NUREG/CR-5042
UCID-21223
Supplement 1

Evaluation of External Hazards to Nuclear Power Plants in the United States

Seismic Hazard

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Prepared for
U.S. Nuclear Regulatory Commission

Lawrence
Livermore
National
Laboratory

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Manuscript Completed: February 1988
Date Published: April 1988

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NRC FIN No. A0815

Abstract

As part of the research program supporting the implementation of the NRC Policy Statement on Severe Accidents, the Lawrence Livermore National Laboratory (LLNL) has performed a study of the risk of core damage to nuclear power plants in the United States due to seismic initiated events. The broad objective has been to gain an understanding of whether or not seismic events are among the major potential accident initiators that may pose a threat of severe reactor core damage or of large radioactive release to the environment from the reactor.

The analysis was based on two figures-of-merit, one based on core damage frequency and the other based on the frequency of large radioactive releases. Using these two figures-of-merit as evaluation criteria, it has been possible to ascertain that the risk from seismic initiated accidents is an important contributor to overall risk for the U.S. nuclear power plants studies.

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

LIST OF FIGURES

LIST OF TABLES

EVALUATION OF THE SEISMIC EXTERNAL HAZARD

EXECUTIVE SUMMARY

1. BACKGROUND

The Nuclear Regulatory Commission (NRC) has issued its Policy Statement on Severe Accidents [1]. This Policy Statement sets the goals and schedule for addressing issues relevant to severe accidents in the licensing of future plants and for the systematic examination of existing plants.

The NRC Severe Accident Policy Statement does not differentiate between events initiated within the power plant and events caused by external initiators, such as earthquakes, floods, and high winds. The evaluation of internally initiated events is more developed than the evaluation of externally initiated events. Therefore, the NRC staff is implementing the Severe Accident Policy Statement for internally initiated events [3] at this time. However, the implementation allowing inclusion of external events evaluation, is an option of the utility.

The evaluation of severe accidents initiated by external events will proceed in two phases [2]. The first phase will be an assessment of the margin provided by past and ongoing programs relative to external events beyond the design basis. In addition, this phase includes identification of areas where examination for external vulnerabilities is needed. The second phase will consist of a program for plant specific evaluation, if needed. Information developed in phase one will be used as guidance for phase two.

As part of the research program supporting the NRC Policy Statement on Severe Accidents [2], the Lawrence Livermore National Laboratory (LLNL) has performed a study of risk of core damage to nuclear power plants in the United States caused by external floods, high winds, internal fires and transportation accidents, [4]. The broad objective has been to gain an understanding, for existing U.S. light water reactor (LWR) power plants, of whether or not these external initiator are major potential accident initiators that may pose a threat of severe reactor core damage or of a large radioactive release from the reactor core. This report addresses this concern for the seismic external hazard and supplements the earlier report [4].

The specific tasks accomplished in the project have been:

- to consider the effects of the evolution of design requirements and design practices on plant seismic capacity;
- to identify other specific review areas of potential seismic vulnerability, including seismically induced fires and floods, spent fuel pools, and seismic common-mode failures;

- o to identify programs which address items 1 and/or 2 above, and assess the extent to which these programs provide useful information on seismic capacity of nuclear plants; and

- o to recommend incorporating the above items into the seismic margins program or other seismic vulnerability searches.

2. EVALUATION CRITERIA (FIGURES-OF-MERIT) SELECTED

The approach used in this study is the same as Ref. 4 but is directed at seismic event which lead to accident scenarios resulting in damage to the reactor core or release of radioactive material from the reactor core.

In order to examine the risk to U.S. nuclear power plants from any specific externally initiated event, it has been necessary to employ specific evaluation criteria to discriminate between the significant and the less significant levels of risk. These evaluation criteria have used the guidance provided by the NRC in their Safety Goal Policy Statement [5] and Policy Statement on Severe Accidents [1]. Two different figures-of-merit have been selected.

The first figure-of-merit is the core damage frequency. According to the NRC's Policy Statement on Safety Goals, the Commissioners explicitly stated one of their objective as:

"providing reasonable assurance, giving consideration to the uncertainties involved, that a core-damage accident will not occur at a U.S. nuclear power plant." [5]

In numerical terms (based on about 100 nuclear power plants operating over about a 40 year time period), this objective can be met if individual plants have a mean core damage frequencies in the range of about 10^{-5} or less per reactor year. This is not a firm numerical objective, but a range whose method of application by the NRC staff will continue to evolve over the next few years.

The second figure-of-merit is the frequency of a large release. In this same NRC Policy Statement on Safety Goals, the following was given as a general performance guideline:

"Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation." [5]

The current status of this guideline is that the NRC staff is giving detailed consideration to how such a performance guideline can be implemented, including how to define more precisely the definition of "a large release of radioactive materials to the environment". For the purposes of this report, a radioactive materials to the environment". For the purposes of this report, a large release of radioactive material to the environment has been defined as a release of a substantial fraction of the radioactive core in a time period relatively early in the postulated accident scenario. This definition has been taken from Probabilistic Risk Assessment (PRA) literature which has defined a "large early release". This rather imprecise definition will include those accident sequences whose release of significant fractions of the reactor core radioactive inventory occur within a few hours of the initiation of the accident. Operationally, this means that for the purposes of this report, a "large radioactive release" will correspond to those few PRA plant damage states and release categories associated with "large, early release" and such a release should occur with a frequency equal to or less than 1×10^{-6} per reactor year.

Using these two figures-of-merit, it has been feasible to ascertain whether the risk from externally initiated accidents is, or is not, an important contributor to overall risk for the U.S. nuclear power plants studied.

Because PRA methods are the most widely known and widely used approach for calculating the expected frequency of reactor core damage or of large radioactive release from the reactor core per reactor year of operation, this report has emphasized insights from various PRA studies. Large numerical uncertainties are associated with PRA's, and therefore the evaluation criteria has been based on broad (order-of-magnitude) types of comparison. It has been assumed that because the figures-of-merit are broadly based and not a direct numerical comparison of PRA results, it is possible to obtain insights into the issues surrounding the risk to U.S. nuclear power plants from seismic events even taking account of the large uncertainties in PRA results.

3. MATERIAL REVIEWED FOR THIS STUDY

The following technical material was reviewed for this project:

1) The NRC's regulatory approach to assuring that nuclear power plants are adequately protected against the seismic external initiator , i.e., Title 10 of the Code of Federal Regulations (10 CFR), other applicable regulations, the standard review plans (SRP), and applicable regulatory guides (RGs).

2) Technical papers and reports, including discussions of PRA methodology for the seismic initiator, technical studies on associated phenomena, and studies of the traditional engineering approaches to assuring protection against seismic events.

3) Documentation on how nuclear power plants are designed and constructed by their owners and reviewed by the NRC staff for protection from earthquakes, i.e. FSARs, PSARs, SARs, and Safety Evaluation Reports (SERs).

4) PRA literature on seismic initiators, including both full-scope PRA studies and partial PRA-type analyses.

5) Event data bases at nuclear power plants, including both the larger events of interest and the smaller events that did not cause serious problems.

6) NRC's past and current programs, such as the Systematic Evaluation Program, Seismic Safety Margins Research Program (SSMRP), Seismic Design Margins Program (SDMP) and Seismic Qualification of Equipment at Nuclear Power Plants (A-46).

Important aspects of each category have been examined, although not all of this material has been studied in detail.

4. CONCLUSIONS

The seismic external hazard has been found to be important with respect to both figures-of-merit. Accidents initiated by seismic events have core damage frequencies comparable to the first figure-of-merit of E-5 per year and large release frequencies comparable to the second figure-of-merit of E-6 per year. Even when considering the uncertainty ranges of these probabilistic estimates, the core damage and release frequencies bracket the figures-of-merit. Therefore, the seismic external hazard should be included in the Severe Accident Policy implementation.

It is possible to separate seismic probabilistic risk analysis into two parts. The first part is the probability of the seismic initiating event and the second part is the response of the plant to this initiating event. Any consideration of the seismic external hazard must include both the local seismic hazard and the plant's seismic response.

The seismic external hazard has been well studied including the performance of several seismic PRAs. There are presently several ongoing efforts that address different aspects of seismic hazards. These efforts include seismic hazard characterization, and assessments of a plant's response to design basis and beyond-design-basis earthquakes. The techniques and results of these efforts can be used to address severe accident concerns.

In addressing seismic vulnerability analyses, full-scope seismic PRA type of analysis is always an acceptable approach to assessing plant response and vulnerabilities. However, a seismic margins approach appears to provide the necessary degree of analysis, accuracy, and results needed to address these concerns. This methodology along with a possible screening approach based on site-specific seismic hazards can be used. The seismic margins approach should integrate the several seismic analysis efforts into a single application that will address several issues including the implementation of the severe accident policy.

EVALUATION OF THE SEISMIC EXTERNAL HAZARD

Chapter 1

INTRODUCTION

1.1 BACKGROUND

On August 8, 1985 the Nuclear Regulatory Commission (NRC) issued the Policy Statement on Severe Accidents [1]. This Policy Statement sets the goals and schedule for addressing issues relevant to severe accidents in the licensing of future plants and for the systematic examination of existing plants.

The NRC Severe Accident Policy Statement does not differentiate between events initiated within the power plant and events initiated externally, such as earthquakes, floods, and high winds. The evaluation of internally initiated events is more developed than the evaluation of externally initiated events. Therefore, the NRC staff is currently implementing the Severe Accident Policy Statement for internally initiated events. The implementation plans are stated in NRC internal papers, SECY 86-162 [2] and SECY 86-76 [3].

The evaluation of severe accidents initiated by external events is proceeding in two phases [2]. The first phase addresses the assessment of the margin provided by past and on-going programs relative to external events beyond the design basis. In addition, the first phase identifies areas to examine for external vulnerabilities. The second phase will consist of a program for plant specific evaluation, if needed. Information developed in phase one will be used as guidance for phase two.

As part of the implementation program supporting the NRC Policy Statement on Severe Accidents [2], the Lawrence Livermore National Laboratory (LLNL) has performed a study of the risk of core damage to nuclear power plants in the United States caused by internal fire, external floods, high winds and transportation accidents [4]. The broad objective of that study for existing U.S. light water reactor (LWR) power plants, was to understand whether or not these external hazards are major potential accident initiators that may pose a threat of severe reactor core damage or of a large radioactive release to the environment. This report addresses this concern for the seismic external hazard and supplements the earlier report [4]. A study to assess the remaining "other" external initiators is presently underway.

1.2 WORK REQUIREMENTS

The overall objective of this report is to present information that assists the NRC staff in deciding whether seismic vulnerability searches for nuclear power plants should be included in the implementation of the Severe Accident Policy Statement. To accomplish this objective, this report:

1. Considers effects of the evolution of design requirements and design practices on plant seismic capacity.

2. Identifies other specific review areas of potential seismic vulnerability, including seismically induced fires and floods, spent fuel pools, and seismic common-mode failures.

3. Identifies programs which address items 1 and/or 2 above, and assess the extent to which these programs provide useful information on seismic capacity of nuclear plants.

4. Recommends incorporating appropriate items from above into the seismic margins program or other seismic vulnerability searches.

The remainder of this chapter discusses evaluation criteria used to indicate whether seismic events need to be considered as part of the severe accident policy implementation.

1.3 EVALUATION CRITERIA

The approach used in this study is the same used in the Ref. 1 but is directed at seismically initiated nuclear power plant accidents which lead to scenarios resulting in damage to the reactor core or to the release of radioactive material.

In order to examine the risk to U.S. nuclear power plants from any specific externally initiated event, specific evaluation criteria were employed to discriminate between significant and less significant levels of risk. These evaluation criteria used the guidance provided by the NRC in their Safety Goal Policy Statement [5] and Policy Statement on Severe Accidents [1]. Two different figures-of-merit were used as evaluation criteria.

There is no implication here that individual plants that are currently licensed to operate or authorized for construction must meet these figures-of-merit. These two figures-of-merit were used solely for the purpose of screening externally initiated events to determine whether they should be considered further as part of the severe accident implementation. These figures-of-merit are met if corresponding values are comparable to these criteria.

The first figure-of-merit is core damage frequency. According to the NRC's Policy Statement on Safety Goals, the Commissioners explicitly stated one of their objectives as:

> "providing reasonable assurance, giving consideration to the uncertainties involved, that a core-damage accident will not occur at a U.S. nuclear power plant."[5]

In numerical terms (based on about 100 nuclear power plants operating for about 40 years), this objective can be met if individual plants have mean core damage frequencies in the range of about 1 E-5 or less per reactor year. This is not a firm numerical objective, but a range whose method of application by the NRC staff will continue to evolve over the next few years.

The second figure-of-merit is the frequency of a large release of radioactive material. In this same NRC Policy Statement on Safety Goals, the following general performance guideline was given:

> "Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."[5]

The NRC staff is considering how such a performance guideline can be implemented, including how to define more precisely the definition of "a large release of radioactive material to the environment." For the purposes of this report, a large release of radioactive material to the environment has been defined as a release of a substantial fraction of the radioactive core in a time period relatively early in the postulated accident scenario. This definition has been taken from Probabilistic Risk Assessment (PRA) literature which has defined a "large early release." [11,13,16,18,20,21,49] This imprecise definition includes those accident sequences whose release of significant fractions of the reactor core radioactive inventory occur within a few hours after the initiation of the accident. Operationally, this means that for the purposes of this report, a "large radioactive release" will correspond to those few PRA plant damage states and release categories associated with "large, early release" and such a release should occur with a frequency equal to or less than 1 x E-6 per reactor year.

Using these two figures-of-merit as evaluation criteria, it has been feasible to ascertain whether the risk from seismically initiated accidents is, or is not, an important contributor to overall risk for the U.S. nuclear power plants studied. If the core damage and/or radioactivity release values compare to these figures-of-merit on an order of magnitude basis, these criteria have been met and indicates that seismic initiated accidents need to be considered in the Severe Accident Policy Implementation.

Because PRA methods are the most widely known and widely used approach for calculating the expected frequency of reactor core damage or of large radioactive release from the reactor core per year of operation, this report emphasizes insights from various PRA studies. Large numerical uncertainties are associated with these PRA analyses, and therefore, the evaluation has been based on a broad (order-of-magnitude) type of comparison. Because the evaluation criteria are broadly based and not a direct numerical comparison of PRA results, it is possible to obtain insights into the issues surrounding the risk to U.S. nuclear power plants from seismic events, even taking account of the large uncertainties in the PRA results.

1.4 REPORT ORGANIZATION

This report is organized into five chapters. Chapter 2 discusses past and current seismic design requirements and practices. Chapter 3 discusses reviews of plant seismic analyses including seismic PRAs. Chapter 4 discusses past and on-going seismic programs and their applicability to implementation of the severe accident policy. Chapter 5 gives recommendations and conclusions regarding the need to include seismic hazards in the implementation of the severe accident policy.

CHAPTER 2

SEISMIC DESIGN REQUIREMENTS AND PRACTICES

2.1 INTRODUCTION

This chapter discusses past and current seismic design requirements and practices. Seismic events are here defined as earthquakes that occur at or near the nuclear power plant site.

2.2 NRC REGULATORY REQUIREMENTS

When the U.S. Atomic Energy Commission (AEC now the NRC) regulatory staff first began to review nuclear power plant designs, its scope of review was less defined than it is now. The requirements for acceptability evolved as new facilities were reviewed. The primary regulatory requirements for Nuclear Power Plants are given in Title 10 of the Code of Federal Regulations (10CFR).

In 1971, the General Design Criteria (GDC) for Nuclear Power Plants were formally adopted as the minimum requirements for the principal design standards. The GDC has been used as guidance in reviewing new plant applications since then. Safety guides issued in 1970 became part of the Regulatory Guide Series in 1972. These guides describe methods acceptable to the regulatory staff for implementing specific portions of the regulations, including certain GDC's. They formalize staff techniques for performing a facility review. In 1972, the NRC released the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, now known as Regulatory Guide 1.70. It provides a standard format for these reports and identifies the principal information needed by the staff for its review. The Standard Review Plan (SRP, NUREG-75/087) was published in December 1975 and updated in July 1981 (NUREG-0800) to provide further guidance for improving the quality and uniformity of staff reviews, to enhance communication and understanding of the review process by interested members of the public and the nuclear power industry, and to stabilize the licensing process.

Because of the evolutionary nature of licensing requirements and the development of technology over the years, nuclear power plants employ a broad spectrum of design features and requirements depending on when the plant was designed and constructed, who was the manufacturer, and when the plant was licensed for operation. The amount of documentation that defines these safety-design characteristics has also changed with the age of the plant. Older plants tend to have less documentation, are designed to less stringent criteria, have less design margins and potentially greater differences than plants designed to current licensing criteria.

2.2.1 Title 10 of the Code of Federal Regulations

Title 10 of the Code of Federal Regulations (10 CFR) [6] specifies in general terms, the conditions and factors that must be considered in constructing, licensing and operating a nuclear power plant and the regulatory process that must be followed. Specific parts of 10 CFR consider earthquakes:

10 CFR Part 50.55a Codes and Standards

10 CFR Part 50 Appendix A

General Design Criteria for Nuclear Power Plants

Criterion 1 - Quality Standards and Records

Criterion 2 - Design Bases for Protection against Natural Phenomena

Criterion 44 - Cooling Water

10 CFR Part 50 Appendix B

Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

10 CFR Part 100 Reactor Site Criteria

10 CFR Part 100 Appendix A

Seismic and Geologic Siting Criteria for Nuclear Power Plants.

These regulations provide two earthquake definitions that are used to establish the seismic design of a nuclear power plant.

10 CFR PART 100 Appendix A, III.(d) [6] defines an

OPERATING BASIS EARTHQUAKE (OBE) as

"that earthquake which, considering the regional and local geology, seismology, and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant; it is that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional."

10 CFR PART 100 Appendix A, III.(c) [6] defines a

SAFE SHUTDOWN EARTHQUAKE (SSE) as

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

The safe shutdown earthquake in the past has been referred to as the design bases earthquake (DBE).

2.2.2 Regulatory Guides

The specific guidance issued by the NRC staff to help industry in designing nuclear power plants for protection against earthquakes is given by documents known as NRC Regulatory Guides. The specific NRC Regulatory Guides for earthquakes are listed below:

Regulatory Guide 1.12 - Instrumentation for Earthquakes

Regulatory Guide 1.29 - Seismic Design Classification

Regulatory Guide 1.48 - Design Limits and Loading Combinations for Seismic Category I Fluid System Components

Regulatory Guide 1.55 - Concrete Placement in Category I Structures,

Regulatory Guide 1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

Regulatory Guide 1.61 - Damping Values for Seismic Design of Nuclear Power Plants

Regulatory Guide 1.92 - Combining Modal Response and Spatial Components in Seismic Response Analysis

Regulatory Guide 1.100 - Seismic Qualification of Electric Equipment for Nuclear Power Plants

Regulatory Guide 1.122 - Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.

2.2.3 Standard Review Plan

The NRC Standard Review Plan (SRP) [7] guides the NRC staff in their review of Preliminary Safety Analysis Reports (PSARs) and Final Safety Analysis Reports (FSARs) submitted by an applicant. The following SRP subsections call for earthquakes to be considered when reviewing safety analysis reports (SARs) for nuclear power plants:

SRP No. 2.5.1 Basic Geologic and Seismic Information

SRP No. 2.5.2 Vibratory Ground Motion

SRP No. 2.5.3 Surface Faulting

SRP No. 2.5.4 Stability of Subsurface Materials and Foundations

SRP No. 2.5.5 Stability of Slopes

SRP No. 3.7.1 Seismic Design Parameters

SRP No. 3.7.2 Seismic System Analysis

SRP No. 3.7.3 Seismic Subsystem Analysis

SRP No. 3.9.1 Computer Codes used in analysis of Seismic Category I systems and supports

SRP No. 3.9.2 Seismic analysis for all Category I systems, components, equipment and supports

SRP No. 3.9.3 Load combinations and stress units for Seismic Category I Systems and components using ASME Code.

SRP No. 3.10 Qualification of Equipment in Licensing Plants.

2.3 SEISMIC DESIGN PRACTICES

The following standards and codes have been used for nuclear power plant design:

American Concrete Institute (ACI) Standards

American Institute of Steel Construction (AISC) Standards

American National Standards Institute (ANSI) Standards

Institute of Electrical and Electronic Engineers (IEEE) Standards

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Codes

Uniform Building Codes (UBCs)

American Welding Society

American Society of Civil Engineers (ASCE) Manuals and Standards

Department of the Navy, Design of Protective Structures

Atomic Energy Commission Documents, Nuclear Power Plants and Earthquakes (TID 7024).

Army Corp. Of Engineers Design Manuals.

Not all of these standards and codes are currently being used.

Many parameters are considered during the design of nuclear power plants [8]. The methods used to quantify these parameters have changed over the years partly based on changes in the seismic design standards and codes as a result of past earthquakes. The primary design and analysis parameters for nuclear power plants appear below:

Safe Shutdown Earthquake (SSE)	-	defined by 10 CFR 100 App. A, III.(c)
Operating Basis Earthquake (OBE)	-	in many plants OBE = 1/2 SSE. Some plants use OBE = 1/3 SSE. For many plants the vertical component of ground motion was taken as 2/3 of the horizontal component.
Ground Design Spectra	-	representative shape anchored to some local magnitude, OR synthetic shape developed from a combination of earthquake time histories.
Response Spectrum Analysis	-	Modal Combinations, absolute sum, SRSS, algebraic sum, R.G. 1.92 method.
Type of Design Spectra	-	Housner, Reg. Guide 1.60 (Newmark), NUREG-0098 (Newmark/Hall), Others.
Generation of Floor Response	-	time histories, direct spectral generation.
Foundation/Liquifaction	-	foundation type, bearing information, shear wave velocity profile, groundwater, rock/soil description, nearby dams.
Soil Structure Interaction (SSI)	-	modal methods, conventional stick model, lumped spring/finite element (soil), fixed base method

		(rock), soil shear modulus profile, material damping of soil, limit on modal damping.
Category I Structures	-	structures, mechanical & electrical components important to safety, damping value (OBE/SSE), design criteria, load combinations, allowable stress, acceptance criteria, method of quantification (testing, analytical).

2.4 MATERIAL REVIEWED FOR THIS STUDY

The following technical material was reviewed for this project:

1) The NRC's regulatory approach to assuring that nuclear power plants are adequately protected against the seismic external initiator, i.e., Title 10 of the Code of Federal Regulations (10 CFR), other applicable regulations, the standard review plan (SRP), and applicable regulatory guides (RGs).

2) Technical papers and reports, including discussions of PRA methodology for the seismic initiator, technical studies on associated phenomena, and studies of the traditional engineering approaches to assuring protection against seismic events.

3) Documentation on how nuclear power plants are designed and constructed by their owners and reviewed by the NRC staff for protection from earthquakes, i.e. FSARs, PSARs, SARs, and Safety Evaluation Reports (SERs).

4) PRA literature on seismic initiators, including both full-scope PRA studies and partial PRA-type analyses.

5) Event data bases at nuclear power plants, including both the larger events of interest and the smaller events that did not cause serious problems.

6) NRC's current research programs, such as the Systematic Evaluation Program, the SSMRP, Seismic Design Margins Program (SDMP) and Seismic Qualification of Equipment at Nuclear Power Plants (A-46).

Important aspects of each category have been examined, although not all of this material has been studied in detail.

CHAPTER 3

REVIEW OF SEISMIC ANALYSES

3.1 INTRODUCTION

This chapter reviews and evaluates what is known about the risk of core damage accidents and the potential for large radioactive release to the environment caused by seismic events at U.S. nuclear power plants. The broad objective of this review and evaluation is to understand, for U.S. light-water reactors, whether or not seismic events are among the major accident initiators that may pose a threat of severe core damage or large radioactive release. The project scope was limited to an examination of a few specific plants whose seismic response and behavior has been studied in greater detail.

The evaluation criteria, which consist of two figures-of-merit, are compared to the results of seismic analyses of nuclear power plants. The first figure-of-merit is met if an individual plant has a mean core damage frequency resulting from a seismic initiating event in the range of about one part in 100,000 per reactor year. This is not a firm numerical objective, but a range. The second figure-of-merit is met if a "large release" is calculated to occur as the result of a seismic initiating event with a mean frequency in the range of 1 in 1,000,000 per reactor year.

Postulated reactor accidents caused by seismic events can be characterized as having two parts: (1) the seismic initiating event and (2) the plant response to the seismic event. Unlike internal initiating events, the occurrence of the seismic initiator is completely independent of the existence of the plant. Seismic events will occur whether the plant exists or not.

Another characteristic of the seismic initiating event is the common-mode effect on the plant systems and components. Earthquake motion will be felt by all plant components simultaneously, resulting in some correlation between their seismic responses.

3.2 SEISMIC ANALYSES REVIEWED

To study the degree of protection that has been achieved in the seismic safety of nuclear power plants, the approach has been to use seismic PRA literature, supplemented with the results of the Seismic Safety Margins Research Program (SSMRP), the analyses performed under the Task Action Plan on Decay Heat Removal Requirement (TAP A-45), and recent seismic hazard studies at nuclear power plant sites.

The above seismic analysis literature represents the best "realistic" set of analyses of the estimated response and behavior of nuclear power reactors during and following earthquakes. In addition, seismic PRAs and other probabilistic seismic analyses provide probabilistic values of core damage and "large" radioactive release for comparison with the two figures-of-merit.

The following assumptions were made following the literature review:

- The methodology used in the seismic PRAs is assumed valid, that is, it provides an accurate determination of the core damage frequency from this initiating event, within the quoted uncertainty range.

- All of the seismic PRA analyses are assumed to have been performed using a similar methodology, so their results can be compared on a common basis.

- The methodology used for the TAP A-45 and SSMRP seismic analyses is also valid and provides an accurate determination of the core damage frequency from the seismic initiating event within the limitations given.

These assumptions are not completely valid for the following reasons:

- The earlier seismic PRAs in general were not as comprehensive as the more recent studies.

- Some of the reactors were modified (backfits) during or after the PRA study.

The review and evaluation of seismic analyses of nuclear power plants was performed for the two characteristic parts of a seismic accident sequence: the seismic initiating event and the plant response. The review and evaluation of the seismic initiating event, i.e. site-specific seismic hazard, is given in Section 3.3. The review and evaluation of the plant response is given in Section 3.4.

3.3 SEISMIC HAZARD REVIEW

Because of the change in perception of the Charleston Earthquake, the NRC initiated a program to characterize the seismic hazard at eastern reactor sites. The initial thrust of this program was to develop a methodology for assessing the site specific seismic hazard at particular plant sites. The results of this program are reported in "Seismic Characterization of the Eastern United States, Volume 1: Methodology and Results for Ten Sites," UCID-20421, April 1985 [9]. The on-going effort is a characterization of the seismic hazard for all eastern (east of the Rocky Mountains) reactor sites.

The Electric Power Research Institute (EPRI) has a parallel program employing a similar methodology. That study also assessed plant site-specific seismic hazards for the same ten sites as the NRC program. Reference 10 gives a comparative evaluation of the results between the NRC and the EPRI studies.

The evaluation of seismic hazards at particular plant sites was performed to estimate the annual occurrence frequency of the initiating event. Table 3-1 shows probability-of-exceedance values along with their 85% and 15% confidence limits for both the NRC and EPRI studies.

Table 3-1 indicates that a majority of the plant sites have a median annual probability of occurrence at the SSE level earthquake greater than E-4. The NRC study indicates that eight of the nine plants fall into this category, while the EPRI study indicates that six of the nine plants are in this category. For the EPRI study, three sites were found to have a median annual occurrence frequency at the SSE in the range of E-5.

The values stated above are median values. An approximate indication of the mean annual occurrence frequency at the SSE can be obtained at the 85% confidence level.

The results indicate that the median occurrence frequency of seismic initiating events for which plants are designed (SSE) is generally in the range of E-4 per reactor year, and that an approximate mean occurrence frequency (indicated by the 85th percentile) is in the range of about E-3 per occurrence. Comparing these values with the core damage evaluation criterion indicates that the plant response should provide a level of protection in the range of 0.01 to 0.001 for the probability of core damage contingent on experiencing an earthquake at the SSE level.

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Plant Site	SSE (g)	NRC (median)	85%/15%*	EPRI (median)	85%/15%	REF. 10 Figure No.
		Annual Probability of Exceedance at SSE from [10]				
Braidwood Units 1 & 2	0.20	7.E-5	3.E-4/1.E-5	9.E-5	1.E-4/2.E-5	2.3.1
Limerick Units 1 & 2	0.15	3.E-4	2.E-3/8.E-5	4.E-4	1.E-3/2.E-4	2.3.2
Maine Yankee	0.10	2.E-3	5.E-3/4.E-4	1.E-3	4.E-3/7.E-4	2.3.3
Millstone Units 1, 2 & 3	0.17	3.E-4	2.E-3/7.E-5	3.E-4	4.E-4/3.E-5	2.3.4
River Bend Unit 1	0.10	1.E-4	6.E-4/8.E-6	4.E-5	1.E-4/2.E-7	2.3.5
Shearon Harris Unit 1	0.15	2.E-4	1.E-3/7.E-5	4.E-5	1.E-4/1.E-5	2.3.6
Vogtle Unit 1	0.20	4.E-4	2.E-3/7.E-5	1.E-4	4.E-4/2.E-5	2.3.7
Watts Bar Units 1 & 2	0.18	4.E-4	2.E-3/1.E-4	2.E-4	4.E-4/9.E-5	2.3.8
Wolf Creek Unit 1	0.12	1.E-4	6.E-4/1.E-5	1.E-4	3.E-4/5.E-5	2.3.9

Note: * These values were obtained from figures that were generated using only one ground motion model (Nuttli, 1984), no site correction and considering only contributions of earthquakes greater than magnitude 5. The EPRI values are draft interim results for comparison purposes.

CHAPTER 3

REVIEW OF SEISMIC ANALYSES

3.1 INTRODUCTION

This chapter reviews and evaluates what is known about the risk of core damage accidents and the potential for large radioactive release to the environment caused by seismic events at U.S. nuclear power plants. The broad objective of this review and evaluation is to understand, for U.S. light-water reactors, whether or not seismic events are among the major accident initiators that may pose a threat of severe core damage or large radioactive release. The project scope was limited to an examination of a few specific plants whose seismic response and behavior has been studied in greater detail.

The evaluation criteria, which consist of two figures-of-merit, are compared to the results of seismic analyses of nuclear power plants. The first figure-of-merit is met if an individual plant has a mean core damage frequency resulting from a seismic initiating event in the range of about one part in 100,000 per reactor year. This is not a firm numerical objective, but a range. The second figure-of-merit is met if a "large release" is calculated to occur as the result of a seismic initiating event with a mean frequency in the range of 1 in 1,000,000 per reactor year.

Postulated reactor accidents caused by seismic events can be characterized as having two parts: (1) the seismic initiating event and (2) the plant response to the seismic event. Unlike internal initiating events, the occurrence of the seismic initiator is completely independent of the existence of the plant. Seismic events will occur whether the plant exists or not.

Another characteristic of the seismic initiating event is the common-mode effect on the plant systems and components. Earthquake motion will be felt by all plant components simultaneously, resulting in some correlation between their seismic responses.

3.2 SEISMIC ANALYSES REVIEWED

To study the degree of protection that has been achieved in the seismic safety of nuclear power plants, the approach has been to use seismic PRA literature, supplemented with the results of the Seismic Safety Margins Research Program (SSMRP), the analyses performed under the Task Action Plan on Decay Heat Removal Requirement (TAP A-45), and recent seismic hazard studies at nuclear power plant sites.

The above seismic analysis literature represents the best "realistic" set of analyses of the estimated response and behavior of nuclear power reactors during and following earthquakes. In addition, seismic PRAs and other probabilistic seismic analyses provide probabilistic values of core damage and "large" radioactive release for comparison with the two figures-of-merit.

The following assumptions were made following the literature review:

- o The methodology used in the seismic PRAs is assumed valid, that is, it provides an accurate determination of the core damage frequency from this initiating event, within the quoted uncertainty range.
- o All of the seismic PRA analyses are assumed to have been performed using a similar methodology, so their results can be compared on a common basis.
- o The methodology used for the TAP A-45 and SSMRP seismic analyses is also valid and provides an accurate determination of the core damage frequency from the seismic initiating event within the limitations given.

These assumptions are not completely valid for the following reasons:

- o The earlier seismic PRAs in general were not as comprehensive as the more recent studies.
- o Some of the reactors were modified (backfits) during or after the PRA study.

The review and evaluation of seismic analyses of nuclear power plants was performed for the two characteristic parts of a seismic accident sequence: the seismic initiating event and the plant response. The review and evaluation of the seismic initiating event, i.e. site-specific seismic hazard, is given in Section 3.3. The review and evaluation of the plant response is given in Section 3.4.

3.3 SEISMIC HAZARD REVIEW

Because of the change in perception of the Charleston Earthquake, the NRC initiated a program to characterize the seismic hazard at eastern reactor sites. The initial thrust of this program was to develop a methodology for assessing the site specific seismic hazard at particular plant sites. The results of this program are reported in "Seismic Characterization of the Eastern United States, Volume 1: Methodology and Results for Ten Sites," UCID-20421, April 1985 [9]. The on-going effort is a characterization of the seismic hazard for all eastern (east of the Rocky Mountains) reactor sites.

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The results indicate that the median occurrence frequency of seismic initiating events for which plants are designed (SSE) is generally in the range of E-4 per reactor year, and that an approximate mean occurrence frequency (indicated by the 85th percentile) is in the range of about E-3 per occurrence. Comparing these values with the core damage evaluation criterion indicates that the plant response should provide a level of protection in the range of 0.01 to 0.001 for the probability of core damage contingent on experiencing an earthquake at the SSE level.

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Maine Yankee	0.10	2.E-3	5.E-3/4.E-4	1.E-3	4.E-3/7.E-4	2.3.3
Millstone Units 1, 2 & 3	0.17	3.E-4	2.E-3/7.E-5	3.E-4	4.E-4/3.E-5	2.3.4
River Bend Unit 1	0.10	1.E-4	6.E-4/8.E-6	4.E-5	1.E-4/2.E-7	2.3.5
Shearon Harris Unit 1	0.15	2.E-4	1.E-3/7.E-5	4.E-5	1.E-4/1.E-5	2.3.6
Vogtle Unit 1	0.20	4.E-4	2.E-3/7.E-5	1.E-4	4.E-4/2.E-5	2.3.7
Watts Bar Units 1 & 2	0.18	4.E-4	2.E-3/1.E-4	2.E-4	4.E-4/9.E-5	2.3.8
Wolf Creek Unit 1	0.12	1.E-4	6.E-4/1.E-5	1.E-4	3.E-4/5.E-5	2.3.9

Note: * These values were obtained from figures that were generated using only one ground motion model (Nuttli, 1984), no site correction and considering only contributions of earthquakes greater than magnitude 5. The EPRI values are draft interim results for comparison purposes.

3.4 PLANT RESPONSE REVIEW

Probabilistic seismic analyses of nuclear power plants have been reviewed to understand the behavior and response of these plants to earthquakes. addition, the probabilistic results for seismic core damage and "large" radioactivity release can be compared to the figures-of-merit to indicate whether accidents initiated by seismic events need to be considered in the severe accident policy implementation.

The probabilistic results of these seismic analyses have been given as mean values, median values, and point estimates of core damage and radioactive release. To better understand the results it is necessary to understand the definition of these three values.

The mean and median values are both considered measures of central tendency, that is, a measure of the center of a distribution. The expected value, more frequently call the arithmetic mean or the mean, may be regarded as the center of gravity of a distribution, since it is that point around which the sum of the distribution to the left times the probability weight exactly balances out the corresponding sum of weighted values to the right. For a simple discrete example, let's say you have five data points. The mean of that data is the sum of those five data points divided by five.

Another measure of central tendency is the mid-point or median of a distribution. The median is that value that has exactly one half of the area under the probability density function to its left and one half to its right. For a simple discrete example, let's say, as above, that you have five data points. The median of that data is the third data point given that you ordered them by either increasing or decreasing value. Note that the mean is sensitive to extreme observations while the median is less affected.

A third measure that is used in seismic analyses is the point estimate. Point estimates are obtained by performing the analysis using mean values. For example, in an SSMRP type of analysis, distributions are used for both the component response and fragility. Using the concept of stress-strength interference a probability of component failure is generated. This probability is a point estimate for that component failing. Component point estimates are then combined discretely and the result is point estimates on core damage or radioactive release.

In the Safety Goal Policy Statement [5] and the Policy Statement on Severe Accidents [1], the objectives correspond to mean core damage frequencies for individual plants and stated an overall mean frequency of radioactivity release. Corresponding to these values, the criterion developed for this study represent mean values. However, these criterion were used solely for the purpose of screening. If either of these criterion are met, this indicates that seismically initiated accident should be considered further as part of the severe accident policy implementation.

The review of plant response to the seismic initiating event has been performed for seven published seismic PRAs, five TAP A-45 plants and two SSMRP analyses. In addition, a review of the seismic behavior of a typical BWR spent fuel pool is also included since these facilities have a potential for radioactivity release. The following sections give the results of these reviews along with insights into the seismic behavior of nuclear power plants.

3.4.1 SSMRP Analyses

The Seismic Safety Margins Research Program (SSMRP) was established in 1978 to develop mathematical models that realistically predict the probability of core damage and radioactivity release from seismic initiating events in nuclear power plants [11]. The methods and techniques developed in this program were used to perform a seismic probabilistic analysis of the Zion Nuclear Power plant. These method and techniques are briefly discussed in the next Chapter.

Subsequent to the initial SSMRP development effort, a simplified method was derived to more easily utilize the SSMRP methodology [12]. This simplified SSMRP method was used in a seismic probabilistic analysis of the LaSalle County Unit 2 Nuclear Power Plant [13].

Table 3-2 shows the results of seismic analyses of Zion and LaSalle. This table gives point estimate core damage frequencies for seismic initiated events along with an indication of the percentage of the seismic core damage frequency for two earthquake ranges.

An uncertainty analysis [11] was performed for Zion and resulted in mean and median core damage frequency estimates of 2.E-4 per year and 3.E-5 per year, respectively. No uncertainty analysis for LaSalle was performed.

Table 3-2 indicates that neither plant has a point estimate seismic core damage frequency that is comparable to the core damage figure-of-merit. However, the mean and median core damage frequencies for Zion are comparable to this figure-of-merit. The dominant earthquake range for seismic core damage is between 0.2-0.4g.

3.4.2 Seismic PRAs

The published seismic PRAs that were reviewed for this study are for Zion 1 & 2 [14,15], Indian Point 2 & 3 [16,17], Limerick 1 & 2 [18,19,20], Millstone 3 [21,22], Seabrook [23] and Oconee 3 [24]. General PRA methodology was used in the performance of these analyses which were sponsored by the utilities and performed between 1981 and 1985.

The overall results of these PRAs are given on Table 3-3. This table shows the mean core damage and release frequencies for seismically initiated plant damage states along with an indication of how these seismically

TABLE 3-2

SEISMIC CORE DAMAGE FREQUENCIES
FROM THE SSMRP PLANTS

PLANT	TYPE	SSE (g)	SEISMIC CORE DAMAGE Per Year (point estimate)	DOMINANT EARTHQUAKE RANGE (g)	SEISMIC AS PERCENT OF TOTAL CORE DAMAGE FREQUENCY
ZION 1 & 2	PWR	0.17	3.6E-6 (Mean-2.E-4) (Median-3.E-5)	0.20-0.32 0.32-0.42	39% 44%
LaSalle 2	BWR	0.20	6.0E-7	0.18-0.27 0.27-0.36	57% 22%

TABLE 3-3

SEISMIC CORE DAMAGE AND RELEASE FREQUENCIES
FROM PUBLISHED PROBABILISTIC RISK ASSESSMENTS

PLANT	TYPE	SSE (g)	SEISMIC CORE DAMAGE FREQUENCY (mean) Per Year	SEISMIC RELEASE FREQUENCY (mean) Per Year	% OF TOTAL CORE DAMAGE	RANK OF RELEASE SEQUENCE	DOMINANT EARTHQUAKE LEVEL (g)
Zion 1 & 2	PWR	0.17	5.6E-6	--------	3	1	>0.35
Indian Point 2	PWR	0.15	1.4E-4 (rev. 4.8E-5)	1.4E-4	30	1	>0.30
Indian Point 3	PWR	0.15	3.1E-6 (rev. 2.5E-5)	2.4E-6	1	8	>0.30
Limerick	BWR	0.15	4.0E-6	2.0E-7	----	1	>0.35
Millstone 3	PWR	0.17	9.4E-5	-----	68	3	>0.30
Seabrook	PWR	0.25	2.9E-5	-----	13	30	>0.30
Oconee 3	PWR	0.15	6.3E-5	6.0E-5	25	1	>0.15

initiated damage states compare with the overall core damage and release frequencies from all initiating events.

Table 3-3 indicates that four of the seven plants have a core damage frequency comparable to the core damage figure-of-merit of E-5 per year and meet this evaluation criterion. Three of the four plants that provided a seismic release frequency were also comparable to the release figure-of-merit of E-6 per year.

Uncertainty ranges in core damage and release frequency estimates from PRA analyses are generally an order of magnitude or greater. When accounting for these uncertainty ranges, the estimated ranges for core damage and large release frequencies bracket the figures-of-merit for most plants examined in the PRA studies.

Other insights gained from this table indicate that the contribution to the total core damage from seismically initiated events ranges from 1 to 68 percent, and the dominant earthquake level for seismic core damage is generally greater than 0.3g. In addition, the rank of the seismic initiated sequences with respect to radioactive release ranges from first to thirtieth. Four of these seismic initiated sequences are ranked first with respect to radioactivity release.

In conclusion, this review of published industry-sponsored PRAs indicates that there is no basis for excluding accidents initiated by seismic events from consideration in the Severe Accident Policy implementation.

3.4.3 TAP A-45 Analyses

The Task Action Plan A-45, Decay Heat Removal Requirements at nuclear power plants was performed to provide a technical basis for resolution of the associated unresolved safety issues. As part of this program, probabilistic risk assessments were performed for five nuclear power plants. These risk assessments focus on the plant's response to small loss-of-coolant accidents (LOCAs) and transient events which require the various decay heat removal systems to bring the plant to cold shutdown.

As part of this risk assessment program, simplified seismic risk assessments were performed for St Lucie Unit 1 [25], Quad Cities Units 1 & 2 [26], Point Beach Units 1 & 2 [27], Arkansas Nuclear One Unit 1 [28], and Turkey Point Units 1 & 2 [29]. The simplifications utilized in these analyses were derived from the extensive results of the NRC-sponsored SSMRP [11]. These analyses all employed the SSMRP methodology.

Simplifications to the analysis included the use of reduced systems models, generic component fragility information, and the characterization of the seismic hazard at the particular plant site. In addition, failure of the reactor protection system was not in the scope of the TAP A-45 analysis. Therefore, the analysis did not consider anticipated transients without scram.

The seismic hazard used for these analyses was characterized by one of several methods:

1) a site specific seismic hazard curve for the plant,

2) an existing PRA of a nearby plant,

or if no curve exists,

3) a hazard curve scaled to an exceedance probability of 2.5E-4/ year at the SSE. The slope of the curve for higher peak ground acceleration values was estimated from hazard curves for other sites in the same broad seismological province.

The SSMRP methodology allows the analysis to be divided into a number of earthquake intervals with risk estimate being calculated for each intervals.

The results of the TAP A-45 seismic analyses for the five plants are given in Table 3-4. This table gives point estimate seismic core damage and release frequencies along with a percentage of the seismic core damage frequency for two earthquake intervals. In comparison to the SSMRP analysis, presented in a previous section, mean and median values for core damage appear to be larger than point estimates.

Table 3-4 indicates that the core damage frequencies are consistent and that all of these frequencies meet the figure-of-merit for core damage frequency. All five plants have a core damage frequency within the E-5 probability decade. In addition, each plant for which a release frequency was given also meets the figure-of-merit for the large release. Four of the five plants have a point estimate release frequency in the range of E-6 or greater.

Inspection of the percentage of the core damage frequency within each earthquake interval indicates that the dominant contribution to core damage is from earthquakes in the range of 0.2-0.4g.

The consistency in the risk estimates for these analyses may be due to the manner in which the seismic hazard is characterized. While the shape of hazard curves may vary considerably, the magnitude of the hazard at the SSE will be consistent among the analyses. This fact coupled with the fact that the core damage frequency is dominated by earthquakes of lower magnitude (closer to the SSE) may result in the consistency of risk estimates. However, the simplified approach to plant modeling and response analysis may have also contributed to the consistency in the results.

In conclusion, the results of the Decay Heat Removal Requirement TAP A-45 analyses indicates that the two figures-of-merit have been met but not with large margins, and that accidents initiated by seismic events should be considered in the implementation of the severe accident policy.

TABLE 3-4

SEISMIC CORE DAMAGE AND RELEASE FREQUENCIES
FROM THE DECAY HEAT REMOVAL REQUIREMENTS TAP A-45 PLANTS

PLANT	TYPE	SSE (g)	SEISMIC CORE DAMAGE FREQUENCY (point estimate) Per Year	SEISMIC RELEASE FREQUENCY (point estimate) Per Year	DOMINANT EARTHQUAKE RANGE (g)	SEISMIC AS PERCENT OF TOTAL CORE DAMAGE FREQUENCY
Point Beach 1 & 2	PWR	0.12	6.0E-5	2.5E-5	0.12-0.24	49%
					0.24-0.36	38%
St. Lucie 1	PWR	0.10	1.3E-5	5.8E-6	0.20-0.30	52%
					0.30-0.40	39%
Quad Cities 1 & 2	BWR	0.24	8.3E-5	-----	0.24-0.48	62%
					0.48-0.72	23%
Arkansas Nuclear One 1	PWR	0.20	7.3E-5	3.7E-5	0.20-0.40	55%
					0.40-0.60	25%
Turkey Point	PWR	0.15	1.0E-5	4.6E-6	0.15-0.30	44%
					0.30-0.45	39%

3.4.4 Spent Fuel Pool Analyses

With increasing amounts of spent fuel being stored at nuclear power plants, there may exist a significant potential for severe accidents involving spent fuel stored in spent fuel pools. While the radioactive material inventory of the spent fuel will be significantly less than an active core, there is a potential for a severe accident in a spent fuel pool to involve the equivalent of several cores of fuel bundles.

The major concern is for the spent fuel pool to lose its cooling capability by loss of its cooling water inventory. Without sufficient cooling, the zircaloy cladding material of fuel rods can initiate and sustain a rapid oxidation (fire) that can spread to nearby fuel rods with the potential of releasing significant amounts of long lived radioactive isotopes.

This section presents the results of a review of a spent fuel pool response to seismic events. The literature reviewed was a probabilistic analysis of spent fuel pool behavior that was performed in response to the Generic Safety Issue 82 "Beyond Design Basis Accident in Spent Fuel Pools" [28].

The analysis presented in [47] used data and information from several plant sites including Ginna, Millstone, and Oyster Creek. The results of this analysis indicate a large range of values for seismically induced spent fuel pool failure. For a pressurized water reactor (PWR) plant, the annual frequency of seismically induced spent fuel pool failure was given as 2.6E-4 to 1.6E-11. For a boiling water reactor (BWR) plant this range was given as 6.5E-5 to 4.5E-11 per year.

The reason for such a large range in these values was attributed to the uncertainty in the seismic hazard and the fragility of spent fuel pool structures and components. These results do not represent any specific plant but were an attempt to gain a generic understanding of the probability and severity of spent fuel pool failures.

A comparison of the results of the fuel pool analysis with the two figures-of-merit is difficult since fuel pool failure does not constitute core damage and any potential release only involves long lived radioactive material. In addition, it is difficult to draw conclusions concerning spent fuel pools based on only a single generic analysis. Therefore, any decision on the inclusion of spent fuel pools into the severe accident policy implementation requires more data and analysis, and cannot be concluded at this time.

3.5 INSIGHT FROM SEISMIC PROBABILISTIC ANALYSES

There are a number of overall insights gained from the review of the probabilistic seismic assessment performed in this study. This section presents a brief discussion of these overall insights. The insights presented are taken largely from earlier reviews [30]. The dominant accident initiator at nuclear power plants for seismic events is loss-of-offsite power (LOOP) caused by the low capacity of switchyard components, in particular, ceramic insulators.

For seismically initiated events, the main contributors to large release occur from early core damage. Here, early relates to the injection phase of the accident progression, roughly within a few hours following accident initiation.

Insights specific to PWRs include:

- Small loss-of-coolant pipe breaks are an important initiating event for seismic events.
- Dominant plant functions important to seismic core damage are failure to shut down the nuclear chain reaction in the core and failure of early emergency core cooling. Here again, early relates to the injection portion of the accident progression.

Insights for BWRs include:

- Large loss-of-coolant pipe breaks are an important initiating event primarily due to failures in the reactor vessel and recirculation pump supports.
- No seismic probabilistic analysis has been performed on a BWR Mark I plant.
- There were no overall general funtional insights; however, for Mark II and III plants, the suppression pool always succeeds and the power conversion system always fails primarily due to loss-of-offsite power.

Review of the probabilistic seismic analyses found a number of dominant component failures. These components are listed below:

- Yard Tanks

  - condensate storage tanks, refueling water storage tanks

- Electrical Equipment

  - batteries, buses, cabinet anchorage, contacts, relays, transformers

- Diesel Generator Peripherals

- fuel oil tanks, lube oil tanks, coolers

o Structural Failures

- block walls, services water buildings, Reactor Internals

o Equipment Anchorages.

Seismically induced fires have not been systematically considered in the seismic PRA literature for nuclear power plants. A review of recent earthquakes throughout the world indicates that fires have been caused by earthquakes at industrial facilities particularly where there is a significant amount of flammable material like refineries and chemical facilities. A majority of the damage during the 1906 San Francisco earthquake was caused by fire. There where as many as 86 fires caused by the Whittier 1987 earthquake.

Seismically induced floods were analyzed in the Oconee PRA [24]. No other analysis of seismically induced floods is given in the seismic PRA literature. The Oconee analysis considered late failure of long-term cooling due to the seismic failure of the Jocassee Dam. Short term cooling is initially successful, but fails when the site is flooded due to the dam failure. The mean probability of core damage as a result of the Jocassee Dam failure was given as 2.6 E-6 per reactor year.

Comparison of the core damage frequency for the seismic failure of the Jocassee Dam indicated that it does not meet the core damage figure-of-merit. However, a judgment as to the inclusion of this type of failure in the severe accident policy implementation cannot be made on the virtue of only one analysis. This seismic failure of an upstream dam is a unique feature of the Oconee site because the Jocassee Dam was not built to nuclear safety grade standards as was the nearby Keowee Dam. Upstream dam failures at other plant sites may pose a potential for severe accidents. Other seismic induced flooding issues involve the seismic failure of threaded fire water piping, other liquid carrying piping and liquid storage tanks that result in local flooding of electrical equipment, distribution and control panels and rooms without drains that house these type of components.

Common-cause failures are characteristic of external initiators, in particular, seismic events. Several plant components and systems can fail simultaneously because they are subjected to the same initiating event or caused as a result of the same external initiator. The plants ability to respond to the external event is then reduced because of multiple failures. Earthquake motion is felt by all plant components and systems simultaneously. The resulting seismic responses of the components are correlated based on their location in the plant and their elevation above ground level.

Chapter 4

SEISMIC EVALUATION EFFORTS FOR NUCLEAR POWER PLANTS

4.1 INTRODUCTION

Seismic evaluation efforts for nuclear power plants have been initiated primarily because of a perceived change in the seismic hazard in the Eastern United States and/or due to changes in seismic design requirements and practices by industry and the NRC. These seismic evaluation efforts have been directed at assessing the adequacy of plant safety systems and components to withstand earthquake levels equal to and greater than the plant design SSE.

These efforts involve several groups including the NRC Seismic Design Margins Working Group, the Seismic Qualification Utility Group (SQUG), the Senior Seismic Review and Advisory Panel (SSRAP), and the Expert Panel on the Quantification of Seismic Margins. The sponsors to these efforts include the NRC, and the Electric Power Research Institute (EPRI).

The purpose of this chapter is to provide a brief discussion of the seismic evaluation programs and an assessment of the extent to which they provide useful information on seismic capacity of nuclear power plants. The programs discussed in this section include the Systematic Evaluation Program (SEP) [31], the Seismic Safety Margins Research Program (SSMRP) [11], the efforts to address the NRC's Unresolved Safety Issue A-46 (USI A-46) for "Seismic Qualification of Equipment in Operating Plants" [32] and two seismic margins review programs, one conducted by Lawrence Livermore National Laboratory (LLNL) for the NRC entitled "Seismic Design Margins Program" [30] and the other conducted by EPRI on Seismic Margins Evaluation Methodology [33].

4.2 SYSTEMATIC EVALUATION PROGRAM

The Systematic Evaluation Program (SEP) sponsored by the NRC consisted of a plant-by-plant limited reassessment of the seismic safety of eleven operating nuclear power plants that received construction permits between 1956 and 1967. Because many safety criteria changed since these plants were initially licensed, the overall purpose of the SEP was to develop a current documented basis for the seismic safety of these older facilities. The eleven plants reviewed were Big Rock Point, Dresden 1 and 2, Ginna, Hadam Neck, LaCrosse, Millstone 1, Oyster Creek, Palisades, San Onofre 1, and Yankee Rowe.

The approach and methods developed for the SEP provided the basis for the several seismic assessment and evaluation programs that followed. The primary objective of the SEP seismic review program was to make a seismic safety assessment of the plants based on a limited sample of important safety structures, systems and components and, where necessary, recommend backfitting in accordance with 10 CFR 50. The SEP review concept was to determine whether or not a given plant meets the general level of safety and "intent" of current licensing criteria as defined in the Standard Review Plan.

The SEP evaluations relied upon a limited analysis of selected structures and sampling of representative components from a generic group of equipment. The component sample was augmented by a plant walkthrough inspection to select additional components, based on their potential seismic capacity. The seismic reevaluation centers on:

- o An assessment of the integrity of the reactor coolant pressure boundary; that is, major components that contain coolant for the core and piping or any component not isolatable (usually by a double valve) from the core.

- o A general evaluation of the capability of essential structures, systems, and components to shut down the reactor safely and to maintain it in a safe shutdown condition, including removal of residual heat during and after a postulated safe shutdown earthquake (SSE). The assessment of this subgroup of equipment can be used to infer the capability of such other safety related systems as the Emergency Core Cooling System.

Not all equipment was examined as part of this reassessment. The intent was to examine mechanical and electrical equipment representative of items installed in the reactor coolant system and safe shutdown systems for structural integrity and for electrical and mechanical functional operability. Components that potentially have a high degree of seismic fragility were selected for review in order to estimate the lower-bound seismic capacity in generic classes of equipment. The selection was made during a site visit.

Structures housing the selected systems were analyzed to demonstrate structural adequacy and to generate seismic input to equipment. The structures include the reactor building, with its related internal structures, and portions of the turbine and auxiliary buildings. For the structural evaluation, a peak horizontal ground acceleration corresponding to the original SSE value was used along with a NUREG-0098 (Newmark/Hall) response spectrum.

To ensure safety in a seismic evaluation, certain elements and components of an entire system must continue to function under normal operation both during and following an earthquake. The seismic review team did not review all aspects of the plant's operation nor did they review the safety margins available to assure that vital elements and components would withstand unusual operating conditions, operator error, or other non-seismic events. The reviews addressed systems and components in the as-built condition, including modifications made to all seismic Category I components since the issuance of the operating license.

4.3 SEISMIC SAFETY MARGINS RESEARCH PROGRAM

The Seismic Safety Margins Research Program (SSMRP) [11] was sponsored by the NRC in late 1978 and was completed in 1982. The overall goal of the SSMRP was to develop tools and data bases to evaluate the risk of earthquake initiated radioactive release from commercial nuclear power plants.

The methodology developed in this program provided the basis for seismic PRAs by defining the steps needed to assess the seismic risk at nuclear power plants. The methodology developed by the SSMRP was subsequently simplified based on continuing research and experience in performing probabilistic seismic analyses and seismic PRAs. This section briefly describes the SSMRP methodology and the subsequent simplified methodology.

4.3.1 SSMRP Methodology

There are five steps in the SSMRP methodology for calculating the seismic risk at a nuclear power plant:

1. Characterize the seismic hazard.
2. Determine response of structures and subsystems to seismic excitation.
3. Determine fragility functions.
4. Identify accident scenarios.
5. Calculate probability of failure and frequency of radioactive release.

A brief discussion of each of these steps is given below.

Step 1: Characterize the Seismic Hazard

The earthquake hazard at a given power plant site is characterized by a hazard function which gives the probability of exceedance (per year) of a ground motion parameter, such as peak ground acceleration. Figure 4.1 shows a representative hazard curve for a nuclear power plant site. This curve is derived from a combination of recorded earthquake data, estimated earthquake magnitudes of known events for which no data are available, review of local geological investigations, and use of expert opinion based on a survey of seismologists and geologists familiar with the region in question.

The frequency characteristics of the earthquakes are required, as well as their likelihoods. Response spectra are used to define their frequency characteristics. From these response spectra, artificial acceleration time histories are generated. Three orthogonal components (two horizontal and one vertical) of acceleration time histories are generated for each earthquake simulation.

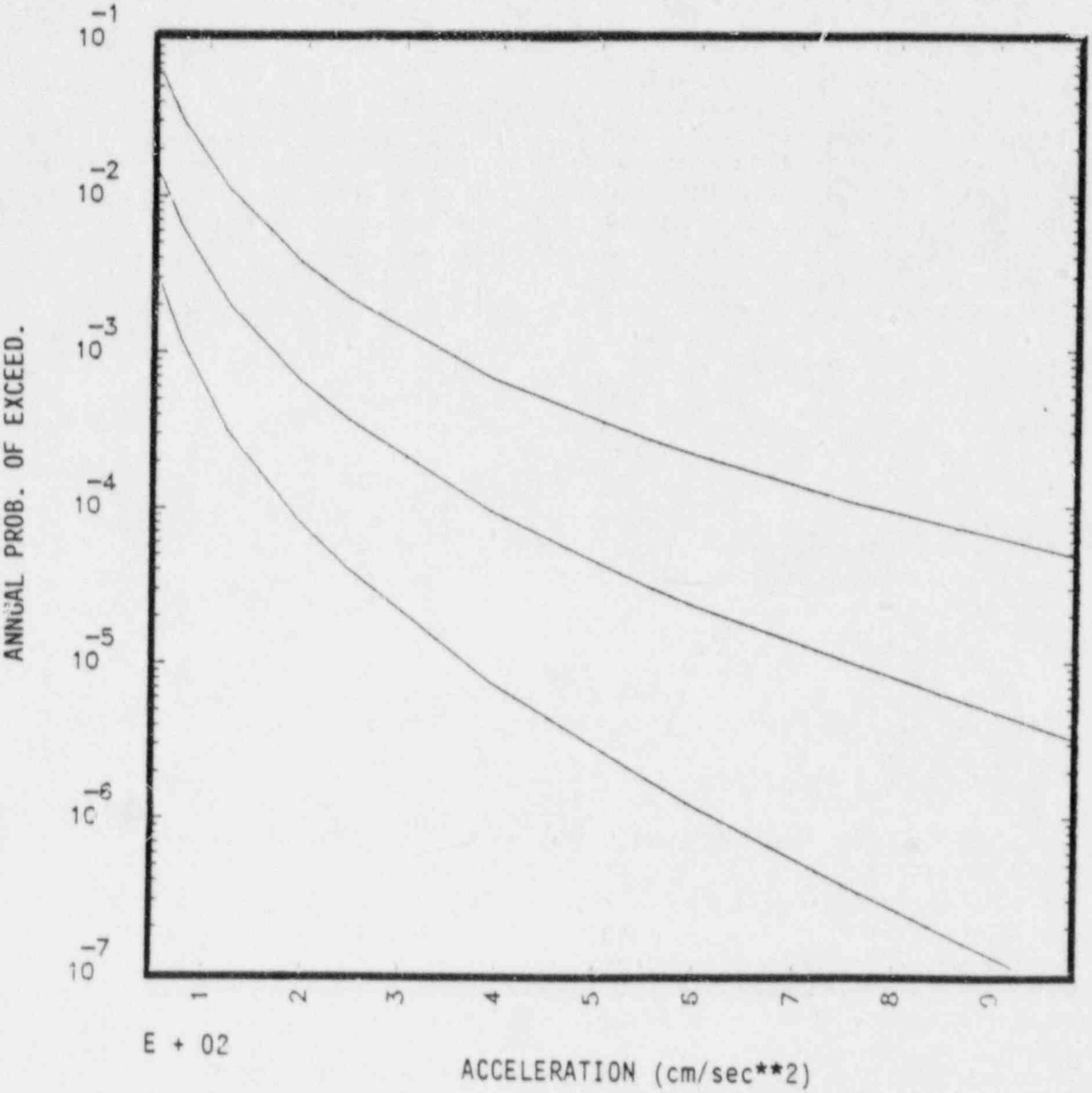

Figure 4-1 Representative Seismic Hazard Curves

Step 2: Determine Response of Structures and Subsystems to Seismic Excitation

Seismic excitation is given by ensembles of acceleration time histories in three orthogonal directions, obtained as we described above for Step 1. Soil-structures interaction (SSI) and detailed structure response are determined simultaneously using the substructure approach to SSI. The response of subsystems is calculated by multi-support time history analysis procedures. Uncertainty is treated explicitly in each link of the seismic methodology chain by analyzing an ensemble of free-field acceleration time histories and by varying a discrete number of input parameters of the soil, structures, and subsystems. Repeated deterministic analyses are performed, each analysis simulating an earthquake occurrence.

Step 3: Determine Fragility Functions

Different subsystems, structures, parts of structures, and components have different susceptibilities to failure as a result of an earthquake. These different susceptibilities must be determined. This susceptibility is specified by a fragility function, which is a cumulative probability of failure as a function of loading. Fragility functions are developed for structures, large components, and many categories of small components.

Step 4: Identify Accident Scenarios

All failures are not equally serious. In some accidents, safety systems will be effective, either permitting the continued operation of the plant or bringing about a safe shutdown. In other accidents, safety systems could be ineffective, and in extreme cases radioactive release would occur.

In this step of the risk analysis process, the possible accident scenarios during an earthquake-induced shutdown are identified using event trees. Accident scenarios vary from minor to severe. Fault trees are used to determine the success or failure of each of the safety systems whose success or failure make up the accident sequence.

Once a probability is associated with each event in an accident scenario, the probability of each scenario can be calculated. The process used to determine the probabilities is described in Step 5.

Step 5: Calculate Probability of Failure and Frequence of Radioactive Release.

Step 5 combines the results of Steps 1-4 to express plant risk as the frequency of 1) failure of structures and components, 2) failure of a group of structures and components, and 3) radioactive release.

(a) Calculate Cut Set Probabilities. Each accident sequence consists of the union of sets of events (successes or failures of components) which must occur together to have system failure. (In systems analysis terminology, these sets are called cut sets). Each accident sequence contains up to 5000 of these component failure groups and each component failure group (cut set) is allowed to have up to ten basic events (component failures).

The computer code SEISIM [45] was written expressly to Calculate the probability of such component failure groups including all common-cause failures. Given the individual component responses and fragilities (in terms of the means and variances of their distributions) and given the computed correlations between the responses (obtained from the time history response calculations at each earthquake level), SEISIM constructs a multi-variate lognormal distribution for each component failure group, and then uses n-dimensional numerical integration to compute the probability of occurrence of the component failure group.

(b) Calculate Frequency of Radioactive Release Once the component failure group probabilities have been computed, the probability of each accident scenario can be found from the expression for the union of disjoint cut sets, which is an upper bound to an accident scenario probability. Then each accident scenario probability is multiplied by the probability of the earthquake's occurrence, the probability of the initiating event, and the probability of failure of the containment, to obtain the frequency of radioactive release. Several different containment failure modes of different severity are identified, ranging from rupture of the containment shell to leakage of containment isolation valves. Different containment failure modes are assigned to different accident scenario according to our understanding of the physical processes involved. One accident scenario can result in one or more containment failure modes.

Finally, accident sequence probabilities are assigned to different release categories to reflect their severity with respect to radioactive release to the surrounding population. These release categories relate to the type and energy content of the radioactive fission product release, as well as the mode and timing of the release. They range from rupture of the top of the containment with a rapid, high energetic release (due to a fuel/water explosion or steam overpressure) down to slow melt-through of the containment concrete foundation, which is expected to have the least effect on the surrounding population. The containment failure modes and the release categories are those derived and used in the Reactor Safety Study.

4.3.2 Simplified Methodology

This section highlights some of the features of the simplified methodology [12] and indicate how it differs from the previous, more detailed SSMRP methodology.

Plant and Site Familiarization. There are no significant differences between the SSMRP detailed and simplified approaches in the plant and site familiarization area.

Earthquake Hazard. In the detailed analysis, the seismic hazard is characterized in terms of seismic hazard functions, ground response spectra, and earthquake time histories. At each acceleration range of interest, typically 30, but as many as 90 sets of three components of realistic time histories of motion, are specified at the surface of the soil.

In the simple analysis, the hazard curves are mostly based on the methodology and data given in the NRC sponsored Eastern United States Seismic Characterization study [9] and comparable industry studies [10]. Western sites would require a site-specific hazard characterization as part of the risk assessment.

The simplified methodology is fundamentally different from the detailed in that the simplified methodology requires spectra but not time histories. Some time histories may be developed as part of the simplified seismic risk assessment for the purpose of benchmarking the seismic response of structures or piping systems, but this is optional.

Plant Logic Models. There are no significant differences between the SSMRP detailed and simplified approaches in the development of plant logic models.

Seismic Response. For each level of earthquake described by the seismic hazard curve, three aspects of seismic response are necessary to perform the seismic risk analysis: best-estimate response, variability of response, and correlation of responses.

The simplified approach performs a limited amount of recalculation of the responses using best-estimate methods and parameters and applies scale factors to the design responses. Rather than generate these responses for all earthquake level (as would be done in a detailed analysis), the simplified analysis performs selected response analyses of structures for only two ranges of earthquakes--a lower level earthquake and a higher level earthquake. These two levels permit interpolation of responses for other earthquake levels.

The variabilities of response are not based on calculations (as in the detailed approach) but are taken from the SSMRP Simplified Methods report [12]. This report specifies the random and modeling uncertainty values shown in Table 4.1.

The correlations of response are not based on calculations (as in the detailed approach) but are taken from the SSMRP Simplified Methods report [12]. This report specified correlations of response for four different earthquake levels. This information is used to develop a correlation matrix that is weighted over the earthquake range. These correlations are given in Table 4.2.

Fragility. There are no significant differences between the SSMRP detailed and simplified approaches in the fragility area. Fragilities have to be developed for each risk assessment based on the information available at the time.

Plant Risk Quantification. There are no significant differences between the SSMRP detailed and simplified approaches in this area. Point estimates are useful for sensitivity studies but should be used with caution as an estimator of the mean. The mean should be estimated through uncertainty analysis using recommended values for the modeling uncertainty.

Assessment of Results and Interpretation. There are no significant differences between the SSMRP detailed and simplified approaches in the analysis results and interpretation area.

TABLE 4-1 RESPONSE RANDOM AND MODELING UNCERTAINTIES

	Random Uncertainty	Modeling Uncertainty
1. Structure	.28	.27
2. Piping Moment	.40	.57
3. Valve acceleration	.34	.43
4. Equipment mounted in building with fundamental frequencies not near soil structure frequencies	.28	.27
5. Equipment mounted in buildings with fundamental frequencies near soil structure frequencies	.40	.57

TABLE 4-2 RESPONSE CORRELATION DATA

	Structure/ Equipment	Piping Moment	Valve Acceleration	Electrical
Structures/Equipment	0.91	0.10	0.26	0.40
Piping Moment		0.29	0.14	0.04
Valve Acceleration			0.76	0.24
Electrical				0.99

4.4 SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (USI A-46)

There have been significant changes in the seismic design requirements and practices for the seismic qualification of equipment at nuclear power plants. The most recent changes occurred in Regulatory Guide 1.100, IEEE Standard 344/1987, and Standard Review Plan 3.10 for the qualification of equipment in licensed plants. The goal of the A-46 [32] effort is to address these changes by reviewing the seismic adequacy of certain equipment at older operating nuclear power plants against seismic criteria not used when these plants were licensed. All plants not reviewed to these current qualification requirements are included in the A-46 reviews.

The objective of these reviews is to verify the seismic adequacy of mechanical and electrical equipment that is required to bring the plant to a safe shutdown condition and maintain this condition for at least 72 hours. The approach is to review essential plant functions, systems, components, instruments and controls required to establish and maintain hot shutdown during and following an SSE level earthquake.

The implementation of A-46 uses an extensive earthquake experience data base gathered by SQUG and their consultants, and reviewed by SSRAP. This data base provides information about the seismic ruggedness of equipment at fossil fuel power plants and heavy industrial facilities.

Seismic test data collected by EPRI was also used in this program. This data was used initially to identify 8 classes of equipment for review and to give guidelines for performing their review. However, more recent data have allowed the review to be extended to the 20 classes of equipment given below:

1. motor control centers
2. low voltage switchgear
3. medium voltage switchgear
4. medium/low voltage transformers
5. horizontal pumps
6. vertical pumps
7. fluid operated valves
8. motor operated valves
9. fans
10. air handlers
11. chillers
12. air compressors
13. motor generators
14. distribution panels
15. battery racks
16. battery chargers
17. engine generators
18. instrument racks
19. temperature sensors
20. control and instrumentation cabinets.

More recently, tanks and heat exchangers have been considered along with relays, contactors and motor starter chatter. Unreinforced masonary walls have been the subject of recent inspection, review and upgrade. However, these walls have not been considered as part of this program.

Each class of equipment is reviewed for one of three seismic motion bounding spectra to compare the potential seismic exposure of identified equipment with the estimated motion that similar equipment actually resisted in earthquakes described in the data base. The implementation of these reviews calls for detailed plant walkdowns and inspections of identified equipment by an experienced team of engineers and operations personnel. The walkdowns concentrate on equipment anchorage and supports, compliance of the equipment with the A-46 review guidance, and seismic spatial systems interaction issues.

The results of an A-46 review include a list of equipment required for the establishment and maintenance of safe shutdown, a certification of the plant walkdown and inspection, identified deficiencies, identified outliers (equipment not within review classes), and actual or planned modifications and upgrades. The information obtained from these reviews is a plant-specific data base applicable to the assessment of the seismic behavior of the plant up to the plant's seismic design (SSE) level. Tap A-46 reviews have been initiated along with a trial plant review of the Nine Mile Point Nuclear Power Plant.

4.5 SEISMIC MARGINS PROGRAMS

The seismic margins programs for nuclear power plants were initiated because of the changing perception of the seismic hazard. The goal of these programs is to assess how much larger an earthquake must be above the design basis SSE before it compromises the safety of the plant. A seismic PRA can answer this question along with providing considerably more information about the plant. However, the seismic margins approach, though based on PRAs, was developed to be performed in less time and at less cost.

Two seismic margins programs are directed at the assessment of seismic capacity of nuclear power plants: the NRC-sponsored Seismic Design Margins Program and the EPRI-sponsored Seismic Margins Evaluation Program. This section briefly discusses each of these programs.

4.5.1 NRC Seismic Design Margins Program

The Seismic Design Margins Program (SDMP) was initiated by the NRC in 1984 [46] to address regulatory needs and the changing perception of the seismic hazard. The NRC formed the Expert Panel on the Quantification of Seismic Margins and charged them to work closely with an NRC staff Working Group on Seismic Margins. The overall goal of the program was to develop a methodology and guidelines that can be readily used by the NRC and industry for assessing the inherent quantitative seismic capacity of nuclear power plants.

The Expert Panel developed the seismic margins review approach [30]. Their document formed the basis for the development of guidelines for performing seismic margin reviews. These guidelines are given in Ref. 34.

The approach to performing seismic margin reviews is to determine the capacity of a plant to respond to a specified earthquake level greater than the SSE and to identify any seismic plant vulnerability or "weak-links." The specified earthquake level for margin reviews has been referred to as the Seismic Margin Earthquake (SME).

A screening approach was developed from a review of seismic PRAs, earthquake experience data from industrial facilities, test data, and expert opinion. This screening approach involves screening components based on their importance in preventing core damage and their inherent seismic capacity.

The adopted measure of margin is the earthquake level for which there is a High Confidence, Low Probability of Failure (HCLPF). The HCLPF is a conservative representation of seismic capacity. It corresponds to the earthquake level at which it is extremely unlikely that plant, system, and component failure or core damage will occur. From a mathematical perspective, the HCLPF capacity is approximately the capacity at which there is a 95% confidence of not exceeding about a 5% probability of failure.

Available sources of fragility information were used to arrive at conclusions as to which components should be assessed from a seismic capacity standpoint. Three ranges were used for a capacity assessment of each component stated in peak ground acceleration (pga). These ranges are: (1) less than 0.3g, (2) 0.3g to 0.5g, and (3) greater than 0.5g. Each type of nuclear power plant component was assessed to have a generic HCLPF capacity within one of these ranges. This assessment resulted in an extensive table of components indicating at what earthquake level each component will either require a margin review or be removed (screened out) from the review process.

Systems screening is performed by considering those components that makeup the systems needed to perform and support the important plant functions to avoid seismic core damage. For PWR plants, these functions are reactor subcriticality and early emergency core cooling. <u>Early</u> refers to the injection portion of the accident progression. For BWR Mark I plants, all functions need to be considered. For BWR Mark II and III plants, all functions need to be considered except the suppression pool function [35]. In addition, the establishment and maintenance of shutdown needs to be considered for all plants.

The first step in performing a seismic margins review requires the specification of the seismic margin earthquake including its level and spectral shape. Once the SME has been specified, the implementation of a review requires data gathering and review, and two plant walkdowns performed by two teams of experienced engineers: a fragility analysis team and a systems analysis team.

The first plant walkdown is performed to gather data and information on the plant configuration and operation, and to verify that the condition of the plant warrants the use of the generic information given in the components capacity table for screening components to the selected SME range. The walkdown also concentrates on the identification of seismic spatial systems interactions and any plant unique features such as upstream dams that could cause flood-induced failures. From this initial plant visit, the information gathered is used to develop a plant model employing event trees and fault trees for the front-line and support systems that perform the two important plant functions. In addition, those components with HCLPF capacities greater than the SME, represented by the generic capacities given in the screening table, were screened out of the remainder of the analysis. The plant models are analyzed to derive Boolean expressions that indicate the cut sets and important components to seismic core damage.

A second plant walkdown verifies the plant models and collect detailed information needed for the assessment of the HCLPF capacity of those remaining important components. Component HCLPF capacities can be determined by two methods: conservative deterministic failure margins (CDFM) and fragility analysis (FA) methods.

The Boolean expressions are analyzed using component HCLPF capacities to determine an overall plant HCLPF capacity for the selected SME. The analysis of the Boolean expression can include non-seismic events such as human error, and test and maintenance activities. The results of the analysis also provide an indication of those low-capacity components that are dominant to seismic core damage and can be considered plant seismic vulnerabilities.

If the plant HCLPF value is found to be greater than the SME, then only a qualitative indication that the plant has a HCLPF capacity greater than the SME can be made. However, if the plant HCLPF value is less than the SME, then a plant HCLPF capacity is determined.

The NRC seismic margins review methodology was used for a trial review of the Maine Yankee Atomic Power Station [36, 37, 38]. The objectives of this trial review were:

- o To demonstrate the use of this methodology and guidelines for seismic margins reviews.
- o To provide a basis for revising and upgrading the approach and guidelines based on lessons learned.
- o To provide a benchmark for possible future seismic margins reviews, including an understanding of the level of effort in performing a seismic margin review.

- o To provide an assessment of the plant's capability to withstand a specific earthquake level greater than the SSE.

The Maine Yankee plant was reviewed for an SME of 0.3g. The analysis considered both a small LOCA and transient-without-LOCA as initiating events. Seismic and non-seismic failures were considered during the analysis. Component HCLPF capacities were derived using the fragility analysis method. The analysis also considered the common mode dependence between component seismic failures. The results of the seismic margin review indicated that the plant HCLPF capacity was 0.21g. This capacity is dominated by the small LOCA accident sequences with the refueling water storage tank (RWST) being the dominant component. There was no effect on the plant HCLPF capacity when dependence between component failures was considered and little effect from the inclusion of non-seismic failures in the analysis.

Other components that were found to have a low seismic capacity were:

- o Aged lead-antimony station batteries for which no experience data were available
- o an important 4.16 kV station service transformer
- o a block wall near HVAC equipment needed to cool a pump enclosure that houses long-term cooling equipment
- o component cooling water heat exchangers
- o diesel generator fuel oil day tank anchorage.

The plant HCLPF capacity of 0.21g included the planned upgrade or modification of the above listed components. This plant HCLPF capacity is governed by the RWST and represents a conservative estimate of the seismic capacity of the plant. After completion of this review, Maine Yankee decided to upgrade the RWST, this will result in an increased plant HCLPF capacity of at least 0.27g.

The NRC staff reviewed the results of the seismic margin review and Maine Yankee's commitment to upgrade identified components. They concur with the findings and have issued a safety evaluation report which concludes that the plant has an adequate seismic margin [39, 40].

4.5.2 EPRI Seismic Margins Program

The EPRI Seismic Margins Program [33] was initiated after the start of the NRC Seismic Design Margins Program. This EPRI program has the same overall objective as the NRC program in providing a methodology and guidelines for assessing the inherent seismic capacity of nuclear power plants that can be readily used by the NRC and industry.

The approach of the EPRI program is similar to the SDMP, that is, to determine the capacity of the plant to respond to a specified earthquake level greater than the SSE. One major difference is that the EPRI approach does not specifically address plant seismic vulnerabilities.

The EPRI approach uses the HCLPF capacity as the figure-of-merit for the review and involves screening components based on their capacity and importance to plant hot or cold shutdown. The NRC component capacity table developed in the SDMP was updated and revised during this program to reflect an advancement in the state of knowledge about the seismic capacity of components. The EPRI methodology recommends the CDFM approach for the assessment of component HCLPF capacity. The EPRI methodology also embraced issues not considered in the SDMP, such as consideration for soil sites and seismically induced relay and contactor chatter.

The implementation of the EPRI review involves experienced seismic capacity assessment engineers and systems engineers directed by a "Seismic Review Team." They perform an initial seismic capacity walkdown followed by subsequent walkdowns if needed.

The major difference between the EPRI methodology and the NRC SDMP methodology is in plant modeling and systems analysis. The EPRI methodology replaces the event tree and fault tree technique with a "success path" approach. This approach defines those components required for an operational sequence of plant systems that will bring the plant to a stable condition (either hot or cold shutdown) and maintain that condition for at least 72 hours. The set of components needed for the success of the systems that perform this operational sequence is called a "success path."

There are many possible success paths. The object is to select the success path that will most likely be used by plant operation personnel and indicated by operational (standard and emergency) procedure to address the accident situation being considered. Only those components within this success path need to be reviewed for the seismic margin analysis. One primary and one alternative success path need to be considered for each postulated initiating event (LOCA, transient, etc.). The seismic margin capability (expressed in terms of HCLPF) for any success path is then equal to the seismic capacity of the weakest component in the success path. While it is possible not to find the path with the highest capacity, the resulting HCLPF value is taken as the plant capacity.

The ERPI methodology was used in a trial review of the Catawba Nuclear Station [41]. The seismic margin earthquake for the Catawba review was 0.3g. Three shutdown success paths were analyzed: (1) feed and bleed with subsequent open loop recirculation using the residual heat removal (RHR) system, (2) steam generator cooling via the auxiliary feedwater and subsequent closed decay heat removal via the RHR system, and (3) steam generator cooling with high pressure injection or charging for inventory make-up. Success paths

1 and 3 addressed the small LOCA-initiating event while success path 2 assumed that no LOCA occurs.

The results of the Catawba review indicate a plant HCLPF capacity of 0.24g. The dominant components were motor control centers and diesel generator peripherals. A very conservative floor spectrum was used for this trial review. Revision to this floor spectrum would increase the plant HCLPF capacity above the 0.3g level.

The EPRI methodology was reviewed by a "Panel to Review the EPRI Seismic Margins Methodology" under NRC sponsorship [42]. This panel also examined the Catawba trial review, but only to gain insight into how the methodology is applied. A detailed technical review of the Catawba analysis was not performed. In summary, the panel found that the EPRI methodology can accomplish its main objective, can be performed repeatedly, and is reasonably accurate. The methodology can determine whether a given "success path" has a HCLPF value above the SME; and if not, what the HCLPF value is for that success path and what components/structures are dominant contributors to the HCLPF value for that success path.

4.6 A DISCUSSION OF SEISMIC ASSESSMENTS IN RESPONSE TO SEVERE ACCIDENT POLICY IMPLEMENTATION

Review of the various programs for the assessment of seismic capacity of nuclear power plants, strongly indicate that these programs should be combined into a single integrated approach that will address the various concerns. It would not be desirable to require a plant to perform an A-46 plant assessment for their SSE, followed by a seismic margins assessment for larger-than-SSE-level earthquakes in response to eastern seismicity concerns, and then to perform a severe accident policy implementation analysis of plant core damage and radioactivity release from a seismically initiated event.

The various attributes of each program could be combined into a intregrated approach to seismic plant assessment following an overall seismic margins methodology. Each type of plant analysis could provide information and data that can be applied to another part of the analysis. The ultimate goal of the review process is to provide a quantitative assessment of a plants ability to achieve and maintain shutdown for beyond design basis earthquakes and to identify any seismic plant vulnerabilities.

A brief discussion of a possible approach follows:

1) For those plants that are undergoing an A-46 review, the information and data collected and the results can be used along with the seismic margins review table as the basis for initially dividing components into those that will be screened out from further review and those that will remain in the analysis. The A-46 results may even provide the necessary information for making preliminary analysis of component capacities and HCLPF values.

TABLE 4.3

COMPARISON OF THE WALKDOWN REQUIREMENTS
BETWEEN THE A-46
AND
THE NRC SEISMIC DESIGN MARGINS PROGRAMS

	USI A-46 Requirements	SDMP Requirements
Earthquake Description	Verify that data base spectra envolopes site ground surface free field spectra for SSE	NUREG-0098 or site specific spectra anchored at seismic margins earthquake (SME > SSE, 0.3g typical)
Walkdown Team(s)	One Team - structural engineer, electrical engineer, mechanical engineer, operations person (supervisor, SRO)	Two Teams - Fragility Team: structural engineer, equipment seismic behavior engineer, utility mech./struc. Systems Team: plant systems engineer with PRA experience, operations person (supervisor, SRO), maint./testing person
Equipment	Equipment necessary to perform functions related to plant shutdown and required during strong motion	Equipment necessary to perform functions of subcriticality, early ECCS and plant shutdown
Review	one plant walkdown, review for SSE level, 20 classes of equip., primarily anchorage and supports, systems interactions, relay functionality, no structures	two plant walkdowns, review estimate, strctures, anchorage and supports, seismic capacity of tanks and equip., plant unique features, systems interactions
Results	equipment required for shutdown equipment required during strong motion, their adequacy, deficiencies and outliers, improved anchorage and supports	verification of use of screening table, screened-out components, HCLPF capacities for screened-in components, improved anchorage and supports, strengthened outliers

In the future, the A-46 review can be modified such that the information and data collected, and the results will contain the necessary information to constitute the first walkdown that would be performed during a seismic margin review. A comparison of the walkdown requirement between the A-46 and the NRC Seismic Design Margins programs is shown in Table 4.3.

Along with the result needed to fulfill the A-46 effort for the plant, the review should verify the use of the generic information contained in the seismic margins screening table (Table 5 of Ref. 31), identify any possible seismic spatial systems interaction, and any plant-unique features.

2) If there is a concern about the eastern seismicity at a particular plant site, a seismic margins review can be performed for an appropriate seismic margins earthquake (SME) level. If the particular plant has had an A-46 review, only one plant walkdown will be needed to complete the review. Initial information gathering (concerning a plant's configuration and operation), initial component screening, and plant modeling can be performed prior to a seismic margins plant walkdown. The remainder of the seismic margins review can then address the eastern seismicity concerns. If the plant has not had a A-46 review, then a complete seismic margins review can address eastern seismicity concerns.

3) To address the implementation of the severe accident policy for seismic events, a seismic margins review can be performed, in conjunction with 1) and 2) above, if required. The selection of the seismic margins earthquake (SME) used for the review can be based on the site specific hazard curve at a particular acceleration level as compared to some evaluation criterion. This comparison could indicate at which level the review should be performed (e.g., less than 0.3g, between 0.3g and o.5g, or above 0.5g). In addition, plants located in areas of low seismic hazard that have a median annual recurrence frequency at their SSE less than the evaluation criterion could possibly be screened out. Such plants would have to have sufficient capacity at the SSE level and above to withstand an earthquake with a core damage frequency consistent with the evaluation criterion as compared to the figure-of-merit. Studies would have to be undertaken to see if plants have such capacity before screening based on the SSE reoccurence frequency could be established.

The results of the seismic margin review would be a plant HCLPF capacity for the SME level, and an indication of seismic plant vulnerabilities. The plant HCLPF value will either be a conservative estimate of the plant's capacity or an indication that the plant has a HCLPF capacity greater than the SME.

The plant HCLPF value, if smaller than the SME, can then be used along with a site specific hazard curve to estimate the recurrence frequency for this level earthquake. A plant HCLPF capacity equal to or greater than the SME would be considered to have adequate capacity since the SME would be chosen to assure adequacy.

If the resultant recurrence frequency derived from the plant HCLPF capacity is less than the evaluation criterion decided on, then the plant could be considered to have adequate capacity to meet severe accident policy concerns. If however, the recurrence frequency at the plant HCLPF value is greater than evaluation criterion, plant vulnerabilities should be addressed.

The plant HCLPF capacity represent a conservative estimate at which there is a high confidence of a low probability of core damage. A more realistic parameter is the plant median capacity which is more than a factor of two greater than the HCLPF capacity [30]. Therefore, it has been suggested that two times the plant HCLPF capacity could be used in conjunction with the median site specific hazard curve to obtain a recurrence frequency for comparison with some evaluation criterion. In light of the screening approach used for seismic margins reviews, it is not clear what the appropriate factor should be and whether the resultant recurrence frequency presents an effective parameter for comparison to some evaluation criterion. Research is needed to address what may be the appropriate factor that can be used along with the plant HCLPF capacity and what would be an appropriate evaluation criterion.

If the fragility analysis technique is used to determine plant HCLPF capacity for a seismic margins review, a fragility curve for plant capacity is derived. This plant fragility curve can be convolved with a site specific hazard curve to obtain a distribution for the probability of core damage. The median or mean frequency for core damage can then be compared to some probabilistic evaluation criterion or safety goal. This approach can also take into account non-seismic failures such as test and maintenance unavailabilities, random failures, and human errors. However, due to the screening approach of seismic margins reviews, the contribution to core damage from earthquakes above the SME is not considered. Research is needed to understand the significance of above SME contributions and on the possible use of this approach.

The seismic margins approach is directed toward the evaluation of core damage from seismically initiated accidents and does not consider radioactive release. Additional analysis would be required to consider the radioactive release portion of severe accidents. Consideration would need to be given to the accident phenomenology following seismic core damage and the resultant containment response. In addition, the seismic capacity of containment structures for gross failure would need to be considered along with the seismic capacity of containment penetrations to prevent "large" releases.

A seismic margins approach that addresses the severe accident policy implementation issue should also incorporate recent efforts on providing technical insights into seismically induced large radioactivity release [43] and techniques for analysis of relay and contactor chatter [44].

Chapter 5

CONCLUSIONS AND RECOMMENDATION

The seismic external hazard has been found to be important with respect to both figures-of-merit. Accidents initiated by seismic events have core damage frequencies comparable to the first figure-of-merit of E-5 per year and large release frequencies comparable to the second figure-of-merit of E-6 per year. Therefore, the seismic external hazard should be included in any vulnerability search and considered further as part of the severe accident policy implementation.

It is possible to separate seismic probabilistic risk analysis into two parts. The first part is the probability of the seismic initiating event and the second part is the response of the plant to this initiating event. Any consideration of seismic external hazards must include both the local seismic hazard and the assessment of the plant's response.

The seismic external hazard has been well studied including the performance of several seismic PRAs. There are presently several ongoing efforts that address different aspects of seismic hazard. These efforts include seismic hazard characterization, and assessments of a plant's response to design basis and beyond-design-basis earthquakes. The techniques and results of these efforts can be used to address severe accident concerns.

In addressing seismic vulnerability analyses, a full-scope seismic PRA type of analysis is always an acceptable approach. However, a seismic margins approach appears to provide the necessary degree of analysis, accuracy, and results needed to address these concerns. This methodology along with a possible screening approach based on site-specific seismic hazards can be used. The seismic margins approach should integrate the several seismic analysis efforts into a single application that will address several issues including the implementation of the severe accident policy.

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NRC FORM 335 (8-87) NRCM 1102, 3201, 3202

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BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by PPMB, DPS, add Vol. No., if any)	NUREG/CR-5042 UCID-21223 Supplement 1
2. TITLE AND SUBTITLE	Evaluation of External Hazards to Nuclear Power Plants in the United States Seismic Hazard
3. LEAVE BLANK	
4. DATE REPORT COMPLETED (MONTH / YEAR)	February 1988
5. AUTHOR(S)	P. G. Prassinos
6. DATE REPORT ISSUED (MONTH / YEAR)	April 1988
7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)	Lawrence Livermore National Laboratory P. O. Box 808 Livermore, CA 94550
8. PROJECT/TASK/WORK UNIT NUMBER	
9. FIN OR GRANT NUMBER	A0815
10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)	Division of Reactor and Plant Systems Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555
11a. TYPE OF REPORT	Technical
b. PERIOD COVERED (Inclusive dates)	
12. SUPPLEMENTARY NOTES	

13. ABSTRACT (200 words or less)

As part of the research program supporting the implementation of the NRC Policy Statement on Severe Accidents, the Lawrence Livermore National Laboratory (LLNL) has performed a study of the risk of core damage to nuclear power plants in the United States due to seismic initiated events. The broad objective has been to gain an understanding of whether or not seismic events are among the major potential accident initiators that may pose a threat of severe reactor core damage or of large radioactive release to the environment from the reactor.

The analysis was based on two figures-of-merit, one based on core damage frequency and the other based on the frequency of large radioactive releases. Using these two figures-of-merit as evaluation criteria, it has been possible to ascertain that the risk from seismic initiated accidents is an important contributor to overall risk for the U.S. nuclear power plants studies.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS	severe accidents seismic hazards
b. IDENTIFIERS/OPEN-ENDED TERMS	
15. AVAILABILITY STATEMENT	Unlimited
16. SECURITY CLASSIFICATION (This page)	Unclassified
(This report)	Unclassified
17. NUMBER OF PAGES	
18. PRICE	

*U.S. GOVERNMENT PRINTING OFFICE:1988-202-292:80133

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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NUREG/CR-5042, Supp. 1

Evaluation of External Hazards to Nuclear Power Plants in the United States

April 1988