIVIDUAL PLANT EXAMINATIONS FOR EXTERNAL EVENTS: REVIEW PLAN AND EVALUATION CRITERIA**

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Prepared for  
U.S. Nuclear Regulatory Commission

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## **ABSTRACT**

This document presents the Review Plan and Evaluation Criteria for Individual Plant Examination for External Events (IPEEE) including: Seismic, Internal Fires, Extreme Winds, External Flooding, and Transportation and Nearby Facility Accidents.

The material presented in this document is intended to provide a review plan and evaluation criteria for the NRC staff reviewers responsible for the IPEEE. This document is to be used with the companion document NUREG/CR-5259, Individual Plant Examination for External Events: Guidance and Procedures, which provides the general framework for conducting the IPEEE.



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## **PREFACE**

This Review Plan and Evaluation Criteria for Individual Plant Examination for External Events was prepared by Lawrence Livermore National Laboratory (LLNL) and selected subcontractors. This report represents a collective effort of the IPEEE Team.

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## 1.0 INTRODUCTION

The purpose of this report is to provide a plan and evaluation criteria for reviewing an Individual Plant Examination for External Events (IPEEE) as prepared following NUREG/CR-5259. The procedures presented in this report are intended to give overall guidance for viewing vulnerability searches. NUREG/CR-5259 and NUREG/CR-5260 are intended to be used together in developing and reviewing IPEEEs.

The criteria the staff will use to review IPEEEs and acceptable analysis methods are discussed. Details of information submittal requirements and format are presented. Staff documentation requirements are also indicated.

This document has been written like the Standard Review Plan. Duplication exists in each of the external event sections since different reviewers may be involved with each external event. The reviewer should use Section 1.0, Introduction and the appropriate external event section as guidance for conducting the review.

### 1.1 Background

On August 8, 1985, the Nuclear Regulatory Commission (NRC) issued a Severe Accident Policy Statement (50 FR 32138). The Commission concluded, based on available information, that existing nuclear power plants pose no undue risk to the public health and safety and that there is no present basis for immediate action for any regulatory requirements at these plants. However, the Commission recognized, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. Therefore, each existing nuclear power plant should perform a systematic examination to identify plant-specific vulnerabilities to severe accidents and report the results to the Commission.

The policy statement does not differentiate between events initiated within the plant and events caused by external events. Current risk assessments indicate that the risk from external events could be a significant contributor, although some possibility exists that risk from external events may have been over-estimated because of the conservatisms used in the analyses. To date, both the industry and the NRC staff have concentrated on developing procedures for review of internally initiated events. Therefore, the NRC intends to proceed first with the implementation of the severe accident policy for internally initiated events (SECY-86-76). This implementation will include an integrated and systematic evaluation of each nuclear power plant for possible significant plant-specific risk contributions [Stello 1988]. This is called an Individual Plant Examination (IPE). The evaluation of external events will proceed later on a different schedule. The systematic examination of external event vulnerabilities is called an Individual Plant Examination for External Events (IPEEE).

The NRC is proceeding in two phases with the evaluation of severe accidents initiated by external hazards. The first phase consisted of a Lawrence Livermore National Laboratory (LLNL) study which assessed the margins that past design bases provide relative to external events and to identify areas where an examination for external vulnerabilities may be needed. The second phase of the external events evaluation program is the development of guidance so that plant-specific evaluations can be made.

The studies performed by LLNL [(Kimura, Budnitz 1987), (Prassinis 1988), and (Kimura, Prassinis 1989)] and the information obtained during the External Events Workshop held in Annapolis, MD on August 4-5, 1987, indicated that some selected external events need to be evaluated because they are expected to contribute significantly to, and sometimes dominate, the risk. These external events are:

- Earthquakes
- Internal Fires

- Extreme Wind and Tornadoes
- External Flooding
- Transportation and Nearby Facility Accidents

The evaluation of the potential for severe accidents initiated by seismic events [Prassinis 1988] has indicated that the seismic external hazard is important to nuclear power plant safety and should be included in the Severe Accident Implementation. This evaluation indicated that core-damage frequency from seismic events is in the range of  $10^{-7}$  to  $10^{-5}$  per reactor year. Insights from this study indicate that the dominant contributors to seismic core-damage are plant-specific and include such components as yard tanks, electrical equipment, diesel generator peripherals, structural failures, and equipment anchorages.

The main insights from the study of fire PRAs are that fire-initiated accident sequences can sometimes be important contributors to overall core-damage frequency; that the vulnerabilities to fires are very plant-specific; and that the overall numerical results for core-damage frequency have large uncertainty bounds. Despite these large numerical uncertainties, there is a general consensus that the engineering insights about vulnerabilities are valid.

In recent years, probabilistic risk assessment studies performed on a number of nuclear power plants have included the consideration of extreme wind effects. These studies indicated a mean annual core-damage frequency in the range of  $3 \times 10^{-5}$  to  $3 \times 10^{-4}$  when the associated loss of offsite power was assumed to be non-recoverable. With offsite power recoverable, the mean annual frequency of core-damage ranges from  $4 \times 10^{-6}$  to  $2 \times 10^{-5}$ . Therefore, risk studies that considered the extreme wind effects have reported rather large contribution of these events to the total core-damage frequency. A review of these risk studies has also provided an identification of dominant contributors to the core-damage frequency: these include metal sided structures, structures with thin concrete exterior walls and roof slabs, steel and concrete stacks, and yard tanks exposed to tornado missiles. Based on these findings, the effects of extreme winds cannot be generically screened out.

For external flooding, there is little of a generic nature that can be learned from the PRA literature. This is because the flooding PRAs for reactors cover a variety of sites (ocean, Great Lakes, Mississippi River, etc.). Furthermore, for several of the reactor sites studied, flooding was found to be a very minor issue because of local site conditions in relation to the adjacent water bodies. Thus, there are only a few sites where flooding must be analyzed in detail. The most important insight found as a result of a recent study [Kimura, Budnitz 1987] is that there are some sites (high and dry) where flooding can be ruled out based on hazard recurrence intervals, for other sites more analysis will be needed.

Transportation accidents are among the factors that are considered in selecting a nuclear power plant site. However, the transportation industry changes with time. Minor transportation routes develop into major transportation routes with increased traffic and greater traffic density. Vehicle speeds increase due to regulatory changes and the introduction of new technology. Vehicle weights increase with the introduction of new vehicle models. New industries near a plant site may increase hazardous material shipments (toxic/flammable/explosive material). Cargo types may be altered because of changes in the regional or national economy. Transportation accident analysis that was considered very conservative at the time the plant was originally licensed may no longer be conservative. Also, previously perceived hazards may no longer exist due to discontinued routes. Therefore, the analysis of transportation accidents at nuclear power plants must consider the movement of hazardous materials in or near a plant site whose release, detonation or burning due to an accident involving a vehicle, barge or pipeline could result in core-damage or release of radioactive material. Accidents at nearby industrial/military facilities and on-site hazardous material release are similar to transportation accidents involving hazardous material and can be analyzed in the same way.

The evaluation of "Other External Events" at nuclear power plants has included accidents at nearby industrial/military facilities, on-site release of hazardous material, severe

temperature, severe weather storms, lightning strikes, external fires, extraterrestrial activity, volcanic activity, earth movement such as land slides, and abrasive windstorms [Kimura, Prassinos 1989]. The results of this evaluation indicate that these "other" external events cause challenges to the plant that are analyzed during the process of performing an IPE or an IPEEE. For example, severe temperature transients, external fires, and abrasive windstorms cause loss-of-offsite power that is considered as an initiating event in the internal events IPE. External fires and abrasive windstorms cause challenges to control room ventilation systems which are also considered in the internal event IPE. These "Other External Events" need to be considered on a plant-specific basis and included or excluded as discussed in [Kimura, Prassinos 1989].

Willful acts such as arson, sabotage and war have been specifically excluded from the IPEEE.

## **1.2 Objectives**

The general objective of the systematic examination for external events, is to:

- determine which external hazards need to be included based on specific site and plant conditions,
- develop an appreciation of severe accident behavior initiated by external events,
- understand the severe accident sequences that could occur at the plant,
- gain a more quantitative understanding of the overall probability of core-damage and fission product release, and,
- if necessary, reduce the overall probability of core-damage and fission product release by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

It is anticipated that the achievement of these objectives will help verify that probabilities of severe core-damage and large radioactive release at U. S. nuclear power plants are consistent with the Commission's Severe Accident Policy.

## **1.3 Definition of "Vulnerabilities"**

For the purpose of this discussion, a "vulnerability" will be considered as any plant feature (equipment, structure, system, or procedure) that contributes to an accident sequence using IPE-type screening criteria.

The identification of vulnerabilities requires some type of probabilistic analysis to indicate both accident sequences that are important to plant safety and individual dominant components within those accident sequences. In this context a component can mean an item of equipment, systems, structure, or procedure. Plant vulnerabilities can then be considered as those components important to core-damage that have a relatively high failure potential with respect to the specific external event.

Plant vulnerabilities may also be identified as those important components for which very large uncertainties exist as to their ability to function given a specific external event (that is, a component for which there is not a high confidence in its capacity to perform its function and, therefore, very little basis for screening).

The individual sections will discuss vulnerability in a more specific and detailed manner as it relates to each external event.



## 1.4 Screening Criteria

Each method for the analysis of a specific external hazard employs a different screening approach using a combination of deterministic and probabilistic analyses. Screening criteria based on deterministic analysis are directly applicable to the individual hazard being evaluated and discussed in each respective section. The general approach to screening for external events is to first review the likelihood of a specific external event occurring at a particular plant site. If the occurrence frequency of this event is less than the screening criteria, the plant is not vulnerable to this external event and no further analysis is required. However, if the occurrence frequency is larger than the screening criteria, an analysis is required to understand the severe accident behavior of the plant and identify vulnerabilities.

Screening criteria based on probabilistic analyses for the reporting of potentially important accident sequences have been suggested by the NRC in the Integration Plan for Closure of Severe Accident Issues [Stello 1988]. Of these criteria adopted for IPE, only one is usable for the identification of externally initiated accident sequences:

- A functional accident sequence must be considered and evaluated further if it contributes  $1 \times 10^{-6}$  or more per reactor year to mean core-damage frequency.

This screening criterion is not an acceptance or threshold criterion for action; it is only a screening value above which an internally initiated accident sequence must be identified and documented to determine if a cost-effective change could be made to improve plant safety. For external hazard, the specific criteria is explained in each individual section and is generally on the order of  $1 \times 10^{-5}$  per year.

## 1.5 General Procedure for the Individual Plant Examination for External Events

While the individual plant examination for each external event is specific to the external event being analyzed, the general procedure shown in Figure 1-1 presents the overall approach for the performance of an examination for all external events that need to be evaluated at a specific plant site. Some external events may be screened out at a particular plant site.

The evaluation need not follow the steps given below in exactly the same order. Often, the requirements of a particular plant-specific analysis of external events will dictate a different order among the steps.

The five steps are briefly discussed below:

Step 1: Identifying the functions, systems, and components important to plant safety.

Step 2: Reviewing plant documentation in order to screen out any of the above plant functions, systems, or components that will not be affected by the particular initiating event.

Step 3: Performing a plant walkdown in order to collect as-built information for screening and the development of plant models.

Step 4: Performing plant analyses to identify core-damage accident sequences and plant vulnerabilities.

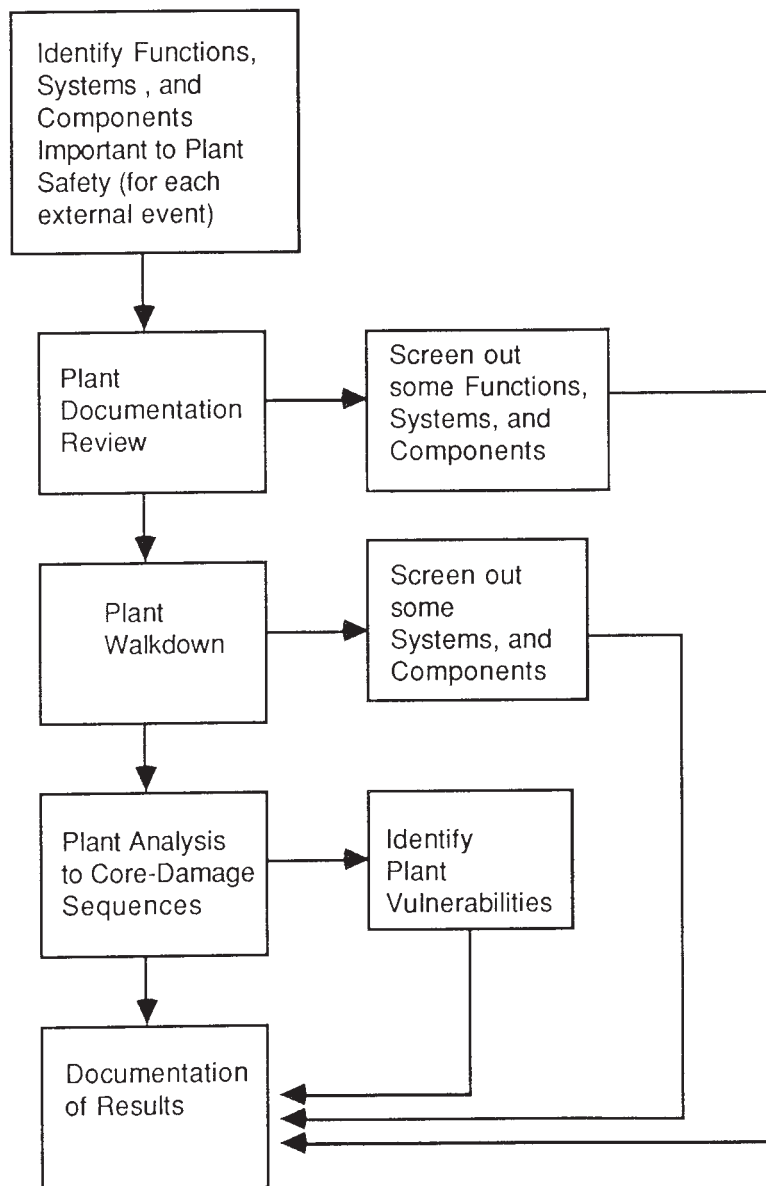
Step 5: Providing documentation on the performance of the IPEEE.

The documentation of the IPEEE should provide a level of detail sufficient to enable the NRC to:

- (1) understand and review the validity of all input data and calculational models used,

- (2) verify the sensitivity of the results to all key aspects of the analysis,
- (3) understand the reason and results of any screening, and
- (4) audit any calculations and results.

It is not necessary to submit all documentation needed for such an NRC review, but its existence must be cited and its availability must be in easily useable form.



**Figure 1-1 General IPEEE Procedure**



## **1.6 IPEEE Examination Process**

The performance of an IPEEE for each external hazard requires a team of analysts that include an expert in the external hazard being evaluated and knowledgeable systems engineers. The NRC requests the utility to use its staff to the maximum extent possible in performing the IPEEE. The maximum benefit from the examination would be realized if the licensee's staff were involved in all aspects so that the knowledge gained from the examination becomes an integral part of the plant procedures and training program. The NRC expects the utility's staff participating in the IPEEE to:

- (1) examine and understand the adequacy of the plant's emergency procedures, design, operation, maintenance, and surveillance to identify potential severe accident sequences for the plant;
- (2) understand the quantification of the accident sequence frequencies;
- (3) determine the leading contributors to core-damage and poor containment performance, and develop an understanding of their underlying causes;
- (4) identify any proposed plant improvements for the prevention and mitigation of severe accidents;
- (5) examine each of the proposed improvements, including design changes, as well as changes in maintenance, operating and emergency procedures, surveillance, staffing, and training programs; and
- (6) identify which proposed improvements will be implemented and their schedule.

Acceptable methodology for performing an IPEEE has been developed and documented in [Prassinis et al. 1988]. Other methods can be used provided adequate information is provided to allow staff review.

## **1.7 Accident Management**

The following material on accident management has been taken directly from the IPE Generic Letter 88-20, Section 9, page 7, and should be applicable to IPEEE's.

An important aspect of severe accident prevention and mitigation is the total organizational involvement. Operations personnel have key roles in the early recognition of conditions or events that might lead to core damage. The availability of procedures specifying corrective actions and the training of operators and emergency teams can have a major influence on the course of events in case of a severe accident.

Because the conclusions you will draw from the IPE for severe accident vulnerabilities (1) will depend on the credit taken for survivability of equipment in a severe accident environment, and (2) will either depend on operators taking beneficial actions during or prior to the onset of severe core damage or depend on the operators not taking specific actions that would have adverse effects, the results of your IPE will be an essential ingredient in developing a severe accident management program for your plant.

At this time you are not required to develop an accident management plan as an integrated part of your IPE. We are currently developing more specific guidance on this matter and are working closely with NUMARC to (1) define the scope and content of acceptable accident management programs, and (2) identify a plan of action that will ultimately result in incorporating any plant-specific actions deemed necessary, as a result of your IPE, into an overall severe accident management program. Nevertheless, in the course of conducting your IPE you may identify operator or other plant personnel actions that can substantially reduce the risk from severe accidents at your plant and that you believe should be immediately

implemented in the form of emergency operating procedures or similar formal guidance. We encourage each licensee to not defer implementing such actions until a more structured and comprehensive accident management program is developed on a longer schedule, but rather to implement such actions immediately within the constraints of 10 CFR 50.59.

### **1.8 IPEEE Review Process**

This document defines (1) the criteria to be used for Staff Review, which includes how to select accident sequences and the type of review analyses needed, (2) acceptable analysis methods to be used to identify vulnerabilities, (3) information that the utility needs to submit and the standardized format for the submittal, and (4) the document requirements that the NRC staff must carry out to complete the review.

### **1.9 Organization of the Report**

The remainder of this report is divided into five sections, one for the review of each external event:

Section 2 - Seismic

Section 3 - Internal Fires

Section 4 - Extreme Winds

Section 5 - External Flooding

Section 6 - Transportation and Nearby Facility Accidents

### **1.10 References**

Kimura, C.Y. and R.J. Budnitz (1987), *Evaluation of External Hazards to Nuclear Power Plants in the United States*, (NUREG/CR-5042, UCID-21223).

Kimura, C.Y. and P.G. Prassinis (1989), *Evaluation of External Hazards to Nuclear Power Plants in the United States - "Other External Hazards"*, (NUREG/CR-5042, UCID-21223, Supplement 2).

Prassinis, P.G. (1988), *Evaluation of External Hazards to Nuclear Power Plants in the United States - Seismic Hazard*, (NUREG/CR-5042, UCID-21223, Supplement 1).

Prassinis, P.G., J.B. Savy, C.Y. Kimura, G.E. Cummings, R.C. Murray, R.J. Budnitz, and M.K. Ravindra (1989), *Individual Plant Examinations for External Events: Guidance and Procedures*, (NUREG/CR-5259, UCID-21554, Draft).

Stello Jr., V. (1988), Memorandum to NRC Commissioners, *Integration Plan for Closure of Severe Accident Issues*, (SECY-88-147).

## **2.0 SEISMIC IPEEE**

This section provides procedures for reviewing a Seismic IPEEE submitted by a utility in response to the severe accident policy statement. The objective of performing a Seismic IPEEE is to systematically examine the plant and identify seismic plant-specific vulnerabilities to severe accidents.

The purpose of this review is to verify that sufficient information has been submitted and to determine whether the utility has;

- Gained an appreciation for severe accidents initiated by seismic events through significant utility involvement,
- Identified the most likely seismic initiated accident sequences and dominant components with respect to core damage and fission product release,
- Gained a quantitative understanding of the overall probability of seismic core-damage and fission product release, and
- Identified possible plant modification or upgrades that would reduce the probability of seismic core damage and fission product release.

The areas to be given attention during the review are based on the information submitted to evaluate whether the utility has addressed the items above. The reviewer should concentrate on determining whether the review was performed in a comprehensive systematic manner and that the information supplied is technically accurate. Attention should be given to the assumptions and limitations of the review, the uncertainties in the analysis, the use of up-to-date/state-of-the-art methods, and the data used in arriving at the conclusion supplied by the utility.

### **2.1 Criteria Used for NRC Staff Review**

Screening criteria are used to identify important accident sequences and plant vulnerabilities. These criteria are intended as only thresholds above which an accident sequence must be documented and considered further, to determine if a cost effective change should be made to improve plant safety. These criteria do not represent acceptance values or thresholds for action.

These criteria are used by the NRC staff to identify and differentiate those sequences that require further staff review and those that do not.

#### **2.1.1 Criteria for Seismic PRA Approach**

Screening criterion for identifying potentially important accident sequences and plant vulnerabilities were developed by the NRC for internally initiated events. However, these internal events criterion may not be directly applicable to the identification of plant vulnerabilities to seismic initiated events. Studies of past Seismic PRAs [Prassinis 1988] indicate that dominant seismic initiated core-damage accident sequences have mean frequencies in the range of  $1 \times 10^{-5}$  with corresponding median frequencies of about  $1 \times 10^{-6}$ , primarily due to uncertainties in the seismic hazard and the seismic failure of equipment and structures. In addition, external event analyses, such as that used in seismic PRAs, tends to agglomerate accident sequences resulting in correspondingly larger frequency values. Therefore, using a mean frequency criterion of  $1 \times 10^{-5}$  should identify the seismically initiated accident sequences and plant vulnerabilities.

### **2.1.2 Criteria for Seismic Margins Approach**

Plant vulnerabilities can be identified using the seismic margins approach by comparing accident sequences and components High Confidence Low Probability of Failure (HCLPF) capacities [Budnitz et al. 1985] to the review level earthquake used for the review. The criteria used to identify accident sequences for further evaluation are:

- Any accident sequence that has a HCLPF capacity less than the review level earthquake.
- Any plant component (equipment, structure, or operator) that is a constituent part of the accident sequence identified above that has a HCLPF capacity less than the review level earthquake and is a dominant contributor to the accident sequence.

### **2.2 Acceptable Analysis Methods**

A seismic IPEEE can be accomplished by performing a seismic probabilistic risk assessment (PRA) or a seismic margins review. A general discussion of using these methods for performing a seismic IPEEE is given in Individual Plant Examinations for External Events: Guidance and Procedures [Prassinis et al. 1989]. The essential ingredients in using these methods includes:

- An integrated, systematic seismic plant response analysis which accounts for component response, system response, containment response, operator action, relay chatter and breaker trip, and component unavailability due to random failures, maintenance, test and inspection.
- Detailed seismic walkdown(s) of the plant and the identification of seismic systems interactions.
- The identification of plant vulnerabilities which are dominant accident sequences of high probability or low capacity and the identification of the dominant component within these accident sequences.

As discussed in [Prassinis et al. 1989], guidance for using these methods to perform a seismic IPEEE was given as a seven step procedure. The performance of a seismic IPEEE need not follow these seven step exactly in the order and relationship shown since a particular plant review may dictate a different order or relation among the steps. However, any analysis that covers all of the steps fully is acceptable. These step are:

STEP 1 - Identify the front-line and support systems important to plant safety. Identify the relationship among and components within these systems.

STEP 2 - Determine the plant-specific seismicity characteristics.

STEP 3 - Perform an initial plant walkdown.

STEP 4 - Develop plant systems logic models.

STEP 5 - Perform a second plant walkdown.

STEP 6 - Analyze the plant systems model and determine seismic initiated accident sequences.

STEP 7 - Identify plant vulnerabilities.

Not only does this procedure provide the framework for performing the review, it also provides a format to document the results of the review. An outline of the format for the utility submittal is given in the Appendix.

### **2.2.1 Definition of "Vulnerabilities"**

A plant vulnerability is any plant configuration (equipment, structure, system or procedure) that is a dominant contributor to an accident sequences identified using the IPE-type screening criteria. Seismic plant vulnerabilities are those components important to seismic core damage that have a low capacity. Low capacity, in this sense, is the ability of the component to properly perform its safety function during and following an earthquake.

Seismic Plant vulnerabilities can also be identified as those important components for which large uncertainties exist as to their ability to function given an earthquake (that is, a component for which there is not high confidences in its capacity to perform its function and, therefore, very little basis for screening).

## **2.3 Information Submittal Requirements**

### **2.3.1 Introduction**

The NRC staff should review the information submitted upon the completion of a Seismic IPEEE. The submittal must provide the information indicated below or provide a reference to the information already submitted under other regulatory requirements.

### **2.3.2 Seismic IPEEE Review Team**

One of the objective of performing a Seismic IPEEE is for the utility to gain an understanding and appreciation of the seismic initiated accident behavior of their plant and the ability of the plant to respond to this initiating event. Therefore, the team performing the review should have significant participation by utility and plant personnel. Seismic IPEEE review team make-up should follow the guidance given in reference [(Prassinis et al. 1989), (Prassinis, Ravindra, and Savy 1986), and (EPRI NP-6041)]. The utility personnel should participate in the review including information gathering, screening, plant walkdowns, model development and analysis, and plant vulnerability identification. The reviewer should review the make up of the Seismic IPEEE team for the utility's participation in the Seismic IPEEE review.

### **2.3.3 Description of the Plant - Seismic Design Philosophy**

The overall seismic design philosophy of the plant is reviewed including the design basis and operating basis earthquakes, basis for their selection and any site specific hazard curves that have been developed for the site. In addition, the seismic categorization of the plant systems should be indicated along with the basis for the selection and installation of equipment anchorage and supports. This part of the review is not extensive but is performed to gain an understanding of the seismic basis for the plant design and construction, and the possible magnitude of the seismic hazard at the plant being reviewed.

### **2.3.4 Identification of Plant Systems and Components**

#### **2.3.4.1 Seismic PRA Approach**

For the seismic PRA approach, the front-line and support systems identified as those important to seismic plant safety are reviewed. A list of front line systems should be given along with tables indicating the interrelation between these systems and the support systems. The components within these systems should be listed including equipment, structures that are near or house the systems, and the normal and emergency procedures used to respond to an earthquake.



#### **2.3.4.2 Seismic Margins Approach**

For the seismic margins approach, the review should be consistent with the section above for those front-line and support systems and components included as part of the plant review. When a seismic margins review is performed, the components of the front-line and support systems are categorized into those that have generic HCLPF capacity greater than the review level earthquake and those that do not. During the plant walkdowns, information will be gathered to confirm this categorization and allow the determination of component HCLPF capacities. The reviewer should review the basis for determining the category of the components. This review should include component fragilities, initial component HCLPF capacity estimates and the proper use of the screening tables used to determine the component category. Guidance on use of the component screening tables is given in [Budnitz et al. 1985 and EPRI NP-6041].

### **2.3.5 Seismic Hazard and Response Analysis**

#### **2.3.5.1 Seismic PRA Approach**

The information and data used to derive the site specific hazard curve is reviewed. This review includes the data base used, whether site specific or derived from the NRC or EPRI seismic hazard characterization projects, the method for combining the various hazard estimates, and the method and analysis used to determine the building and component responses. The review of the response analysis should include any local site specific soil characteristics, soil-structure interaction (SSI) and the computer codes used in the analysis.

#### **2.3.5.2 Seismic Margins Approach**

For the seismic margin review, the review should concentrate on the basis for selecting the motion generated by a review level earthquake. This review should include enough spectra information to assure applicability of the seismic margin screening method. The review level earthquake should be specified as a uniform spectral shape anchored to a selected instrumental peak ground acceleration. The reviewer should also verify that the review level earthquake that was used was the same as the one agreed upon review of the particular plant.

### **2.3.6 Plant Walkdowns**

Seismic plant walkdowns are an essential part of the seismic IPEEE. These walkdowns are considerably more directed and thorough than plant walkdowns previously associated with seismic PRAs, and should conform to those developed as part of the NRC Seismic Design Margins Program and the EPRI Seismic Margins Program. The performance of the seismic plant walkdowns and the results are applicable to both approaches to the Seismic IPEEE.

The information gathered during the plant walkdown(s) should be reviewed with the understanding of determining component capacity and identifying any plant seismic systems interaction. This review should also take note that an A-46 review could have been performed in conjunction with this plant walkdown or information from a previously performed A-46 review was used as part of the plant walkdown.

During the plant walkdowns, the identified components are inspected to look for outlier, lack of similarity, anchorages and supports that are different from drawings or the prescribed criteria previously reviewed. In addition, the components are inspected with respect to the applicability of the seismic screening table for eliminating components from further consideration. The result of the plant walkdown(s) is the collection of information that is used to identify plant seismic systems interactions, develop and verify plant logic models, and to determine component capacities.

From this information, detailed capacity analyses are performed to determine the component fragilities and/or HCLPF capacities. The methods and data used to determine these component

capacities should be reviewed. For a seismic margins review, those components that have generic HCLPF capacities greater than the review level earthquake and found to be consistent with their generic capacity are screened out from further analysis. The basis for this screening of components should be reviewed including any analytical or empirical information provided.

### **2.3.7 Systems Analysis**

The review of the systems analysis should concentrate on the assumptions and models used to develop the accident sequences and the resultant combination of seismic and non-seismic failures that produce core damage.

#### **2.3.7.1 Seismic PRA Approach**

The reviewer should review the seismic-induced initiating events and the resultant event trees for the appropriate sequences of systems failures. The fault trees should be reviewed for the correct usage of seismic failures and the combination of seismic and non-seismic failures in their development. Of particular interest is the inclusion of seismic relay and breaker trip behavior, and the modeling of operator actions.

#### **2.3.7.2 Seismic Margins Approach**

For a seismic margins review, the systems analysis concentrates on a few initiating events and safety systems, and relies on screening to reduce the size and complexity of the systems models. The review of these model should be performed as above but added attention should be given to development and pruning the fault trees so that paths from screened-out components are left intact between each layer in the tree. This is because the remaining lower level components represent the possible failures for the systems under consideration and propagation within the tree must be preserved. The basic events in the fault trees should include component seismic failures, random failures, unavailabilities, and human errors.

### **2.3.8 Description of Seismic Initiated Accident Sequences**

The methods and computer codes used to analyze the plant systems models should be noted along with any methods or analysis used to truncate the Boolean expression and plant accident sequences. The important accident sequences should be described in qualitative terms as to the progression of failures that lead to core damage along with a discussion of the quantitative aspects that underlie the finding that a particular sequence is important. The review will concentrate on the basis upon which the analysis supports the overall insights, taking into account the uncertainties and limitations of the various inputs.

#### **2.3.8.1 Seismic PRA Approach**

For a seismic PRA, the analysis should describe the important seismic induced accident sequences found to contribute significantly to overall core damage frequency. These descriptions should be reviewed including how sensitive the sequence insights are to the assumptions, limitations and uncertainties that were used in the plant models.

#### **2.3.8.2 Seismic Margins Approach**

For the seismic margins method, accident sequences and cut sets are analyzed by comparing their HCLPF capacity to the review level earthquake. Those cut sets with HCLPF capacities that are greater than the review level earthquake are screened out from further review. The remaining cut sets are combined into the accident sequence. These accident sequence should be reviewed such that the sensitivity of sequence insights to the assumptions, limitations, uncertainties and the combination of seismic and non-seismic element are considered.



### **2.3.9 Identification of Seismic Plant Vulnerabilities**

The seismic IPEEE must include the identification of the important plant vulnerabilities, defined as those dominant accident sequences and dominant component leading to core damage as determined using the appropriate screening criteria. The identification of these vulnerabilities is reviewed with respect to how sensitive the results and insights are to the assumptions and uncertainties in the analysis.

Each identified plant seismic vulnerability should be described in both qualitative and quantitative terms including the factors that were used to determine that the particular vulnerability was important.

### **2.4 Staff Review Requirements**

Selection and emphasis of the various aspects covered by this review will be made by the reviewer. The judgment on the areas to be given attention during the review is to be based on an inspection of the information presented, the similarity of the material to recent reviews of other plants, and whether items of special safety significance are involved.

The reviewer should determine the basis for the information provided, and the underlying limitations and assumptions used to arrive at the results.

The reviewer should examine:

- the basis for conducting the Seismic IPEEE. This review should include the seismicity of the plant site and the plant's seismic design basis, operation, maintenance, and emergency procedures. The reviewer should also indicate whether other seismic plant reviews have been performed and whether their results are used as part of this review.
- the methods used for analyzing and the calculations of the seismic hazard and plant seismic response or the information provided for the selection of and use of the review earthquake level.
- the results of the plant walkdown(s) and indicate any significant findings as to seismic systems interactions, outliers, and inadequate anchorages or supports.
- the plant models, the methods and techniques used for their analysis and the resulting accident sequences, and the data used for their quantification.
- the methods used for the analysis of the calculations of component capacities, including fragility analysis and HCLPF calculations.
- the important seismic initiated accident sequences, and the dominant contributors to seismic core damage and containment response, and their underlying causes.
- the identified plant vulnerabilities and any proposed plant improvements including design changes, maintenance, operating or emergency procedures, surveillance, staffing, and training that may prevent or mitigate severe accidents.

### **2.5 Staff Documentation Requirements**

The reviewer verifies that sufficient information has been provided to satisfy their requirement of performing a seismic IPEEE, and concludes that this review is sufficiently complete and adequate to support the following statements to be included in the staff's evaluation report:

"The analysis of the seismic-induced accident sequences, including the identification of important accident sequences to seismic core damage and of plant specific vulnerabilities, are

acceptable, since this analysis provides a reasonable basis for decision making concerning the importance of the identified sequences and vulnerabilities."

"This analysis and review provides an acceptable basis for satisfying, in part, the requirements of the Commission's Severe Accident Policy Statement insofar as it concerns the analysis and identification of seismic initiated accident sequences and vulnerabilities."

## **2.6     References**

Budnitz, R.J., P.J. Amico, C.A. Cornell, W.J. Hall, R.P. Kennedy, J.W. Reed, and M. Shinozuka (1985), *An Approach to the Quantification of Seismic Margins in Nuclear Power Plants*, (NUREG/CR-4334, UCID-20444).

EPRI NP-6041 (1988), *A Methodology for Assessment of Nuclear Power Plant Seismic Margin*.

Prassinis, P.G. (1988), *Evaluation of External Hazards to Nuclear Power Plants in the United States - Seismic Hazard*, (NUREG/CR-5042, UCID-21223, Supplement 1).

Prassinis, P.G., M.K. Ravindra, and J.B. Savy (1986), *Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants*, (NUREG/CR-4482, UCID-20579).

Prassinis, P.G., J.B. Savy, C.Y. Kimura, G.E. Cummings, R.C. Murray, R.J. Budnitz, and M.K. Ravindra (1989), *Individual Plant Examinations for External Events: Guidance and Procedures*, (NUREG/CR-5259, UCID-21554, Draft ).

### **3.0 INTERNAL FIRES IPEEE**

#### **3.1 Criteria Used for NRC Staff Review**

This section provides procedures for reviewing an Internal Fires IPEEE submitted by a utility in response to the Severe Accident Policy Statement. The objective of performing an Internal Fires IPEEE is to systematically examine the plant and to identify fire related plant-specific vulnerabilities to severe accidents.

The purpose of this review is to verify that sufficient information has been submitted and to determine whether the utility has:

- Gained an appreciation for severe accidents initiated by internal fires through significant utility involvement,
- Identified the most likely internal fire initiated accident sequences and dominant components with respect to core damage and fission product release,
- Gained a quantitative understanding of the overall annual frequency of internal fire core damage and fission product release, and
- Identified possible plant modification or upgrades that would reduce the annual frequency of internal fire induced core damage and fission product release.

##### **3.1.1 Screening Criterion for Selecting Potentially Important Sequences**

In the "IPE Generic Letter" covering internally-initiated accident vulnerabilities, the NRC has developed screening criteria for the selection of potentially important sequences. Of these IPE criteria, one will be mentioned here as particularly applicable to fire-initiated sequences. The criterion is that a functional accident sequence must be considered and evaluated further if it satisfies the following:

- Any functional sequence that contributes  $1 \times 10^{-6}$  or more per reactor year to core-damage frequency.

This is not an acceptance criterion, nor a threshold for action, but only a threshold above which a sequence must be documented and considered, to determine if a cost-effective change should be made to improve safety. The converse is that a functional sequence can be screened out and not considered further if it falls below this criterion.

This criterion is used by the NRC staff to differentiate those sequences that require further staff review from those that do not.

##### **3.1.2 Criteria Used in Reviewing the Analysis**

In reviewing the analysis submitted, the staff examines whether the methods used are up-to-date, state-of-the-art PRA methods, applied properly and using appropriate data. The methods are discussed in the companion document, NUREG/CR-5259. Each sub-element of the analysis is examined from this perspective. The sub-elements of the methodology are discussed further below in Section 3.3.

#### **3.2 Acceptable Analysis Methods**

##### **3.2.1 Acceptable Methodology**

The only methodology currently available that has been reviewed by the staff and found adequate for identifying fire-initiated vulnerabilities in the context of the IPE is a methodology based on probabilistic risk assessment (PRA). At present, no other methodology

exists that can identify vulnerabilities from fires in a way that not only accounts for how fires start, spread, are suppressed, and damage equipment, but also analyzes how the various damaged components and systems combine together to cause a core-damage accident, and with what annual frequency. Therefore, for fire-initiated accidents the PRA methodology is the only currently available approach.

### **3.2.2 Variants on Fire-PRA Methodology**

Since the first LWR internal-fire PRAs were completed in 1980-81, the methodology has matured significantly. This evolution is discussed briefly in NUREG/CR-5042 [Kimura, Budnitz 1987], and also reviewed in Sandia's Fire Risk Scoping Study, NUREG/CR-5088, [Lambright et al. 1988]. There are also a large number of research reports, too numerous to cite here, covering various aspects of the methodology or the supporting data base.

Among the key recent advances are the combining of fire-induced failures with non-fire-induced failures; the more efficient screening of fire zones and areas; the inclusion of a variety of different fire types; an improved data base for the initiating frequency of various fire types; and a better understanding of manual suppression.

It is unlikely that any new fire PRA would be carried out using older methods and ignoring these recent advances. No specific methodology or any specific practitioners are endorsed here; rather, it is felt that almost any "modern" fire PRA methodology is adequate to meet the NRC's needs for an effective "vulnerability search", if applied properly. In this sense, there does not seem to be any one fire-PRA methodology that is to be "preferred" or "recommended" above the others now in use by the (relatively small) community of fire-PRA practitioners. Any can be adequate if applied properly.

This conclusion does not mean to imply that today's fire PRA methodologies have no weaknesses or limitations. Several have been identified, most recently in Sandia's Fire Risk Scoping Study, [Lambright et al. 1988]. However, it is felt that despite these limitations, today's PRA methods can accomplish a useful vulnerability search, with the application of appropriate engineering judgment. Furthermore, it is highly recommended that analysts attempt to incorporate as many as feasible of the insights from NUREG/CR-5088 into their PRA analyses: this will strengthen the methodology and accomplish a better vulnerability search.

In particular, a few topics not always considered in fire PRAs should be studied. Examples include barrier vulnerabilities, interactions between the control room and remote shutdown locations, and secondary failures from fire-extinguishing activities and equipment. Also, NUREG/CR-5088 points out that there are weaknesses in the available codes that deal with fire growth and spread and the way it competes with suppression. Therefore, the numerical values of core-damage frequencies calculated with these codes will have important quantitative uncertainties which need to be recognized.

There is clearly a need for realistic engineering calculations as a part of any probabilistic analysis. These analyses are always essential to develop the fire scenarios, such as understanding the competition between fire spread and suppression. Also, such analyses are vital to develop an understanding of sensitivities or uncertainties.

### **3.2.3 Scope of an Acceptable Analysis**

The scope of the vulnerability-search analysis must satisfy the following, and the review ascertains that this scope has been accomplished:

- The analysis must include all accident sequences initiated by an internal plant fire;

- This includes fires anywhere within the plant site, but explicitly excludes offsite fires, such as explosions from nearby transportation activities (trucks, barges, gas pipelines, etc., which are considered in Section 6);
- Earthquake-initiated internal fires are explicitly to be included, requiring that the analysis be accomplished in coordination with the seismic analysis undertaken at the same time;
- Fire-initiated accident sequences that also involve important non-fire-induced failures (equipment failures, human errors, equipment out-of-service, etc.) must be included as appropriate;
- An "accident sequence" is considered to be any functional combination of events leading to unacceptable consequences (here, this means a "core damage accident") with an unacceptable annual frequency (here, this means that the mean calculated value of "events per year" is above the screening criterion discussed above);
- Arson and other sabotage-type acts are explicitly excluded from the scope.

#### **3.2.4 Definition of "Vulnerabilities"**

A key part of the direction to an analyst performing a vulnerability search is how "vulnerabilities" are defined. For the purposes of this discussion a fire "vulnerability" will be considered as any plant configuration (hardware or operational) that contributes to a fire-initiated accident sequence of concern using the IPE-type screening criterion discussed above (Section 3.1).

It is not possible to develop a complete list of fire-initiated vulnerability types, but a few examples can help to clarify the thrust of the definition. Examples include such aspects as the inability of the on-site fire brigade to accomplish effective manual suppression in an important fire zone; or the inadequacy of automatic suppression in such a zone; or the presence of too much transient fuel, too frequently, at a given location; or the confluence in one location of cabling or support systems for several key safety systems that could all be compromised by a single fire with an unacceptably high probability of happening; or the absence of procedures for securing the plant in a safe condition after a given fire. Other examples are earthquake-initiated fire sequences; secondary failures from fire-extinguishing equipment and activities; and core-damage sequences involving a combination of fire-induced and non-fire-induced failures.

Generally, a "vulnerability" can be either within or outside the "design basis" and NRC's current "regulatory envelope". For the purposes of the vulnerability search being discussed here, no distinction will be made on this basis. Specifically, although one should assume that plants meet NRC's current regulations (if not, that would be an enforcement issue, rather than an issue of concern here), by itself this does not assure that plants will be free of vulnerabilities as they have been defined here.

### **3.3 Information Submittal Requirements**

#### **3.3.1 Introduction**

It is not feasible to provide highly specific guidance on the format and content of the information that must be submitted. In this subsection, general guidance will be provided.

#### **3.3.2 Description of the Plant—Fire Safety Philosophy and General Fire Protection Features**

The overall fire-protection philosophy and fire-protection systems of the plant must be described. The submittal must provide this information, or a reference must be provided to a



description already available (such as in the SAR). The documentation must cover how the combination of several features (design and layout; emergency alarms; fire brigade size and training, etc.) allows for an effective fire response capability.

### **3.3.3 Identification of Critical Fire Areas**

The method used for identifying critical fire areas must be described in detail. Often, the identification begins with the fire areas and zones from the SAR, but these are usually modified somewhat for the purposes of the PRA analysis. The criteria used for deciding on the size of each key fire zone and area, and the way these criteria are applied in practice, must be covered. One key issue is assuring that each fire zone/area is homogeneous enough for use in the subsequent analysis.

### **3.3.4 Criteria for Fire Size, Duration, and Damage Potential**

In a PRA analysis, not all fire sizes and durations can be analyzed individually: certain specific (stylized) fire sizes and durations are usually chosen for analysis, and then used as surrogates for individual fires that might occur in various locations. The criteria for selecting these fire sizes and durations must be documented.

The submittal must also discuss the criteria used in the analysis for deciding whether equipment is damaged to the point of failure by a given fire. This includes the criteria used for defining "failure", which will differ for each class of equipment. Since damage potential for a given fire type is a key aspect of the model used in the analysis, the sensitivity of the results to the criteria and assumptions used must be described.

### **3.3.5 Fire Initiation Frequency**

The data base used for fire initiation frequencies must be documented. This includes the underlying data base, the method for partitioning the data base among different types of fire compartments (areas and zones), and the sensitivity of the results to the partitioning method.

The choice of different classes of compartments is usually dictated by the types and extent of available data. Furthermore, in a given plant, a specific compartment may or may not be similar to "typical" compartments elsewhere—for example, in some plants the diesel-generators are in small self-contained rooms or buildings, while in others they are located in a much larger space within a large building.

The partitioning method is important. The "ratio" method, for example, assumes that the probability of a fire occurrence is proportional only to the ratio of the area that the fire zone occupies within a large area, so that if a critical area is, say, 10% of a larger compartment for which good data exist, the fire initiation frequency in the smaller critical area is assumed to be just 10% of the frequency for the larger compartment. More detailed partitioning methods attempt to account for factors that might change the ratio-method probability, such as the number and types of electrical components, how often the area is occupied, what types of sprinklers or other controls exist, and so on. Still another partitioning approach modifies the ratio-method frequency by corrections for transient fuel loading, use patterns, or other factors that vary over time. The partitioning method, and the sensitivity of the results to the method, must be documented.

### **3.3.6 Fire Growth and Spread, and Likelihood of Disabling Equipment**

The documentation must cover the treatment of fire growth and spread, which is usually analyzed with a computer-based model that contains assumptions, algorithms, and data. The discussion must include how detection is modeled in different compartments, how suppression is modeled, and how the competition between detection and suppression is modeled. The details within the model must be covered, including the spread of the fire itself (spatially and

temporarily) and of hot gases and smoke, assumptions and analysis concerning inter-compartment spreading, and how various types of suppression are assumed to stop the fire.

Usually, a few stylized fires are analyzed in detail. How these are chosen, and how representative they are of the broader types of fires that exist in the data base, were described earlier (see Section 3.3.4). Here, the documentation must discuss the specific applications of the fire-analysis models to specific compartments.

Among the parameters involved in modeling that must be discussed here are how cable-insulation fires are modeled (heat of combustion, surface-controlled burning rate, radiative fraction, critical temperatures for piloted and spontaneous ignition, reflectivity, etc.); how oil fires are modeled (size, surface thickness vs. volume, surface-controlled burning rate, radiative fraction, etc.); and how hot-gas spreading away from the fire is treated.

Assumptions and underlying data about the effectiveness of barriers (fire walls, fire doors, dampers, penetration seals, etc.) must also be documented, including the use of the existing data base or modifications to it.

The definitions of "failure", documented above (Section 3.3.4), are generally applied in this part of the analysis. The discussion here covers how the model determines when a criterion has been met so that damage and failure are assumed to occur.

### **3.3.7 Systems Analysis**

The discussion here must concentrate on the analysis of how fire-induced failures combine together, or combine with other (non-fire-induced) failures to produce a plant-level failure (a core-damage accident or worse).

Often, an assumption is made that, given an on-site fire, the operators will surely trip the plant manually if no automatic plant trip occurs—with this assumption, the sum of all sequence initiators, given a fire occurrence, is always 100%, with one initiator being called "normal manual trip". How this is applied by the analyst must be documented.

Sometimes, certain entire classes of initiating events, such as fire-initiated large LOCAs, are eliminated by the analyst based on an assumption that they are very unlikely probabilistically. The basis for such assumptions, if present, must be documented.

It is common for fire-PRA systems analysts to begin with the same event trees and fault trees that are used in the internal-initiators analysis. Sometimes, these are modified to include fire-specific events, but sometimes not. This aspect of the analysis must be discussed. In particular, to the extent that an internal-initiators systems analysis is the primary basis for the fire-initiators systems analysis, the validity of this approach must be examined with care. On the one hand, such an approach should assure a consistent level of analytical detail; on the other hand, fire-specific sequences of events may be modeled incompletely using this approach.

How non-fire-induced failures are combined with the fire-induced failures in the systems analysis must also be documented.

The documentation should concentrate on the way that the analysis truncates cut sets of sufficiently low probability. (Without such truncation a fire-PRA analysis can sometimes become unmanageably bulky without much increase in useful detail.) Truncation (or screening) can assume, for example, that cut sets involving two compartments can be eliminated on the basis of low probability if they are not adjacent or not connected; and that cut sets requiring three or more fire compartments can also be eliminated.

Often, the systems analysis includes a study of recovery, which is meant to cover the non-fire-fighting actions taken by operating personnel to prevent the fire-initiated sequences from



progressing to a core-damage stage. Assumptions as to whether the existence of the fire impedes recovery actions, and if so how, must be discussed.

### **3.3.8 Descriptions of Fire-Initiated Accident Sequences**

An acceptable fire analysis must include descriptions of the key fire-initiated accident sequences found to contribute significantly to overall plant core-damage frequency. These descriptions, including how sensitive the sequence insights are to various assumptions and uncertainties in input data and models, must be included.

Each key accident sequence should be described in qualitative terms, along with a discussion in quantitative terms of what factors underlie the finding that the particular sequence is "important". The discussion should concentrate on the extent to which the analysis can support its overall insights, taking account of uncertainties and sensitivities to various inputs.

### **3.3.9 Identification of Fire-Induced Vulnerabilities**

An acceptable fire analysis must include the identification of the key fire-initiated vulnerabilities, as defined in the context of the Individual Plant Evaluation program (IPE). The identification of these vulnerabilities is important, including how sensitive the insights are to various assumptions and uncertainties.

Each identified vulnerability should be described in qualitative terms, along with a discussion in quantitative terms of what factors underlie the finding that the particular vulnerability is "important".

## **3.4 Staff Review Guidance**

In this subsection, general guidance will be provided concerning how the staff reviews the types of information that must be submitted: the specific submittal requirements were covered in Subsection 3.3 above. To avoid repetition of the technical detail, the discussion here will be keyed to the discussion in Subsection 3.3.

The staff review will cover, among other topics:

- (See Subsection 3.3.2): The overall fire-protection philosophy and fire-protection systems of the plant. This part of the review is not extensive, and does not duplicate the ordinary staff review of these safety aspects under other regulations, but it must be adequate to support the review of the detailed analysis to follow.
- (See Subsection 3.3.3): The method used for identifying critical fire areas. This includes the criteria used for deciding on the size of each key fire zone and area, and the way these criteria are applied in practice.
- (See Subsection 3.3.4): The criteria for selecting fire sizes, durations, and damage potential. This includes the criteria used for defining "failure", and the sensitivity of the results to the criteria and assumptions used.
- (See Subsection 3.3.5): The data base used for fire initiation frequencies. This includes not only the underlying data base, but the method for partitioning the data base among different types of fire components, and the sensitivity of the results to this method.
- (See Subsection 3.3.6): The treatment of fire growth and spread, and the likelihood of disabling equipment. This includes the specifics of the model used, as discussed in detail in Subsection 3.3.6.
- (See Subsection 3.3.7): The systems analysis. The review will concentrate on how fire-induced failures are combined together, and combined with non-fire-induced failures to

produce a plant-level failure (a core-damage accident or worse). The review will cover, among other topics, assumptions on operator response and recovery, the elimination of whole classes of sequences on a probabilistic basis, truncations, and how the event-trees and fault-trees from the internal-initiators analysis are adapted for the fire analysis.

- (See Subsection 3.3.8): Descriptions of the key fire-initiated accident sequences, including how sensitive the insights are to various uncertainties and assumptions in the input data and models.
- (See Subsection 3.3.9): Identification of key fire-initiated vulnerabilities, including how sensitive the insights are to various uncertainties and assumptions.

### **3.5     Staff Documentation Requirements**

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that this evaluation is sufficiently complete and adequate to support the following type of statement to be included in the staff's evaluation report:

"The analysis of fire-initiated accident sequences, including the identification of key fire-initiated accident sequences and of plant-specific vulnerabilities, is acceptable, since this analysis provides a reasonable basis for decision-making concerning the importance of the identified sequences and vulnerabilities."

"This analysis provides an acceptable basis for satisfying, in part, the requirements of the Commission's Severe Accident Policy Statement insofar as it concerns the analysis and identification of fire-initiated accident sequences and vulnerabilities."

### **3.6     References**

- Kimura, C.Y. and R.J. Budnitz (1987), *Evaluation of External Hazards to Nuclear Power Plants in the United States*, Lawrence Livermore National Laboratory (Report NUREG/CR-5042).
- Lambright, J.A., S.P. Nowlen, V.F. Nicolette, and M.P. Bohn (1988), *Fire Risk Scoping Study*, Sandia National Laboratories, (Report NUREG/CR-5088).
- Prassinis, P.G., J.B. Savy, C.Y. Kimura, G.E. Cummings, R.C. Murray, R.J. Budnitz, and M.K. Ravindra (1989), *Individual Plant Examination for External Events: Guidance and Procedures*, (NUREG/CR-5259, Draft ).

## **4.0 EXTREME WINDS IPEEE**

This section provides procedures for reviewing an Extreme Winds IPEEE submitted by a utility in response to the Severe Accident Policy Statement. The objective of performing an Extreme Winds IPEEE is to systematically examine the plant and to identify extreme wind related plant-specific vulnerabilities to severe accidents.

The purpose of this review is to verify that sufficient information has been submitted and to determine whether the utility has:

- Gained an appreciation for severe accidents initiated by extreme wind events through significant utility involvement,
- Identified the most likely extreme wind initiated accident sequences and dominant components with respect to core damage and fission product release,
- Gained a quantitative understanding of the overall annual frequency of extreme wind core damage and fission product release, and
- Identified possible plant modification or upgrades that would reduce the annual frequency of extreme wind induced core damage and fission product release.

### **4.1 Criteria Used for NRC Staff Review**

#### **4.1.1 Screening Criteria for Selecting Potentially Important Sequences Induced by Extreme Winds**

In the "IPE Generic Letter" covering internally-initiated accident vulnerabilities, the NRC has developed screening criteria for the selection of potentially important sequences. Of these IPE criteria, only one is usable for the identification of extreme wind induced accident sequences:

- A functional accident sequence must be considered and evaluated further if it contributes  $1 \times 10^{-6}$  or more per reactor year to core-damage frequency.

This screening criterion is not an acceptable criterion or threshold criterion for action; it is only a screening value above which an accident sequence must be identified and documented to determine if cost-effective changes could be made to improve plant safety.

This criterion is used by the NRC staff to differentiate those sequences that require further staff review from those that do not.

#### **4.1.2 Criteria Used in Reviewing the IPEEE Analysis of Extreme Wind Effects**

In reviewing the analysis submitted, the staff examines whether the methods used are up-to-date, state-of-the-art methods used properly and using appropriate data [Prassinis et al. 1989]. The sub-elements of the methodology are discussed further below in Section 4.3.

## **4.2 Acceptable Analysis Methods**

### **4.2.1 Acceptable Methodology**

The IPEEE Guidance and Procedures Document [Prassinis et al. 1989] has described procedures for treating extreme wind effects in the IPEEE of nuclear power plants. The primary focus is on deciding whether the extreme wind effects at a particular plant site require an in-depth plant examination or could be screened out as not contributing significantly to the core-damage frequency. Even for those plants where the extreme wind effects are determined to require further evaluation, the methods included in the following aim to show how the IPEEE goal may be met by limited analyses and review of the plant systems and structures.

The analysis methods described in the IPEEE Guidance and Procedures Document [Prassinios et al. 1989] consist of screening procedures at different stages:

- Procedure based on conformance to regulatory requirements: it consists of reviewing the plant design information for conformance to current regulatory requirements (i.e., RG 1.76, 1.117 and Standard Review Plan Sections 3.3.1, 3.3.2, 3.5.1.4 and 3.5.3) and performing a confirmatory walkdown.
- Procedure based on plant design review: more realistic criteria than the current NRC criteria are used in evaluating the plant design against extreme wind effects.
- Bounding analysis for core damage frequency: in this analysis, it is intended to estimate conservatively the contribution of extreme wind effects to the core damage frequency using simplified approaches. If the annual core damage frequency obtained in the bounding analysis exceeds  $1 \times 10^{-6}$ , a detailed risk assessment of the extreme wind effects needs to be performed.

The utility may choose to perform the IPEEE using these procedures or other alternative methods as long as the goal of the IPEEE is met (i.e., identifying the potential vulnerabilities in the plant). The staff's review focuses on confirming that this is accomplished in the IPEEE analysis.

#### **4.2.2 Scope of an Acceptable Analysis**

The scope of the vulnerability-search for extreme wind effects must satisfy the following and the staff's review ascertains that this scope has been accomplished:

- Depending on the geographic region, the analysis must include the effects of tornadoes, hurricanes, and wind storms. The effects include wind loading taking into account pressure differential between the inside and outside of a tornado, and impact of missiles generated in a tornado or hurricane.
- Wind-initiated accident sequences that also involve important non-wind-induced failures (equipment failures, human errors, equipment out-of-service, etc.) must be included as appropriate.
- An "accident sequence" is considered to be any functional combination of events leading to unacceptable consequences (here, this means a "core damage accident") with an unacceptable annual frequency (here, this means that the mean calculated value of "events per year" is above the screening criterion discussed above).

#### **4.2.3 Definition of "Vulnerabilities"**

A vulnerability is considered as any plant configuration (equipment, structure, system or procedure) that is significant to an accident sequence using the IPE screening criterion. In its strict sense, identification of vulnerabilities requires some type of probabilistic analysis to indicate both accident sequences that are important to plant safety and individual dominant components within those accident sequences. However, vulnerabilities could still be identified at different stages of the screening analysis described above. Plant vulnerabilities can then be considered as those components important to core damage that have a relatively high failure potential with respect to the extreme wind effects. Based on the experience and judgement of the analyst, vulnerabilities may be identified in the plant review and walkdown, and bounding analysis. The guidance on walkdown is given with the primary aim of vulnerability search.



### **4.3 Information Submittal Requirements**

The utility should submit a report for the IPEEE for extreme wind effects. It should document the following, depending on the stage at which the extreme wind effects are screened out.

Site Information and Design Criteria. A description of the location of the plant (i.e., inland or coastal), terrain (i.e., mountainous, valley, unusual wind conditions), and major structures, including yard equipment, should be provided. The design criteria used for plant design (such as the design basis wind speed, parameters of the design basis tornado along with missile spectrum, and the allowable stresses and load combinations) should be discussed. If this material is available in the plant Safety Analysis Report, it should be referenced.

Plant Layout and Structural Dimensions. A description of the plant layout showing major structures and design strategy (e.g. designed to withstand tornado and tornado missile effects; designed to vent in a tornado but not collapse; or designed to withstand only design basis wind) used for each structure should be included. It should also identify the material and thickness for each external barrier (i.e., roof and wall) of safety related structures. If there are safety-related items that are not enclosed by structures, the report should discuss how this equipment is protected against tornado missiles.

Conformance to Regulatory Requirements. The report should document how the plant design meets the regulatory requirements specified in the Regulatory Guide 1.76 and 1.117 and the applicable sections of the Standard Review Plan. It should describe the tornado missile spectrum used in the plant design and discuss how it compares with those given in the Standard Review Plan.

Plant Walkdown. The procedures used for plant walkdown and the walkdown findings should be recorded. The discussion should, at a minimum, focus on the walkdown inspection list given in Table 4-1 of the IPEEE Guidance and Procedures Document [Prassinis et al. 1989].

Potential Vulnerabilities. The report should discuss the potential vulnerabilities (e.g., equipment exposed to tornado missile impact and structures that may collapse on critical structures and equipment) identified in the review of the design and plant walkdown. It should also outline the proposed actions to overcome the vulnerabilities.

Methods of Satisfying the IPEEE Goals. The report should describe the approach taken in satisfying the IPEEE goal. If it is based on demonstration of the conformance of the plant design to regulatory requirements, a list of potential vulnerabilities (if any) should be provided. If a bounding analysis is used, the report should describe the hazard analysis for wind and tornado effects, tornado missile risk analysis, fragility analysis for structures and equipment, and the results of the bounding analysis. It should also describe the conservatism in the analysis. If a detailed risk assessment is conducted, it should describe the system analysis models used and the estimated frequency of core damage from extreme wind effects, together with uncertainty bounds. If the mean core damage frequency is greater than  $1 \times 10^{-6}$  per year, the systems and components contributing significantly to the core damage frequency should be discussed. The robustness of the findings should be demonstrated through sensitivity studies by varying the assumptions, models and parameter values.

### **4.4 Staff Review Methodology**

#### **4.4.1 Review of Plant Design Information**

The staff review of this information consists of confirming that the plant design bases (i.e., design wind velocity, design basis tornado parameters, tornado missile spectrum) are correctly stated and used in the IPEEE, that the IPEEE has considered any meteorological and terrain changes subsequent to plant design, and that the structural design details (i.e., layout, design strategies and dimensions) are accurate. The reviewer may rely on the NRC staff's previous review done during the plant design stage and may use the information in the Safety

Evaluation Report to confirm the adequacy of the reported information. The reviewer may also perform an independent review of the plant design along with a plant walkdown if necessary.

#### **4.4.2 Plant Walkdown**

The staff reviews the procedures used by the utility in conducting the walkdown inspection of the plant. The walkdown may have been to confirm that the plant design meets current regulatory requirements and to identify any potential vulnerabilities. The walkdown may also have been to gather additional information such as deviations from design documents and site-specific features (e.g., plant design and layout that may modify the wind flow pattern, and potential sources of tornado missiles). The documentation of the walkdown procedures and findings are reviewed by the reviewer. A confirmatory walkdown may also be performed by the reviewer to aid in this review. The review focuses on the vulnerabilities identified by the utility in its walkdown and confirmed by the reviewer; the methods proposed by the utility for treating the vulnerabilities are examined for their impact on safety and cost.

#### **4.4.3 Screening Procedures**

The reviewer examines the basis for screening procedures used in treating the extreme winds.

##### **4.4.3.1 Procedure Based on Conformance to Regulatory Requirements**

The reviewer focuses on confirming that the plant design meets the current regulatory requirements in the extreme winds area. The design wind velocity, the parameters of the design basis tornado, tornado missile spectrum used, design criteria used for wind and tornado loading including missile protection methods, and structures and equipment designed to withstand tornado effects are reviewed. The applicable standard review plan sections are followed in this review.

The review also focuses on the findings of walkdown in terms of the potential vulnerabilities and their importance to the plant risk.

##### **4.4.3.2 Procedures Based on Plant Design Review**

The staff review focuses on the justification for realistic criteria used in estimating the plant capacity to withstand extreme wind effects. These criteria could be in the selection of design basis tornado parameters, tornado missile spectrum and use of in-place strength of materials. The assumptions and underlying data are reviewed.

##### **4.4.3.3 Bounding Analysis**

All the key elements of the bounding analysis are reviewed depending on the level at which the screening is done. In some cases, the hazard may be shown to be acceptably low; in others, it may be assumed that any safety-related structure failure would lead to core damage and the frequency of that event is shown to be less than the IPEEE screening criterion. If the bounding result does not meet the screening criterion, a detailed risk assessment is performed and reported by the utility.

##### **4.4.3.3.1 Hazard Analysis**

The site specific data and models used in describing the straight wind, hurricane and tornado hazard analyses are reviewed. The staff review also concentrates on the uncertainties in the models and model parameters and on how the uncertainties are propagated to obtain the family of hazard curves.

#### **4.4.3.3.2 Wind Fragilities**

The definition of failure, failure modes and capacities in different failure modes are reviewed. The list of structures and outdoor equipment needed for safety of the plant against extreme wind effects are reviewed. Their fragility calculations are reviewed. The review also focuses on the findings of the detailed walkdown. If a tornado missile risk analysis is reported, the models used in the analysis, and the inventory of objects available as missiles are reviewed. The results of the missile risk analysis are compared with the published results for representative nuclear power plant sites studies by Twisdale et al. [Twisdale, Dunn 1981 and Twisdale 1988].

The bounding assumptions used in the analysis about the hazard and fragility of the plant are reviewed for reasonableness.

#### **4.4.4 Detailed Risk Assessment**

The event and fault trees described in the report are reviewed; the simplifying assumptions made in the event tree and fault tree development are examined using sensitivity studies. If the reported mean core damage frequency is less than the IPEEE screening criterion, then plant vulnerabilities are reviewed. In case the IPEEE criterion is not met, the proposed actions to reduce the extreme wind risk are reviewed to judge if they are reasonable and cost-effective.

#### **4.5 Staff Documentation Requirements**

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that this evaluation is sufficiently complete and adequate to support the following type of statement to be included in the staff's evaluation report:

- "The analysis of extreme wind effects, including the identification of key wind-initiated accident sequences and plant-specific vulnerabilities, are acceptable, since this analysis provides a reasonable basis for decision making concerning the importance of the sequences and vulnerabilities. The methods used for screening, risk quantification and plant inspection for extreme wind effects are acceptable.
- This analysis provides an acceptable basis for satisfying, in part, the requirements of the Commission's Severe Accident Policy Statement insofar as it concerns the analysis and identification of extreme wind induced accident sequences and vulnerabilities."

#### **4.6 References**

- Prassinis, P.G., J.B. Savy, C.Y. Kimura, G.E. Cummings, R.C. Murray, R.J. Budnitz, and M.K. Ravindra (1989), *Individual Plant Examinations for External Events: Guidance and Procedures*, Lawrence Livermore National Laboratory, (NUREG/CR-5259, UCID-21554).
- Twisdale, L.A. (October 1988), *Probability of Facility Damage from Extreme Wind Effects*, (Journal of Structural Engineering, Vol. 114, No. 10, pp. 2190-2209).
- Twisdale, L.A. and W.L. Dunn (August 1981), *Tornado Missile Simulation and Design Methodology*, Electric Power Research Institute, Palo Alto, California (EPRI NP-2005).



## 5.0 EXTERNAL FLOODING IPEEE

This section provides procedures for reviewing a flood IPEEE submitted by a utility in response to the Severe Accident Policy Statement. The objective of performing a flood IPEEE is to systematically examine the plant and identify flood plant-specific vulnerabilities to severe accidents.

The purpose of this review is to verify that adequate and sufficient information has been submitted and to determine whether the utility has:

- Gained an appreciation for severe accidents initiated by flooding, through significant utility involvement in the review,
- Identified the most likely flood initiated accident sequences and dominant components with respect to core damage and fission product release,
- Gained a quantitative understanding of the overall probability of flood core-damage and fission product release,
- Identified possible plant modification or upgrades that would reduce the probability of flood core damage and fission product release

### 5.1 Criteria Used for NRC Staff Review

There are a number of phenomena that can cause external flooding. For each cause, or source, of flooding, a facility may be exposed to one or a number of flood hazards. A distinction must be made between the sources or causes of and the actual nature of the hazard loading that a structure is exposed to. In most cases, the principal hazard of interest is submergence or inundation. However, depending on the cause of flooding (e.g. river flooding, coastal storm surge) and the hazard (e.g., submergence, wave forces) the consequences can be very different.

In most cases, floods hazards are characterized in terms of the depth of flooding. This is reasonable since the depth of inundation is probably the single most relevant measure of flood severity.

Consequently the review of plant safety with respect to flood hazards is divided into two separate groups. The placement of a site in one or the other of those two groups constitutes the first screening in the analysis. The criterion used in this first screening is entirely based on the physical location of the entire plant with respect to the elevation of possible floods.

Thus the first criterion for eliminating the plant site from further analysis of its safety with respect to flood hazards is :

*CRITERION 1 : The plant site is characterized as "high and dry".*

Consistent with the criteria developed by NRC for screening and selection of important sequences for internally-initiated accident vulnerabilities, the second criterion is for the screening of potentially important sequences generated by flooding. It is based on a bounding approach without considering the consequences specifically but by assuming that the maximum probability of core melt given a flood occurs is 0.1 [Prassinis et al. 1989]. Thus, using the criterion that any functional sequence that contributes one millionth or more per reactor per year to core-damage frequency leads to the following second criterion for analyzing the plant site for flood hazard :

*CRITERION 2 : The mean annual probability of flooding at the site is 1/100,000 or more.*

The first criterion is a rejection criterion but criterion 2 is not. It is not a threshold for action, but a threshold above which the consequences must be analyzed and mitigation action considered.

## **5.2 Acceptable Analysis Methods**

### **5.2.1 Level of the Method**

The IPEEE Guidance and Procedures Document has described procedures for treating flood effects in the IPEEE of nuclear power plants. The primary focus is on deciding whether the flood effects at a particular plant site require an in-depth plant examination or could be screened out as not contributing significantly to the core-damage frequency. Even for those plants where the flood effects are determined to require further evaluation, the methods included in the following aim to show how the IPEEE goal may be met by limited analyses and review of the plant systems and structures.

The analysis methods described in the IPEEE Guidance and Procedures Document consisted of screening procedures at different stages:

- Procedure based on physical, geological and topographical conditions only (is the plant "high and dry"?).
- Procedure based on conformance to current regulatory requirements.
- Procedure based on plant design review.
- Bounding analysis for core damage frequency.

As there are two types of criteria, given in section 5.1, there also are two acceptable levels of analysis (Level 1 and Level 2). The first level uses criterion 1 to determine whether the plant site is immune to flooding. If the plant site is deemed "high and dry", then flood hazard is not considered in the rest of the analysis. Level 2 uses criterion 2 and is necessarily more detailed than Level 1. If criterion 2 is satisfied (i.e. the mean annual probability of flooding is greater than 1/100,000) a complete consequence analysis is warranted.

### **5.2.2 Definition of "Vulnerabilities"**

A vulnerability will be considered as any plant configuration (equipment, structure, system or procedure) that is significant to an accident sequence using the IPE screening criterion. In its strict sense, identification of vulnerabilities requires some type of probabilistic analysis to indicate both accident sequences that are important to plant safety and individual dominant components within those accident sequences. However, vulnerabilities could still be identified at different stages of the screening analysis described below. Plant vulnerabilities can then be considered as those components important to core damage that have a relatively high failure potential with respect to flood events. Based on the experience and judgment of the analyst, vulnerabilities may be identified in the plant review and walkdown, and bounding analysis. The walkdown is primarily a vulnerability search.

### **5.2.3 Level 1 Acceptable Methods**

These methods will show beyond any doubts that the plant site is immune to flood hazards. The general method recommended consists of two steps :

#### *step 1*

determine the lowest elevation associated with important safety systems of the plant.

*step 2*

determine the highest possible flood level to be achieved regardless of frequency.

The first step consists in a careful examination of the plant physical characteristics described on its drawings. The second step requires a careful examination of the local, possibly regional, topography and of the local and possibly regional hydrological events characteristics. The main sources of flooding to be considered are :

- Stream flooding
- Storm surge
- Seiche
- Tsunamis
- Floods due to upstream dam failure
- Coastal flooding
- Precipitation (e.g. rainfall, snow melt)
- Any combination of the above

Note that there may exist a number of different situations that lead to flooding due to a particular source. For example, stream flooding may result from storm precipitation, excessive dam-spillway flow, or stream blockage.

A plant located at a site found to be "high and dry" will not have any vulnerability to flood hazard.

#### **5.2.4 Level 2 Acceptable Methods**

These methods provide a quantitative estimate of the frequency of each possible flood elevation so that the frequency of the lowest important plant elevation, described in 5.2.2 above, is evaluated. These hazard estimation methods include, but are not limited to, one of the following current techniques :

- Statistical hazard estimates

These are the flood-based (historical data) statistical techniques which are essentially data analysis techniques. (e.g., [NRC 1988]).

- Probabilistic flood hazard estimates

These techniques rely both on the available flood data and on the empirical and physical modeling of the entire chain of events in the flood process. (e.g., [NRC 1988])

Note that if criterion 2 is satisfied, (i.e., if the mean annual probability of flooding is greater or equal to 1/100,000), it is necessary to provide an estimate of the uncertainty in the flood hazard so that they can be included in the risk assessment calculations. Methods are available to account for these uncertainties as described in references [McCann et al. 1985 and NUREG/CR-2300].

#### **5.2.5 Plant Response Analysis**

In the case of a Level 1 analysis, no response analysis is performed.

The response of a plant to flooding consists in some components or systems of the plants losing their effectiveness or operability leading to local or general loss of functions of some or all safety systems. Thus in Level 2, the plant response analysis consists of a consequence analysis using the flood loading inputs described in 5.2.4 above, and the methods described in [McCann et al. 1985 and NUREG/CR-2300].

A careful application of these methods leads to the identification of the vulnerabilities to flood.

### **5.3 Information Submittal Requirements**

#### **5.3.1 Description of the Plant**

The data needs for this step depend on the level of analysis performed. A Level 1 (see section 5.2) requires only general information on the topography of the site, drawings of the buildings locations, indicating elevations and all information relevant to flood hazard. For example, the existence of flood doors will be mentioned and the details of the flood door will be added to the data on general information. As the level of the analysis moves to a Level 2, the need for more detailed information on plant geometry and equipment description becomes necessary.

It is important that a complete description of the flood barriers be given, and in addition, any flood warning system will be described carefully so that a judgment can be made as to their adequacy for flood mitigation.

#### **5.3.2 Design Basis for Flood Hazard**

The flood history must be provided. This includes the dates, levels, peak discharges, and related information for the major historical flood events in the site region. In this context a flood is defined as any abnormally high water stage or overflow from a stream, floodway, lake, or coastal area that results in significantly detrimental effects. It includes stream floods, surges, seiches, tsunamis, dam failures, ice jams, floods induced by landslides, and similar events.

The general capability of the safety related facilities, systems, and equipment to withstand floods and flood waves must be discussed. The design flood protection for safety related components and structures of the plant should be based on the highest calculated flood water level elevations and flood wave effects (design basis flood) resulting from analyses of several different hypothetical causes. Any possible flood condition up to and including the highest and most critical flood level resulting from any of several different events should be considered. The flood potential from streams, reservoirs, adjacent watersheds, and site drainage should be discussed and documented. The probable maximum water level from a stream flood, surge, seiche, combination of surge and stream flood in estuarine areas, wave action, or tsunami (whichever is applicable and/or greatest) may cause the highest water level at safety related facilities. Other possibilities are the flood level resulting from the most severe flood wave at the plant site caused by an upstream or downstream landslide, dam failure, or dam breaching resulting from a seismic or foundation disturbance. The effects of coincident wind-generated wave action should be superimposed on the applicable flood level. The assumed hypothetical conditions should be evaluated both statically and dynamically to determine the appropriateness of the design flood protection level. The topical information that should be included is generally outlined in section 2.4.3 through 2.4.6 of reference [Regulatory Guide 1.70].

#### **5.3.3 Data Base and Methodologies Used in the Analysis**

The data base on flood information must be carefully documented as to its content and its origin.

The widely accepted methodologies are reviewed only with respect to the way they are used in the particular analysis. Only new or not-reviewed methodologies are the object of complete detailed review of their conceptual and development aspects. For such cases the methodologies must be carefully documented to permit evaluation by the staff.

#### **5.3.4 Overall Findings**

The findings consist of a statement indicating the completeness of the identification of site flood characteristics and flood design bases. It also summarizes the conclusions of the analysis and the arguments used in that analysis.

#### **5.3.5 Detailed Findings**

Specific details of the arguments used in the flood analysis are stated if they constitute important key steps in the analysis. For example the value of the parameters used in a probabilistic flood analysis are identified.

#### **5.4 Staff Review Methodology**

The staff reviews both the methodology and the data used in the flood analysis.

The review of the methodology consists only in checking that the methodology has been used properly if it is a commonly used and accepted methodology. On the other hand a new or not-yet-proven methodology is the object of a thorough evaluation for which the staff may need additional aid from experts in the field.

For the review of the data the staff reviews all available relevant publications, such as the publications of the U.S. Geological Survey (USGS), National Oceanic and Atmospheric Administration (NOAA), Soil Conservation Service (SCS), Corps of Engineers, applicable state and river basin agencies, and others to ensure that historical maximum events and the flood response characteristics of the region and site have been identified. The adequacy of any existing flood barriers is evaluated by the staff as well as the design of existing warning systems and the plans of action following such warning, including estimates of time operator responses. Similar material, in addition to applicant-supplied information, is reviewed to identify independently the potential sources of flooding. The staff assesses the consistency of the results presented by the utility with its own understanding of the data and analysis methods.

#### **5.5 Staff Documentation Requirements**

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this review plan, and concludes that the evaluation is sufficiently complete and adequate to support the following type of statement to be included in the staff's evaluation report:

"The analysis of the hazard at this site has revealed that it is immune to such hazard and therefore does not consider it in the rest of the analysis."

or,

"The analysis of the flood hazard at this site has revealed that the flood levels affecting safety at the plant have a total annual frequency of occurrence no greater than 1/100,000 and is deemed acceptable."

or,

"The analysis of the flood hazard at this site has lead to an estimate of the core-melt frequency no greater than one millionth per year and is deemed acceptable."

or,

"The analysis of the flood hazard at this site, including the identification of the flood-initiated accident sequences and plant-specific vulnerabilities, are acceptable, since it



provides a basis for decision-making concerning the importance of the identified sequences and vulnerabilities."

and finally,

"This analysis provides an acceptable basis for satisfying, in part, the requirements of the Commission's Severe Accident Policy Statement insofar as it concerns the analysis and identification of flood- initiated accident sequences and vulnerabilities."

## **5.6 References**

McCann, M., J. Reed, C. Ruger, K. Shiu, T. Teichmann, A. Unione, and R. Youngblood (1985), *Probabilistic Safety Analysis Procedure Guide*, (NUREG/CR-2815).

National Research Council (1988), *Estimating Probabilities of Extreme Flood Methods and Recommended Research*, National Academy Press, Washington, DC.

NUREG/CR-2300, *PRA Procedures Guide*.

Prassinis, P.G., J.B. Savy, C.Y. Kimura, G.E. Cummings, R.C. Murray, R.J. Budnitz, and M.K. Ravindra (1989). *Individual Plant Examinations for External Events: Guidance and Procedures*, Lawrence Livermore National Laboratory, (NUREG/CR-5259, UCID-21554, Draft).

Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*.

## **6.0 TRANSPORTATION AND NEARBY FACILITY ACCIDENTS IPEEE**

This section provides procedures for reviewing a Transportation and Nearby Facility Accident IPEEE submitted by a utility in response to the Severe Accident Policy Statement. The objective of performing a Transportation Accident IPEEE is to systematically examine the plant and to identify transportation accident related plant-specific vulnerabilities to severe accidents.

The purpose of this review is to verify that sufficient information has been submitted and to determine whether the utility has:

- Gained an appreciation for severe accidents initiated by transportation accidents through significant utility involvement,
- Identified the most likely transportation or nearby facility initiated accident sequences and dominant components with respect to core damage and fission product release,
- Gained a quantitative understanding of the overall annual frequency of transportation or nearby facility accident core damage and fission product release, and
- Identified possible plant modification or upgrades that would reduce the annual frequency of transportation induced core damage and fission product release.

### **6.1 Criteria Used for NRC Staff Review**

#### **6.1.1 Screening Criteria for Identifying Potentially Important Accidents**

For screening purposes the plant response to transportation accidents or accidents at nearby facilities must be evaluated:

- If the mean accident probability, taking into account the uncertainties, within the interaction area summed over all modes and routes is greater than  $10^{-5}$  per year.
- If the mean probability, taking into account the uncertainties, of a single accident is greater than  $10^{-6}$  per year.

These criteria are used by the NRC staff to differentiate those accident sequences that require further staff review from those that do not.

#### **6.1.2 Screening Criterion for Identifying "Vulnerabilities"**

If a single accident or all modes of accidents have a mean probability equal to or greater than the criteria given in Section 6.1.1 then it is necessary to perform a plant response evaluation for the identified accident or set of accidents.

In the "IPE Generic Letter" covering internally-initiated accident vulnerabilities, the NRC has developed screening criteria for the selection of potentially important sequences. Of these IPE criteria, only one will be mentioned here as particularly applicable to transportation or nearby facility accident-initiated sequences. This criterion is that a functional accident sequence must be considered and evaluated further if:

- the functional sequence contributes  $1 \times 10^{-6}$  or more per reactor year to core-damage frequency.

This is not an acceptance criteria, nor a threshold for action, but only a threshold above which a sequence must be documented and considered, to determine if a cost-effective change should

be made to improve safety. The converse is that a functional sequence can be screened out and not considered further if it falls below the above criterion.

This criterion is used by the NRC staff to differentiate those sequences that require further staff review from those that do not.

### **6.1.3 Definition of "Vulnerabilities"**

A key part of the IPEEE process is the identification of plant "vulnerabilities". For the purpose of this Section a transportation or nearby facility accident "vulnerability" will be considered as any plant configuration (equipment, structure, system or procedure) that contributes to an accident sequence that leads to core-damage frequency greater than the criterion given in Section 6.1.2.

A vulnerability can either be within or outside the "design basis" and the NRC's current "regulatory envelope".

### **6.1.4 Criteria Used in Reviewing the Analysis**

In reviewing the analysis submitted, the staff examines whether the methods used are up-to-date, state-of-the-art methods correctly applied using appropriate data. The staff examines whether all modes of transportation accidents have been re-evaluated based on updated local surveys and data. The staff also examines that all potential accidents at any nearby facilities have been re-evaluated based on updated local surveys and data. The staff also examines the analyses to determine if the uncertainties involved have been adequately accounted for.

## **6.2 Acceptable Analysis Methods**

### **6.2.1 Types of Methods Involved**

There are potentially four parts to the analysis:

- 1) Determination if an accident or set of accidents must be further evaluated.
- 2) Determination if an accident sequence poses a threat to the safety of the reactor.
- 3) Determination if an accident sequence leads to a probability of core-damage greater than the criteria given in Section 6.1.2 and identification of transportation accident vulnerabilities.
- 4) Determination of a mitigation approach for any identified vulnerabilities.

Each part of the analysis - if required - has its own set of acceptable methods.

The acceptable methods depend upon the type of nearby facilities and upon the mode of transportation, the route (i.e., if the accident occurs within or outside the plant's exclusion area) and if the transportation modes or nearby facilities involve hazardous materials.

Reference should be made to Kimura, Budnitz 1987; Kimura, Prassinis 1989; and Sections 2.2.1, 2.2.2, 2.2.3, 3.5.1.5 and 3.5.1.6 of NUREG 0800 1981. Kimura, Budnitz 1987; Kimura, Prassinis 1989; and NUREG 0800 1981 give examples of acceptable analysis methods.

It should be noted that many transportation accidents or accidents at nearby industrial facilities involve fire. Thus, it may be useful to follow some or all of the procedures given in Section 3 of this report.

### **6.2.2 Scope of an Acceptable Analysis**

The scope of the vulnerability search analysis must satisfy the following, and the review ensures that this scope has been accomplished:

- The analysis must include all accident sequence initiated by any transportation accident which includes aviation traffic of all types (commercial, general, and military), marine traffic (ships and barge), gas/oil/chemical pipelines, railroad traffic and truck traffic.
- The analysis must also include accidents at any nearby industrial or military facilities from similar transportation accidents and other potential initiators such as earthquake, floods and tornadoes. This will require reference to Sections 2, 4 and 5 of this report.
- Transportation accident or accidents at nearby facilities that initiate accident sequences that also involve important non-transportation-accident related failures (equipment failure, human errors, equipment out-of-service, etc.) must be included as appropriate.
- An "accident sequence" is considered to be any functional combination of events leading to unacceptable consequences (here, this means a "core-damage accident") with an unacceptable annual frequency (here, this means that the mean calculated value of "events per year" is above either of the screening criteria discussed above).
- Arson and other sabotage-type acts are explicitly excluded from the scope here.

### **6.3 Information Submittal Requirements**

The information submittal requirements depend upon whether the accident probability exceeds the levels given in Sections 6.1.1 and 6.1.2. If the accident probability is less than the criteria given in Section 6.1.1, then the submittal requirements are covered in Sections 6.3.1-6.3.3. If the criteria given in Section 6.1.1 are exceeded then the additional information described in Section 6.3.4 as needed. Finally, if the criteria given in Section 6.1.2 are exceeded, then the information described in Section 6.3.5 is needed.

#### **6.3.1 Locations and Routes**

The analysis should provide maps showing the location and distance from the nuclear plant of all significant manufacturing plants; chemical plants; refineries; storage facilities; mining and quarrying operations; military bases; missile sites; transportation routes (air, land, and water); transportation facilities (docks, anchorages, airports); oil and gas pipelines, drilling operations, and wells; and underground gas storage facilities. It should show any other facilities that, because of the products manufactured, stored, or transported, may require consideration with respect to possible adverse effects on the plant. In addition, the analysis should show any military firing or bombing ranges and any nearby aircraft flight, holding, and landing patterns.

The maps should be clearly legible and of suitable scale to enable easy location of the facilities and routes in relation to the nuclear plant. All symbols and notations used to depict the location of the facilities and routes should be identified in legends or tables. Topographic features should be included on the maps in sufficient detail to adequately illustrate the information presented.

All facilities and activities within five miles of the nuclear plant should be considered. Facilities and activities at greater distances should be included as appropriate to their significance.

### 6.3.2 Descriptions

The descriptions of the nearby industrial, transportation, and military facilities identified in Section 6.3.1 should include the information indicated in the following:

Description of Facilities. A concise description of each facility, including its primary function and major products and the number of persons employed, should be provided in tabular form.

Description of Products and Materials. A description of the products and materials regularly manufactured, stored, used, or transported in the vicinity of the nuclear plant should be provided. Emphasis should be placed on the identification and description of any hazardous materials. Statistical data should be provided on the amounts involved, modes of transportation, frequency of shipment, and the maximum quantity of hazardous material likely to be processed, stored, or transported at any given time. The applicable toxicity limits should be provided for each hazardous material.

Pipelines. For pipelines, indicate the pipe size, pipe age, operating pressure, depth of burial, location and type of isolation valves, and the type of gas or liquid presently carried. Indicate whether the pipeline is used for gas storage at higher than normal pressure and discuss the possibility of the pipeline being used in the future to carry a different product than the one presently being carried (e.g., propane instead of natural gas).

Waterways. If the site is located on a navigable waterway, provide information on the location of the intake structure(s) in relation to the shipping channel, the depth of the channel, the location of locks, the type of ships and barges using the waterway, and any nearby docks and anchorages.

Airports. For airports, provide information on length and orientation of runways, type of aircraft using the facility, the number of operations\* per year by aircraft type, and the flying patterns associated with the airport. Plans for future utilization of the airport, including possible construction of new runways, increased traffic, or utilization by larger aircraft, should be provided. In addition, statistics on aircraft accidents should be provided for:

1. All airports within five miles of the nuclear plant,
2. Airports with projected operations greater than  $500d^2$  per year within 10 miles,\*\* and
3. Airports with projected operations greater than  $1000d^2$  per year outside 10 miles.\*\*

Provide equivalent information describing any other aircraft activities in the vicinity of the plant. These should include aviation routes, pilot training areas, and landing and approach paths to airports and military facilities.

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\* The FAA, in compiling airport use statistics, defines an aircraft operation as the airborne movement of aircraft in controlled or noncontrolled airport terminal areas and about given enroute fixes or at other points where counts can be made. There are two types of operations--local and itinerant. Local operations are performed by aircraft which: (a) Operating in the local traffic pattern or within sight of the airport, (b) Are known to be departing for, or arriving from, flight in local practice areas within a 20 mile radius of the airport, and (c) Execute simulated instrument approaches or low passes at the airport. Itinerant operations are all aircraft operations other than local operations [FAA 1982].

\*\* "d: is the distance in miles from the site.



### 6.3.3 Identification of Potentially Important Accidents

On the basis of the information provided in Sections 6.3.1 and 6.3.2 the analysis of accident probabilities for all modes of transportation routes and possible accident sequences, should be provided. Note if there are nearby industrial or military facilities which could pose a threat to the safety of the nuclear plant in terms of generating missiles, over pressure, fire or release of toxic materials then the potential for accidents at the nearby facilities must be evaluated. A discussion of the application of the screening criteria given in Section 6.1.1 for each of the facilities and/or all modes of accidents should be given. If all modes of accidents lead to probabilities of accidents less than the criteria given in Section 6.1.1, then no added information is required. However, if the probabilities are higher than the screening criteria, then at least the added information described in Section 6.3.4 is required.

### 6.3.4 Evaluation of Potentially Important Accidents

For the accidents identified in Section 6.3.3 that have sufficiently high probability of occurrence it is necessary to document the analyses performed to determine if they can pose a threat to the safety of the nuclear plant.

The criterion used to determine if the accidents pose a threat is given in Section 6.1.2. An evaluation of plant's response to each of the potential important accidents should be provided and where applicable an evaluation of the application of the criterion given in Section 6.1.2 is required.

The accident categories discussed below should be considered as applicable.

1. Explosions. Accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels should be considered for facilities and activities in the vicinity of the plant where such materials are processed, stored, used, or transported in quantity. Attention should be given to potential accidental explosions that could produce a blast overpressure on the order of 1 psi or greater at the nuclear plant, using recognized quantity-distance relationships. Missiles generated in the explosion should also be considered, and an analysis should be provided.
2. Flammable Vapor Clouds (Delayed Ignition). Accidental releases of flammable liquids or vapors that result in the formation of unconfined vapor clouds should be considered. Assuming that no immediate explosion occurs, the extent of the cloud and the concentrations of gas that could reach the plant under "worst-case" meteorological conditions should be determined. An evaluation of the effects on the plant of detonation and deflagration of the vapor cloud should be provided. An analysis of the missiles generated as a result of the detonation should be provided.
3. Toxic Chemicals. Accidents involving the release of toxic chemicals (e.g., chlorine) from onsite storage facilities and nearby mobile and stationary sources should be considered. If toxic chemicals are known or projected to be present onsite or in the vicinity of a nuclear plant or to be frequently transported in the vicinity of the plant, releases of these chemicals should be evaluated. For each postulated event, a range of concentrations at the site should be determined for a spectrum of meteorological conditions. These toxic chemical concentrations should be used in evaluating control room habitability.
4. Fires. Accidents leading to high heat fluxes or to smoke, or nonflammable gas-or chemical-bearing clouds or from the release of hazardous materials as the consequence of fires in the vicinity of the plant should be considered. Fires in adjacent industrial and chemical plants and storage facilities and in oil and gas pipelines, brush and forest fires, and fires from transportation accidents should be

evaluated as events that could lead to high heat fluxes or to the formation of such clouds. A spectrum of meteorological conditions should be included in the dispersal analysis when determining the concentrations of nonflammable material that could reach the site. These concentrations should be used to evaluate control room habitability and to evaluate the operability of diesels and other equipment.

5. Collisions with Intake Structure. For nuclear power plant sites located on navigable waterways, the evaluation should consider the probability and potential effects of impact on the plant cooling water intake structure and enclosed pumps by the various size, weight, and type of barges or ships that normally pass the site, including any explosions incident to the collision. This analysis should be used to determine whether an additional source of cooling water is required.
6. Liquid Spills. The accidental release of oil or liquids which may be corrosive, cryogenic, or coagulant, should be considered to determine if the potential exists for such liquids to be drawn into the plant's intake structure and circulating water system or otherwise to affect the plant's safe operation.

### **6.3.5 Identification and Evaluation of Accident Vulnerabilities**

If any of the accidents evaluated in Section 6.3.4 lead to core-damage frequencies greater than the criterion given in Section 6.1.2, then it is necessary to provide the needed information to identify the vulnerabilities as defined in Section 6.1.4. In many instances this information will naturally be developed as part of the information provided in fulfilling the requirements of Section 6.3.4. If this is not the case, then it is necessary to perform and document an analysis that identifies the vulnerabilities arising from transportation accidents or other accidents at nearby facilities.

In addition, it is also necessary to document any proposed mitigation approaches to reduce the probabilities of core-damage to below the levels given in Section 6.1.2.

## **6.4 Staff Review Methodology**

### **6.4.1 Identification of Potential Transportation Accidents and other Hazards in Site Vicinity**

The reviewer must first determine that all potential transportation accident and other hazards in the site's vicinity have been identified. The following procedures are used:

- The reviewer should be especially alert, for any potentially hazardous activities in close proximity of the plant since the variety of activities having damage potential at ranges under about one kilometer can be very extensive. All identified facilities and activities within 5 miles of the plant should be reviewed. Facilities and activities at greater distances should be considered if they otherwise have the potential for affecting plant safety-related features. At the IPEEE stage of the review process most hazards will already have been identified. Emphasis should be placed on any new information. Any analyses pertaining to potential accidents involving hazardous materials or activities in the vicinity of the plant will be reviewed to ensure that results are appropriate in light of any new data or experience which may be available. Facilities which are likely to either produce or consume hazardous materials should be investigated as possible sources of traffic of hazardous materials past the site. The reviewer should determine that all modes of transportation and transportation routes have been identified.
- Information should be obtained from sources other than the data submitted wherever available, and should be used to check the accuracy and completeness of the information submitted. This independent information may be obtained from sources

such as U.S. Geological Survey (USGS) maps and aerial photos, published documents, contacts with State and Federal agencies, and from other nuclear plant applications (especially if they are located in the same general area or on the same waterway). To the extent that definitive information is available, future potential hazards over the proposed life of the plant should be reviewed.

- The specific information relating to types of potentially hazardous material, including distance, quantity, and frequency of shipment, is reviewed to assess the adequacy of the interaction areas used in the screening analysis.

#### **6.4.2 Review of the Screening Analysis for Selecting Potentially Important Accidents**

The data bases used to calculate the probability of given accidents are carefully reviewed. The applicant's probability calculations are reviewed, and an independent probability analysis is performed by the staff if the potential hazard is considered significant enough to affect the identification of plant vulnerabilities.

Probabilistic models should be tested, where possible, against all available information. If the model or any portion of it, by simple extension, can be used to predict an observable accident rate, this test should be performed.

The choice of the interaction areas used in the screening analysis should be carefully reviewed relative to the potential severity of each accident.

Special attention should be given to the review of a site where several man-made hazards are identified, but none of which, individually, has a probability exceeding the screening criteria stated in Section 6.1.1. The objective of this special review should be to assure that the aggregate probability is also less than the aggregate screening criteria given in Section 6.1.1.

#### **6.4.3 Review of the Evaluation of Potentially Important Accidents**

If any of the accidents identified in the Section 6.4.2 have probabilities greater than the criteria given in Section 6.1.1, then it is necessary to review the analysis evaluating the significance of these accidents. It is difficult to be very specific in describing a review methodology because so many different types of analyses can be involved and so many different approaches can be taken to show that the given accident sequence would not lead to core-damage, or if it leads to core-damage the probability is less than the screening criterion given in Section 6.1.2. Reference should be made to the NUREG 0800 1981 Sections 2.2.1-2.2.2, 2.2.3, 3.5.1.5 and 3.5.1.6. Also Kimura, Budnitz 1987 and Kimura, Prassinis 1989 should be consulted. It should also be noted that a number of accidents lead to fires, thus, it may be necessary to invoke the review methodology given in Section 3 of this report.

In Kimura, Prassinis 1989 flow charts and step-by-step procedures are given for performing the various evaluations needed for each mode of transportation. The use of these flow charts and step-by-step procedures are useful to organize the review process in an orderly fashion. Considerable judgment is required and in some cases additional reviewers may be required to judge the adequacy of any given analysis step; e.g., if a complex, structural analysis is required to show that an airplane crash will not damage the core or the given accident sequence is embedded in a large PRA analysis.

The review carefully examines the criteria used in the analysis for deciding whether equipment, systems, etc., are damaged to the point of failure by a given accident sequence. The sensitivity of the results to the criteria and assumptions used are reviewed.

The review also concentrates on the analysis of how accident induced transients or failures can effect other systems and components and possibly lead to plant-level failure (a core-damage accident or worse).

#### **6.4.4 Review of the Accident-Induced Vulnerabilities**

Assuming that some accident sequence leads to a core-damage frequency equal or greater than the criterion given in Section 6.1.2, then an acceptable transportation and nearby facility accident IPEEE analysis must include the identification of plant vulnerabilities as defined in Section 6.1.3. The identification of the vulnerabilities is reviewed, including review of how sensitive the insights are to various assumptions and uncertainties.

Each identified vulnerability should be described in qualitative terms, along with a discussion in quantitative terms of what factors underlie the finding that the particular vulnerability is important.

#### **6.4.5 Review of Proposed Mitigation Measures**

An acceptable transportation and nearby facility accident analysis must include for each vulnerability identified any proposed plant improvements for the prevention and mitigation of severe accidents. The review will examine each of the proposed improvements and the basis used to establish that the proposed improvements would adequately mitigate the identified vulnerabilities.

### **6.5 Staff Documentation Requirements**

The reviewer verifies that sufficient information has been provided to satisfy the requirements of the IPEEE review plan, and concludes that his evaluation is sufficiently complete and adequate to support the following type of statement to be included in the staff's evaluation report:

"The analysis of transportation and nearby facility initiated accident sequences, including the identification of key transportation initiated accident sequences and of plant-specific vulnerabilities, are acceptable, since this analysis provides a reasonable basis for decision-making concerning the importance of the identified sequence and vulnerabilities".

"This analysis provides an acceptable basis for satisfying in part, the requirements of the Commission's Severe Accident Policy Statement in so far as it concerns the analysis and identification of transportation and nearby facility accident initiated accident sequences and vulnerabilities".

### **6.6 References**

Federal Aviation Administration (December 1982), *FAA Statistical Handbook of Aviation, Calendar Year 1982*, U.S. Department of Transportation, FAA, Office of Management Systems, Information Analysis Branch, Washington, D.C. .

Kimura, C. Y. and R.J. Budnitz (December 1987), *Evaluation of External Hazards to Nuclear Power Plants in the United States*, Lawrence Livermore National Laboratory, Livermore, CA., (NUREG/CR-5042, UCID-21223).

Kimura, C. Y. and P.G. Prassinis (February 1989), *Evaluation of External Hazards to Nuclear Power Plants in the United States - "Other External Hazards"*, (NUREG/CR-5042, UCID-21223, Supplement 2)

NUREG-0800 (formerly issued as NUREG-75/087) (July 1981), *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, LWR Edition, U.S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation, Washington, D.C. .



## Appendix Outline for Industry Submittals

This outline gives a standard format for submitting the results of performing an Individual Plant Examination for External Events (IPEEE) by a utility.

### Section 1. INTRODUCTION AND SUMMARY

- 1.1 General Plant Location and Description
- 1.2 General Procedure Used for Conducting IPEEE
- 1.3 Summary of IPEEE Results

### Section 2. SEISMIC IPEEE

- 2.1 Plant Seismic Design Criterion
- 2.2 Description of the Seismic IPEEE Method
  - 2.2.1 Description of the Seismic IPEEE Review Team
  - 2.2.2 Description of the Analysis Methods Used for the Seismic IPEEE Including Screening Criteria
- 2.3 Results of the Seismic IPEEE
  - 2.3.1 Description of the Front-Line and Support Systems
  - 2.3.2 Description of the Seismic Hazard or Review Level Earthquake
  - 2.3.3 Documentation of the Plant Walkdown(s), Data Collected and Findings
  - 2.3.4 Description of the Plant Models and Accident Sequences
    - 2.3.4.1 Identification of Screened-Out Components
    - 2.3.4.2 Identification of Screened-In Components
  - 2.3.5 Description of the Methods, Data and Criterion Used in Analyzing Components and Accident Sequences
  - 2.3.6 Identification of Plant Vulnerabilities - High Frequency or Low Capacity Accident Sequences and Dominant Components
- 2.4 Conclusion from the Seismic IPEEE
  - 2.4.1 Risk Significance of the Plant Vulnerabilities
  - 2.4.2 Possible Plant Modification/Upgrades

### Section 3. INTERNAL FIRES IPEEE

- 3.1 Plant Seismic Design Criterion
- 3.2 Description of the Internal Fires IPEEE Method
  - 3.2.1 Description of the Internal Fires IPEEE Review Team
  - 3.2.2 Description of the Analysis Methods Used for the Internal Fires IPEEE Including Screening Criteria
- 3.3 Results of the Internal Fires IPEEE
  - 3.3.1 Description of the Front-Line and Support Systems
  - 3.3.2 Description of the Internal Fires Hazard
  - 3.3.3 Documentation of the Plant Walkdown(s), Data Collected and Findings
  - 3.3.4 Description of the Plant Models and Accident Sequences
    - 3.3.4.1 Identification of Screened-Out Components
    - 3.3.4.2 Identification of Screened-In Components
  - 3.3.5 Description of the Methods, Data and Criterion Used in Analyzing Components and Accident Sequences



- 3.3.6 Identification of Plant Vulnerabilities - High Frequency or Low Capacity Accident Sequences and Dominant Components
- 3.4 Conclusion from the Internal Fires IPEEE
  - 3.4.1 Risk Significance of the Plant Vulnerabilities
  - 3.4.2 Possible Plant Modification/Upgrades

#### Section 4 . EXTREME WIND IPEEE

- 4.1 Plant Seismic Design Criterion
- 4.2 Description of the Extreme Wind IPEEE Method
  - 4.2.1 Description of the Extreme Wind IPEEE Review Team
  - 4.2.2 Description of the Analysis Methods Used for the Extreme Wind IPEEE Including Screening Criteria
- 4.3 Results of the Extreme Wind IPEEE
  - 4.3.1 Description of the Front-Line and Support Systems
  - 4.3.2 Description of the Extreme Wind Hazard
  - 4.3.3 Documentation of the Plant Walkdown(s), Data Collected and Findings
  - 4.3.4 Description of the Plant Models and Accident Sequences
    - 4.3.4.1 Identification of Screened-Out Components
    - 4.3.4.2 Identification of Screened-In Components
  - 4.3.5 Description of the Methods, Data and Criterion Used in Analyzing Components and Accident Sequences
  - 4.3.6 Identification of Plant Vulnerabilities - High Frequency or Low Capacity Accident Sequences and Dominant Components
- 4.4 Conclusion from the Extreme Wind IPEEE
  - 4.4.1 Risk Significance of the Plant Vulnerabilities
  - 4.4.2 Possible Plant Modification/Upgrades

#### Section 5. EXTERNAL FLOODING IPEEE

- 5.1 Plant External Flooding Design Criteria
- 5.2 Description of the External Flooding IPEEE Method
  - 5.2.1 Description of the External Flooding IPEEE Review Team
  - 5.2.2 Description of the Analysis Methods Used for the External Flooding IPEEE, Including Screening Criteria
- 5.3 Results of the External Flooding IPEEE  
If the plant is found to be "high and dry", Section 5.3 stops here. Otherwise, Section 5.3.1 to 5.3.6 are included in the report when necessary.
  - 5.3.1 Description of the Front-Line and Support Systems
  - 5.3.2 Quantitative Description of the External Flooding Hazard (May Include Estimates of Uncertainty in a Flooding Level Hazard Curve or PMF)
  - 5.3.3 Documentation of the Plant Walkdown(s), Data Collected and Findings
  - 5.3.4 Description of the Plant Models and Accident Sequences
  - 5.3.5 Description of the Methods, Data and Criteria Used in Analyzing Components and Accident Sequences
  - 5.3.6 Identification of Plant Vulnerabilities with Respect to External Flooding
- 5.4 Conclusions from the External Flooding IPEEE
  - 5.4.1 Risk Significance of the Plant Vulnerabilities

#### 5.4.2 Possible Plant Modification and Upgrades

### Section 6. TRANSPORTATION AND NEARBY FACILITIES ACCIDENT IPEEE

#### 6.1 Nearby Industrial, Transportation and Military Facilities

##### 6.1.1 Location and Routes

##### 6.1.2 Descriptions

###### 6.1.2.1 Description of Nearby Facilities

###### 6.1.2.2 Description of Products and Materials

###### 6.1.2.3 Pipelines

###### 6.1.2.4 Waterways

###### 6.1.2.5 Airports

#### 6.2 Identification of Potentially Important Accidents

##### 6.2.1 Identification of all Potential Accidents

##### 6.2.2 Description of the Application of the IPEEE Screening Criteria to all Potential Accidents

###### 6.2.2.1 Identification of Screened Out Accidents

###### 6.2.2.2 Identification of Screened In Accidents (Potentially Important Accidents)

#### 6.3 Evaluation of Potentially Important Accidents

##### 6.3.1 Description of the Application of the Screening Criterion for Identification of Plant Vulnerabilities to the Potentially Important Accidents

##### 6.3.2 Documentation of the Analysis for Each Potentially Important Accident

#### 6.4 Identification and Evaluation of Accident Vulnerabilities

##### 6.4.1 Identification of Plant Vulnerabilities

##### 6.4.2 Significance of Vulnerabilities

##### 6.4.3 Possible Plant Modifications/Upgrades