Risk-Informed Public Safety Policy for Seismic Events in the Vicinity of a Nuclear Power Plant

By

Olubukola Afolayan Jejeloye

B.S. Nuclear Engineering / B.S Electrical Engineering
University of California, Berkeley

SUBMITTED TO THE DEPARTMENT OF NUCLEAR ENGINEERING AND THE ENGINEERING SYSTEMS DIVISION IN PARTIAL FULFILLMENT OF THE REQUIREMENTS FOR THE DEGREES OF

MASTER OF SCIENCE IN NUCLEAR ENGINEERING
and
MASTER OF SCIENCE IN TECHNOLOGY AND POLICY

AT THE
MASSACHUSETTS INSTITUTE OF TECHNOLOGY
FEBRUARY 2002

Signature of Author:..
Department of Nuclear Engineering
Technology and Policy Program
31 January 2002

Certified By:.. George E. Apostolakis
Professor of Nuclear Engineering and Professor of Engineering Systems
Thesis Supervisor

Certified By:.. Michael W. Golay
Professor of Nuclear Engineering
Thesis Reader

Certified By:.. Richard K. Lester
Professor of Nuclear Engineering
Policy Reader

Accepted By:.. Jeffrey A. Coderre
Associate Professor of Nuclear Engineering
Chairman, Committee for Graduate Students

Accepted By:.. Daniel Hastings
Professor of Aeronautics and Astronautics and Professor of Engineering Systems
Director, Technology and Policy Program
Chairman, Committee for Graduate Students

Risk-Informed Public Safety Policy for Seismic Events in the Vicinity of a Nuclear Power Plant

By
Olubukola Afolayan Jejeloye

Submitted to the Department of Nuclear Engineering and the Engineering Systems Division
On January 31, 2002 in Partial Fulfillment of the
Requirements for the Degree of Master of Science in
Nuclear Engineering and the Technology and Policy Program

Abstract

Nuclear Power Plants (NPPs) are potentially vulnerable to accidents, which can either be internally or externally initiated. External events include natural events like tornadoes, hurricanes, and earthquakes. The purpose of this thesis is to understand the characteristics of public risks arising due to a severe external event, in this case an earthquake, which affects the public both directly and via damage to a nuclear power plant. The possibility of developing a comparison basis for the risks from these two events is also investigated. Using the Seabrook Nuclear Power Plant PRA as a case study, consequences of a seismically induced nuclear risk are evaluated. Bases of comparison with direct seismic risks in Seabrook and Boston, calculated using FEMA's HAZUS program, are then analyzed. Results obtained show that the nuclear risks contribute little to the background risks from the direct earthquakes. Some consequences such as prompt fatality from the direct effect of earthquakes are 100 to 500 times bigger than the risks from the seismically induced nuclear risks at different magnitudes of earthquakes. Other consequences used for comparison include injuries and economic damage.

Comparative analyses of the direct earthquake risks and the seismically induced nuclear risks present a good means of communicating the risks posed to the public. Easily understandable, these comparative analyses can be utilized in making societal decisions about risks. Based on the results from the comparisons, risk informed policies for keeping the seismically induced nuclear risks low are proposed, including keeping the ratio of the nuclear risks to the direct effect risks to a level based on societal decisions. The results and proposals obtained were presented to a panel of experts who also suggested the use of the 3-region approach in making the nuclear power plants safe.

Considering events like the Turkey Point NPP experience with Hurricane Andrew, existing plans in place, such as communication and transportation after a major event, are considered. Noting that an earthquake that is strong enough to damage a NPP will affect much of the infrastructure needed to carry out emergency plans, means of strengthening the plans were evaluated. It was concluded that there has to be more cooperation among the different levels of government and the NPPs should be allowed a more active role in the policy and plan development for the safety of the public in their vicinity.

Thesis Supervisor/Advisor: Professor G. E. Apostolakis
Title: Professor of Nuclear Engineering and Professor of Engineering Systems

Thesis Reader: Professor M. Golay
Title: Professor of Nuclear Engineering

Policy Reader: Professor R. Lester
Title: Professor of Nuclear Engineering

Acknowledgements

My first and most thanks go to God, who spared my life on numerous occasions so that I could complete this phase of my life's journey. My gratitude goes to my thesis advisor, Professor George Apostolakis. Your warmth, generosity and guidance are and will always be appreciated. I never mentioned this to you, but you remind me much of my father. Thank you for the lunches, talks and for introducing me to Casablanca. I would also like to thank Professor Michael Golay, who co-supervised of my research work. I am forever in your debt for the challenges, financial support and encouragement you always offered. I want to extend my thanks to the professors, in both of my departments, who touched and shaped my life at MIT: Richard Lester, Ken Czerwinski, Kent Hansen, Richard Tabors, Neil Todreas, Sidney Yip and others who affected my life indirectly. I would like to acknowledge and thank my research colleagues, Jon Kim (my brother), Ying Li and Mingyang Xu. It was a pleasure working with you all.

Much gratitude goes to the staff members of both departments, who always made me feel at home and were always there to assist me when needed. Clare, Rachel, Masha, Paula, Mark, Mike, Linda, Dorian, Gail, Sydney, Melissa, Jean-Marie, Leslie, and others, I will miss you all. My gratitude goes to Deans Colbert, Staton and everyone else in the Graduate Students Office. To my MIT, TPP, NED and non-MIT friends: Thank you for always being there for me; I love you all. Gahaton Peppermint Bebe, much love for the good times.

Last but definitely not the least, I want to thank my family in Nigeria and California for my existence and for their love and support. I am dedicating this thesis to my parents, Mr. & Mrs. Afolayan Jejeloye. You, both, have always been my inspiration.

God bless us all, everyone.

Table of Contents

List of Tables

List of Figures

Nomenclature

ATWS	Anticipated Transient Without Scram
CDF	Core Damage Frequency
EAS	Emergency Alert System
ECL	Emergency Classification Level
EDG	Electric Diesel Generator
EEOP	Earthquake Emergency Operational Plan
EFW	Emergency Feedwater
ELOCA	Excessive Loss of Coolant Accident
EOP	Emergency Operation Plan
EPEDAT	Early Post-Earthquake Damage Assessment Tool
EPZ	Emergency Planning Zone
ESF	Emergency Support Function
FEMA	Federal Emergency Management Agency
GIS	Geographic Information System
GT	General Transients
HAZUS	Hazard U.S.
IE	Initiating Event
LERF	Large Early Release Frequency
LLOCA	Large Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LOSP	Loss of Off-site Power
MLOCA	Medium Loss of Coolant Accident
MMI	Modified Mercalli Intensity
NHRERP	New Hampshire Radiological Emergency Response Plan
NIBS	National Institute of Building Sciences
NGDC	National Geophysical Data Center
NPP	Nuclear Power Plant
NPS	Nuclear Power Station
NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
PCC	Primary Component Cooling Water
PEQIT	Post Earthquake Investigation Team
PGA	Peak Ground Acceleration
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Analysis
PSA	Probabilistic Safety Analysis
RCP	Reactant Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SBO	Station Blackout
SERF	Small Early Release Frequency
SG	Steam Generator

SGTR	Steam Generator Tube Rupture
SLOCA	Loss of Coolant Accident
SSE	Safe Shutdown Earthquake
SSPSA	Seabrook Station Probabilistic Safety Analysis
SSPSS	Seabrook Station Probabilistic Safety Study
SW	Service Water
SWS	Service Water System
SUFP	Startup Feed Pump

A: Introduction and Methodology

Chapter 1: Overview

1.1. Background and Motivation

Earthquakes usually have multiple effects. They involve the major event leading to one or more secondary events such as tsunamis, fire, and failure of dams or nuclear reactors. There are many forms of risks from earthquakes such as economical loss, environmental impact, injuries and death, etc. Earthquakes are a constant threat to the world at large and a significant number of occurrences take place all over the world. Table 1. 1 shows the number of recorded earthquakes in the past decade. According to the National Earthquake Information Center, there is a 100% chance of experiencing an earthquake on any given day somewhere in the world. This is an acknowledgement of the several million earthquakes that occur every year, even though most are so small that it is difficult to locate.

In contrast to a popular misunderstanding that earthquake hazard is a U.S. Pacific Coast problem, earthquakes have been recorded in every state. Earthquake scientists and government agencies charged with planning for natural disasters are focusing on the seismic hazard facing different parts of the country. Some studies have been done to look at the estimates of damage at various levels of seismic event. Some of the first earthquake loss estimation studies were performed in the early 1970's following the 1971 San Fernando earthquake. More recently, there have been investments in earthquake loss estimation methodologies based on geographic information systems (GIS). Two useful references on loss estimation studies are "Estimating Losses from Future earthquakes" (FEMA, 1989)[2] and "Assessment of the State-of-the-Art of Earthquake Loss Estimation Methodologies" (FEMA, 1994)[3]. From these studies, the Federal Emergency Management Agency (FEMA) developed a computer program called HAZUS[11] to estimate damage and loss from different magnitudes of earthquakes. The most recent version of the program, which is used for this project, is HAZUS '99. This GIS-based program is a tool used for estimating future losses from scenario earthquakes. Results are shown in terms of expected seismic consequences that include fatalities, morbidities, damaged buildings and bridges, and fires. The consequence of an earthquake is a combination of direct social losses, direct economic losses, and indirect economic losses. The consequences are also a function of earthquake occurrence probability, direct physical damage, and induced damage.

Nuclear plants designed to be shut safely down during the most severe earthquakes that can be expected to occur during their lifetime. Probabilistic Risk Assessments (PRAs) have become the choice of methodology of performing the safety evaluation of nuclear plants by many countries including the United States, United Kingdom, Germany, Japan, Sweden and a host of others. During the last several years, PRA has evolved to a point where it can be used increasingly as a tool in regulatory decision-making. There have been significant advances in experience with risk assessment methodology since the Reactor Safety Study (WASH-1400)[1] was published in 1975. External events make a significant contribution to the calculated core damage frequency (CDF), dominated by those from earthquakes and fire. Initially, it was thought that seismic events were not large contributors to external initiating events, but following the probabilistic assessments, this frame of thought was found to be erroneous. The methodology for determining the probability of earthquake induced radioactive releases as a result of core melt was developed as part of the overall safety system study for the nuclear power plants.

Estimates of seismic risks produced by nuclear power stations in the United States are needed and based on theoretical considerations and engineering judgment since there is no database due to the fact that no serious accidents were ever induced by earthquakes. Despite this, whenever an earthquake occurs in the vicinity of a nuclear power plant, the public and media's main attention is on the nuclear plant. Some believe the risk from the plant is the worst of their problems. However, this is not necessarily the case. This thesis provides information on the quantification of a seismic nuclear power plant associated risk (using PRA and results from plants in the U.S.), summarizes direct seismic risks (risks posed to the public due to earthquake occurring without the existence of a nuclear power plant in the area), compares both risks and identifies reasonable safety principles upon which to base policies to achieve low levels of overall public risks. Some questions that are used as guides include:

- What are the risks posed directly by seismic events to the public?
- What are the risks posed indirectly by seismic events through a nuclear power plant to the public in its vicinity?
- What policy principles and goals can be enacted to reduce the overall risk to the public?

1.2 Thesis Overview

This thesis consists of four parts.

Part A, which contains Chapters 1, 2 and 3, is an introduction to the project and the methodologies used to achieve our goals. Chapter 1 is a general introduction. Chapter 2 describes the Probabilistic Risk Assessment (including the seismic PRA) methodology for evaluating the risk of a nuclear power plant. Chapter 3 summarizes the methodologies of direct seismic risks and the particular loss estimation program –HAZUS- that was used for this project.

Part B with Chapter 4 presents the case studies that were analyzed and presents the results obtained. It presents the case studies used for the project and the results attained. These include the Seabrook nuclear power plant PRA and HAZUS estimation results.

Part C discusses the safety policy analysis, which is the end goal for the project. Chapter 5 deals with more quantitative aspect of the policy analysis while Chapter 6 discusses qualitative basis for the analysis and presents the result of a workshop of experts.

Part D contains the conclusions that tie everything together and presents conclusions drawn from all the analysis done earlier.

Magnitude of Earthquake (Richter Scale)	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001
8.0–9.9	0	0	0	1	2	3	1	0	2	0	4	1
7.0–7.9	12	11	23	15	13	22	21	20	14	23	14	6
6.0–6.9	115	105	104	141	161	185	160	125	113	123	157	45
5.0–5.9	1,635	1,469	1,541	1,449	1,542	1,327	1,223	1,118	979	1,106	1,318	382
4.0–4.9	4,493	4,372	5,196	5,034	4,544	8,140	8,794	7,938	7,303	7,042	8,114	2,127
3.0–3.9	2,457	2,952	4,643	4,263	5,000	5,002	4,869	4,467	5,945	5,521	4,741	1,624
2.0–2.9	2,364	2,927	3,068	5,390	5,369	3,838	2,388	2,397	4,091	4,201	3,728	1,319
1.0–1.9	474	801	887	1,177	779	645	295	388	805	715	1,028	225
0.1–0.9	0	1	2	9	17	19	1	4	10	5	6	0
No recorded Magnitude	5,062	3,878	4,084	3,997	1,944	1,826	2,186	3,415	2,426	2,096	3,199	749
Total # of Earthquakes in given year	16,612	16,516	19,548	21,476	19,371	21,007	19,938	19,872	21,688	20,832	223,091	64,781

(Source: National Earthquake Information Center, U.S. Geological Survey)[19]

Table 1. 1: Number Earthquakes Worldwide, 1990-2001 (as of May, 2001)

Chapter 2: Probabilistic Risk Assessment (PRA) Methodology

2.1. Introduction

It should be noted that when safety is emphasized rather than risk, the term 'probabilistic safety Assessment' (PSA) is used instead of PRA. These terms will be used interchangeably throughout this thesis. Since the landmark Reactor Safety Study[1], there have been significant developments and PRA techniques have become a standard tool in the safety evaluation of nuclear power plants (NPPs). The PRA of a NPP typically provides:

- *Insights into plant design, performance and environmental impacts*
- *A methodological approach to identifying accident sequences that can follow from initiating events*
- *Systematic and realistic determination of accident frequencies and consequences*
- *An integrated framework for risk-informed decision making*
- *A mathematical tool for deriving numerical estimates of risk for NPPs and industrial installations*
- *Identification of dominant risk contributors and comparison of options for reducing risk*
- *Quantification of uncertainties in safety analysis*

2.2. The Three PRA Levels

There are three levels of PRA used in assessing the risk from a NPP. A level 1 PRA assesses the failures in the plant (accident-frequency analysis) that may lead to core damage. A level 2 PRA performs accident-progression, source-term analyses and analysis of containment response leading to radioactivity release. A level 3 PRA assesses the off-site consequences leading to estimates of public risks. Figure 2. 1 shows the steps taken in carrying out a PRA.

2.3. Steps in PRA Construction[4]-[7]

Step 1 – Initiator List Preparation

The initiators are divided into 2 classes: those for which event tree/fault tree is appropriate and those for which it is not. The former are referred to as internal initiators while the latter are called external initiators (externalities). If dependencies are accounted for by modifying the branching probabilities of the event tree, both the internal and external initiators can be accounted for in the same event tree. For a nuclear power plant, a list of initiating events is accessible in NUREG-1150[4].

Step 2 - System Sequences

After determining the initiators, the operational systems or actions involved in responding to the initiators are determined. The systems shown as A, B, and C in Figure 2. 2 may be ordered in the time sequence that they occur even if dependencies are accounted for in interspersing support systems. Mitigators may also be included in the event tree.

Step 3 – Branching Probabilities

The systems list specifies the systems that need to be analyzed to obtain the branching probabilities of the event tree. Fault trees can be used to evaluate system probabilities for some complex reliable systems. For less reliable systems, plant records usually contain the branching probability.

Step 4 – Damage States

A thorough knowledge of plant and process operations is essential to evaluate which accident sequences lead to particular damage states. These states cluster about specific situations, each having characteristic releases. The maximum number of damage states an event tree is N^S for an N-branch event tree with S number of systems along the top of the event tree. In practice though, simpler trees result from nodes being bypassed for physical reasons; and it is made even easier because many event sequences lead to the same plant damage state (PDS). PDS groups are sequences that are expected to have similar effects on containment response and fission product source terms. Usually, PDSs are grouped into 2 main classes: those that identify when radioactive materials are released to the containment and its status (e.g. intact and isolated, failed or bypass) and when the containment is bypassed or ineffective (e.g. LOCA and SGTR). When dealing with initiators like either internal or external hazards, a new set of distinct PDSs could be defined. However since hazards simply cause dependent failures of plant items, plant model used for internal initiators are usually used for them. Information on PDS can be displayed in the form of a matrix containing the frequencies of each PDS given each initiating event.

Step 5 – Dependency Analysis

This determines which systems depend on other systems and may require repeating the event tree construction to put a support system before the systems affected.

Step 6 – Analysis of Physical Processes

This analysis reveals the amounts of hazardous materials that may be released given the PDSs. Once these physical processes have analyzed and the amounts of release determined, the retention of radioactivity in the plant is calculated.

Step 7 – In-Plant Transport

Thermo-diffusion calculations, using source terms, analyze the movement of hazardous material from compartment to compartment to release in containment. Containment event trees are used in determining the amount, duration and types of hazardous material that leave the containment.

Step 8 – Ex-Plant Transport

The materials leaving the containment are source terms for offsite transport calculations. Codes such as CRAC and CRAC-2 are used to evaluate atmospheric diffusion with different probabilities of meteorological condition. These are used in estimating the radiological health and economy consequences.

Step 9 – Integration of Analysis

Involves calculations of frequency of the various accident sequences and their consequences to the public. Then health effects and monetary damage of each damage state can be determined to compose the plant risk. Risk curves are usually the forms in which results from this level are expressed, usually accompanied by a table of sequences whose frequencies are grouped by release category. The curves are determined as flows: Predicted consequences are evaluated for each combination of a weather sequence and an accident sequence. Associated with this combination is a frequency (the product of the predicted frequency of the accident sequence and the probability of occurrence of the weather sequence). All combinations of a weather sequence and an accident sequence give a probability distribution on the magnitude of the health or other effects, and this probability distribution can be readily shown in cumulative form. Codes containing models for different phenomena (atmospheric dispersion, deposition of airborne material, re-suspension, migration through food chains, pathways to humans and resulting health and economic effects) have been developed for various purposes. These codes may also have a number of different meteorological sampling techniques. Codes range in complexity based on end use and nature of site region. The results from this level can then be compared with corresponding probabilistic safety criteria (if already established). It is also used for impact analysis. Figure 2. 3 is an outline of element of a consequence analysis.

Upon completing this step, an uncertainty analysis is performed to determine the confidence that can be given to the results. More is written about uncertainty analysis later.

Step 10 – Presentation of Results

Upon completion of the risk analysis, PRA results must be presented. It includes a summary containing comparison of risks and identification of main contributors to the risk to people in such a way the general public can understand it. The other part is a technical summary, which gives details like system's importance measure systems, effects of data change and assumptions critical to the conclusions. The last part of the presentation of results includes every detail to the analysis so that a peer can trace and repeat calculations for verification.

2.3. Uncertainty in PRA

Many sources of uncertainty arise at almost every level of PRA. Any discussion of results from a PRA would be meaningless without a discussion of the uncertainties associated with them results. Uncertainties in PRAs are due in part to the stochastic nature of events at a plant, the engineering complexity of the plant and the difficult modeling of the progressions of accidents. Uncertainties in PRA can generally be grouped into: stochastic uncertainty (also known as Aleatory Uncertainty) and state-of-knowledge uncertainty (also known as Epistemic Uncertainty), parametric uncertainty and model uncertainty[8], [9]. In general, aleatory uncertainty refers to the inherent variation of a physical process over many similar trials or occurrences, and epistemic uncertainty refers to both the analyst's state-of-knowledge about the physical processes and the analyst's confidence and ability to measure and model them. Parametric uncertainties are those associated with the values of parameters used in models like the PRA. It is typically characterized by probability distributions, which express the analyst's degree of belief in the parametric values (based on her state of knowledge and the underlying model being correct). Model uncertainties are associated with incomplete knowledge regarding how models used in PRAs should be formulated. These arise from modeling human performance, mechanic failures of structures, and common cause failures for example. Since randomness is a major part of the nature of events such as the occurrence of initiating events and the failure of components, increasing knowledge about these events will not be able to reduce the aleatory uncertainties associated with them. Epistemic uncertainty, on the other hand, can be reduced if more data are gathered or more investigation is carried out on the specific event.

Uncertainty analysis provides estimation of uncertainty in risk results and other intermediate results like radionuclide releases. Performing an uncertainty analysis is important because it adds credibility to the risk assessment and helps in the process of decision-making. Uncertainty analysis can be performed either qualitatively or quantitatively. Due to the subjectivity involved in the quantification of uncertainties in some areas of a PRA (human factor database, treatment of external events, health effects models), some qualitative assessment of uncertainty is required. Examples of methodologies for both qualitative and quantitative uncertainty are presented in detail in some references[5], [8], [9]. Uncertainities grow in number and magnitude as one proceeds from Level 1 to Levels 2 and 3 PRAs. The best developed uncertainty analysis is the one for a level 1 PRA. It includes the estimation on uncertainties in input parameters top event and fault tree models used in describing plant behavior and the propagation of uncertainties through the trees. The uncertainties are evaluated on the assumption of validity and completeness of the fault and event tree models, henceforth are a measure of the uncertainty introduced by an imprecise knowledge of input parameters. A level 2 PRA uncertainty analysis in the accident sequences frequencies is similar to the level 1 analysis and usually analyzed with a subjectivist perspective on uncertainty because the determination of probabilities on branches of containment event trees are based mainly on judgment instead of data. Uncertainty analysis in level 3 PRA addresses uncertainties from the system, containment and consequence analyses. Defense-in-Depth is also used to compensate for completeness issues in a PRA.

2.4. Seismic Probabilistic Safety Assessment of a Nuclear Power Plant

There has never been an earthquake known to have inflicted damage, to lead to core radioactivity release, to a nuclear plant in the United States or elsewhere in the world. In the past decade, only two very destructive earthquakes were close enough to nuclear power plant to elicit concern: the November 1988 Armenian earthquake and the January 1995 Japanese earthquake near Kobe. Both earthquakes only produced minor ground motion at the plant and were well within the design basis. Therefore, the frequency of potential earthquake-initiated core-damage accidents at any nuclear-power station is known from calculations, using real-earthquake data, test data, models of various phenomena, and systems analysis. However, in spite of the lack of actual earthquake experiences, almost all full-scope PSA performed has shown that earthquake-initiated accidents are one of the more important contributors to risk. From the reviews of the seismic individual plant evaluation of external events (IPEEE) of some plants, the Electric Power Research Institute compared the seismic core damage frequencies (CDF) to internal event CDFs and the results are shown in. Earthquakes accident sequences also often appear as important contributors to residual risk in areas where earthquakes are not common. This is typically because in such areas, the attention given to designing NPP against earthquakes is much less than in earthquake-prone area. This makes it obvious that a PSA cannot be complete without examining seismic events.

There are three major steps necessary to evaluate the probability of earthquake initiated core melt[6], [10]:

- Estimation of the ground motion (peak ground acceleration) and uncertainty as functions of annual probability of occurrence
- Estimation of the conditional probability of failure and its uncertainty for structures, equipment, piping, controls, etc., as functions of ground acceleration
- Combination of these estimates to obtain probabilities of earthquake induced failure and uncertainties in estimates to be used in event trees, system models, and fault trees for evaluating the probability of earthquake induced core melt.

The overall seismic PSA methodology can be divided into six (not set in stone, more or less than six could be used) sub-methodologies. They are the seismic hazard analysis, seismic local ground motion and building motion analysis, seismic failure mode and fragility, seismic PSA system analysis, and plant response and offsite releases and consequences analysis. Figure 2. 5 shows a representation of the development of a seismic PRA.

2.4.1. Seismic Hazard Analysis

This involves the analysis of the frequency of earthquakes of various magnitudes at a given site and the spectral shapes of the motion from these earthquakes. Its results are intrinsically probabilistic in nature. Following below is a summary of steps involved in this methodology.

- The first step involves identifying and characterizing the seismic sources (identified faults, point sources or areas assumed to have spatially uniform occurrences of earthquakes [source zones]) in the vicinity of the NPP site. The goal here is to determine the frequency of earthquakes of different sizes (includes the distribution of magnitudes, the depth and physical location and extent of the source and various other physical parameters like annual frequencies of occurrence of measured smaller earthquakes) from each identified source or source zone. A major problem in this step is that, in regions without active faulting or well-identified sources, different experts provide different maps that characterize different zonation schemes based on different interpretations of the little data available.
- The second step involves determining the earthquake recurrence relationship for each source or source zone. Some issues accounted for here are the historical seismic activity rate, the distribution of earthquake magnitudes, the lowest magnitude of concern for the given source, the distribution of depths of earthquake magnitudes, and so on. Models for recurrence relationship range from simple to complex. Limited knowledge of spatial distribution of damage, which cannot be easily transformed into a more scientific parameter like magnitude, is part of the

problem in understanding large earthquakes. Some of the models also require an upper bound cutoff (usually uncertain) on the magnitude otherwise they would have to deal with allowing a finite chance of earthquakes of infinite energy release.

- The third step is figuring out the ground-motion attenuation relationship, i.e., associating a motion vs. distance relationship with each magnitude. Motion parameters used (even though not perfect) include local spectral acceleration, peak ground acceleration, or spectral velocity. Selecting an attenuation model and a ground response spectral shape are two important issues that are sometimes combined in a model to directly attenuate different frequencies differently. There are enough strong-motion earthquake records to provide actual data attenuation for some areas like seismic active areas in parts of the western U.S., while for less active seismic areas like the eastern parts of the U.S., the strong motion record is usually absent. Therefore theoretical models are often used, based partially on data from high seismic areas. Selecting a ground response spectral shape is an area of uncertainty. Site-specific spectrum is developed if there is considerable soil amplification.

- The final step of the hazard analysis involves producing the 'hazard curves' themselves. They are usually expressed in terms of the annual frequency of exceedance vs. a ground motion parameter like the peak ground acceleration or a spectral acceleration.

2.4.2. Seismic Local Ground motion and Building Technology Analysis

The purpose here is to figure out the local motion at the location of each significant structure or component necessary for the safety of the power plant. This analysis usually begins with a family of earthquakes motions being postulated to arrive at the local site from afar or below the site. Some items are located at the ground level, while others are at different elevations. Floor spectra are needed for the latter items, for each elevation in each important building, to represent the seismic input at the base of each component or structure. Firstly, the ground response frequency spectrum at a site (function of distance from the earthquake source, the size of the earthquake and local subsurface condition. Generic broadband spectra have usually been used in PSAs. A structural model for the buildings to show the transmissions of input seismic motion from the foundation to any given location and elevation needs to be developed unless it already exists in original designs or safety analysis report. It is also typical to use linear dynamic analysis for the structure and account for non-linear effects by estimating the inelastic energy absorption capacity of each component, so that the response for the equipment item represents the floor spectrum modified to account for how each equipment item responds in frequency space.

2.4.3. Walkdown Analysis

A well-planned and effectively executed walkdown helps in developing vital information about the plant configuration, specific spatial relationships, anchorages and other features that cannot be found any other way. A walkdown team usually consists of expertise from these areas: seismic-fragility-analysis, system-analysis, and plant operations/maintenance. These various groups should be working together throughout the seismic-PSA effort. The documentation of the walkdown's findings is an important aspect of the process. This is not just for archival reasons, but also more importantly because the documentation is needed by both the seismic-capacity and systems-analysis engineering teams.

2.4.4. Seismic Failure Mode and Fragility Analysis

This is the step where the seismic capacity of individual structures and equipment components are calculated; and eventually the fragility curve for each item and the correlations among them are calculated. This analysis is intrinsically probabilistic because it produces a probability of failure as a function of earthquake size. There are two different aspects of this analysis, the definition of ***failure*** and the determination of ***fragility***. The decision on what a failure means is made by the structural analyst (and agreed upon by the system analyst) on a case by case basis, taking into account the specific safety equipment and safety function that would be vulnerable. For a structure, failure would usually be severe buckling or collapse that would compromise the safety equipment within the structure, or collapse in which the structure could fall onto and damage equipment. Failure does not include minor structural damage. For an item of equipment, failure means the inability to perform its safety function e.g. inability of a pump to move water, of a valve to close or open, etc. Failure here can sometimes involve short-term phenomenon involving no lasting damage. The definitions are highly individualized for specific equipment items. Functional structure failure can be considered when the inelastic deformations under seismic loads are estimated to be sufficient to potentially affect the operability of safety-related equipment attached to the structure. Failure modes as such represent a conservative lower bound on the seismic capacity considering a larger margin of safety against collapse exists for a nuclear structure. There is also the potential for soil failure modes, e.g. liquefaction, toe bearing pressure failure and base slab uplift. Buried structures such as pipes and tanks can be open to failure due to lateral soil pressures. Consideration should also be given failure due to impact of another structure or component because of a seismic event.

The fragility of a component is defined as the conditional probability of its failure as a function of a response parameter, usually an acceleration parameter in seismic PSAs. A family of fragility curves is usually generated and the curves are characterized mathematically by lognormal expressions, anchored to median values and using various uncertainty parameters to capture both variability from randomness and lack of knowledge. Three types of information that can be used to develop fragility curves are data from real earthquake experience, test data, and analysis. The entire fragility curve for any component and the uncertainty in that curve can be expressed in terms of the best estimate of the median ground acceleration capacity, B, times the product of two random variables. Therefore, the ground acceleration, A, corresponding to failure is given by[5]:

$$A = B\varepsilon_R\varepsilon_U$$

Where ε_R and ε_U are random variables with median of unity. They represent, respectively, the randomness about the median and the uncertainty in the median value.

2.4.5. Seismic-PSA Systems Analysis

The objective of this analysis is to determine which core damage accident sequences may result and the core damage frequencies for each sequence given the equipment that would be damaged by the postulated earthquake. Seismic PSAs typically identify not only accident sequences involving one or more seismic induced failures, but also sequences involving a combination of seismic failures, human errors, and non-seismic failures as random failures.

These latter types of accident sequences are as important overall as accident sequences involving only seismic failures.

The results of the seismic fragility analysis, taken into account with issues like the likelihood that other vital equipment might be out of service because of testing, maintenance etc., possible correlation among failures, and the procedures used by the operators are used by the systems analyst for this evaluation. At the center of the work is the development of one or more accident sequenced event trees, which contain the various functions or systems needed for safe shutdown, recovery actions, possible operator prevention, etc. The success-or-failure numerical values on the event tree branch points are then worked out using either data or input from fault tree. A, previously completed, internal initiators PSA can be used for vital information such as the random failure data, the operating crew's procedures and the support system matrix. If not previously completed, this information must be developed newly. The outcome of the systems analysis is the numerical value of core-damage frequency for each of several earthquake sizes.

2.4.6. Seismic-PSA Consequence/Release Analysis

The purpose is to evaluate, for various postulated earthquake sizes associated with various probabilities of core damage, the conditional probability that the postulated accident will evolve into a radiological release event. There are differences in the conditional probability calculated from one postulated core-damage accident sequence to the other. Therefore, each sequence deserves separate treatment, depending on which items of safety equipment were damaged by the earthquake, which equipment failed from other causes, etc. The size of the release will depend on how the phenomena develops both within the primary system, outside the vessel, and in the containment after core damage begins, how ex-plant radiological dispersion phenomena develop and how sheltering and evacuation are accomplished.

Seismic consequence analysis differs from that of internal events because the earthquake may influence some parameters of the consequence analysis. An earthquake can disrupt the communication network and the evacuation routes. It can also affect the sheltering structures in the vicinity of the nuclear power plant thereby affecting the assumption that people can get shelter from gamma rays. People may also act differently in the presence of an earthquake and reactor accident than if only a reactor accident occurred thereby making the distribution of population exposed to radiation in a seismic reactor accident different from that for internal events. In recent seismic risk studies where the level 3 PRA was performed, there has not been a difference in the consequence modeling for seismic and internal events. Some analysts justify that the large uncertainties assigned to the parameters of the consequence models are assumed to cover the differences. In a later section in this thesis, there will be an analysis to check whether this is true or not.

2.4.7. Uncertainty in Seismic PSA[5], [10]

The main sources of uncertainty in a seismic risk assessment are from incomplete data and analytical models. In the seismic hazard analysis, there are parametric uncertainties in the configuration, upperbound magnitude and epicentral intensity of a seismic source and a cutoff value for the effective peak ground acceleration. These depend on specific site region and therefore cannot be ordered according to their contributions to the totals uncertainty. There are uncertainties also in:

- Seismic input and description of dynamic behavior of the soil, structures and systems.
- Material properties such as inelastic energy absorption capacity and strength, definition of failure modes, the use of engineering judgment and generic data and lack of fragility test data on equipment.
- The incomplete identification of all potential accident sequence, modeling of dependent component failures and lack of data on correlation between component capacities.
- Lack of adequate models to predict the effects of large earthquakes on parameters of consequence analysis models, e.g. evacuation time, population distribution and public response.

Two general approaches are used in seismic risk studies so far to propagate uncertainties. One method is used if the event is analyzed with simplified plant level fault trees. Propagation of the uncertainties is achieved by assigning probability distributions for each component-failure frequency in the Boolean expressions. Usually, a family of curve for plant-level fragility for core melts and for each release category is obtained. Integration (accomplished using Monte Carlo error propagation or other statistical techniques) over the hazard-curve family yields probability distributions for core melt frequency and the frequency of each release category. This method is also used to obtain a family of risk curves in the consequence analysis. The second approach involves using the best estimates of the parameters of the seismic hazard, fragility, response, system, sequence, and consequence analyses. This also helps in identifying dominant accident sequences. Significant parameters are also identified by carrying out sensitivity analysis with different parameters. Then the dominant sequences are used to carry out risk assessment many times, each time with a separate set of values of significant parameters. These sets are sampled from the probability of distributions of the parameters. By performing this a number of sufficient times, the probability distributions of core melt frequency and frequencies of each release category and exceedance of damage are obtained. Due to the large uncertainties in a seismic PRA, it is important that the uncertainties be treated consistently and be propagated throughout the analysis in order to quantify the total uncertainty in the plant risk

Plant	EPRI or	LLNL	Total CDF/RY	Seismic % of Total CDF
Beaver Valley 1	9.10E-06	2.46E-05	2.14E-04	4.25
Beaver Valley 2	5.53E-06	2.33E-05	1.92E-04	2.88
Calvert Cliffs 1& 2		1.29E-05	2.40E-04	5.38
Catawba 1& 2	1.60E-05		5.80E-05	27.59
D. C. Cook 1& 2	3.20E-06	1.00E-05	6.26E-05	5.11
Diablo Canyon 1& 2	4.20E-05		8.80E-05	47.73
Hope Creek	1.06E-06	3.60E-06	4.63E-05	2.29
Indian Point 2	1.30E-05	1.50E-05	3.13E-05	41.53
Kewaunee	1.10E-05	1.30E-05	6.65E-04	1.95
La Salle 1& 2	7.60E-07		4.74E-05	1.60
McGuire 1& 2	1.10E-05		4.00E-05	27.50
Millstone 3	9.10E-06		5.61E-05	16.22
Nine Mile Point 2	2.50E-07	1.20E-06	3.10E-06	8.06
Oyster Creek	3.62E-06	6.36E-06	3.69E-06	98.10
Palisades		8.90E-06	5.07E-05	17.55
Point Beach 1& 2	1.40E-05	1.30E-05	1.15E-04	11.30
Salem 1& 2	4.70E-06	9.50E-06	6.25E-05	7.52
San Onofre 2& 3	1.70E-05		3.00E-05	56.67
Seabrook	1.08E-05	1.30E-04	4.57E-05	23.63
South Texas Project 1& 2	1.90E-07	2.20E-05	4.27E-05	0.44
Surry 1& 2	8.20E-06		1.17E-03	0.70
TMI 1	3.21E-05	8.43E-05	4.49E-05	71.49

Table 2. 1: Seismic CDFs For Some U.S. Plants

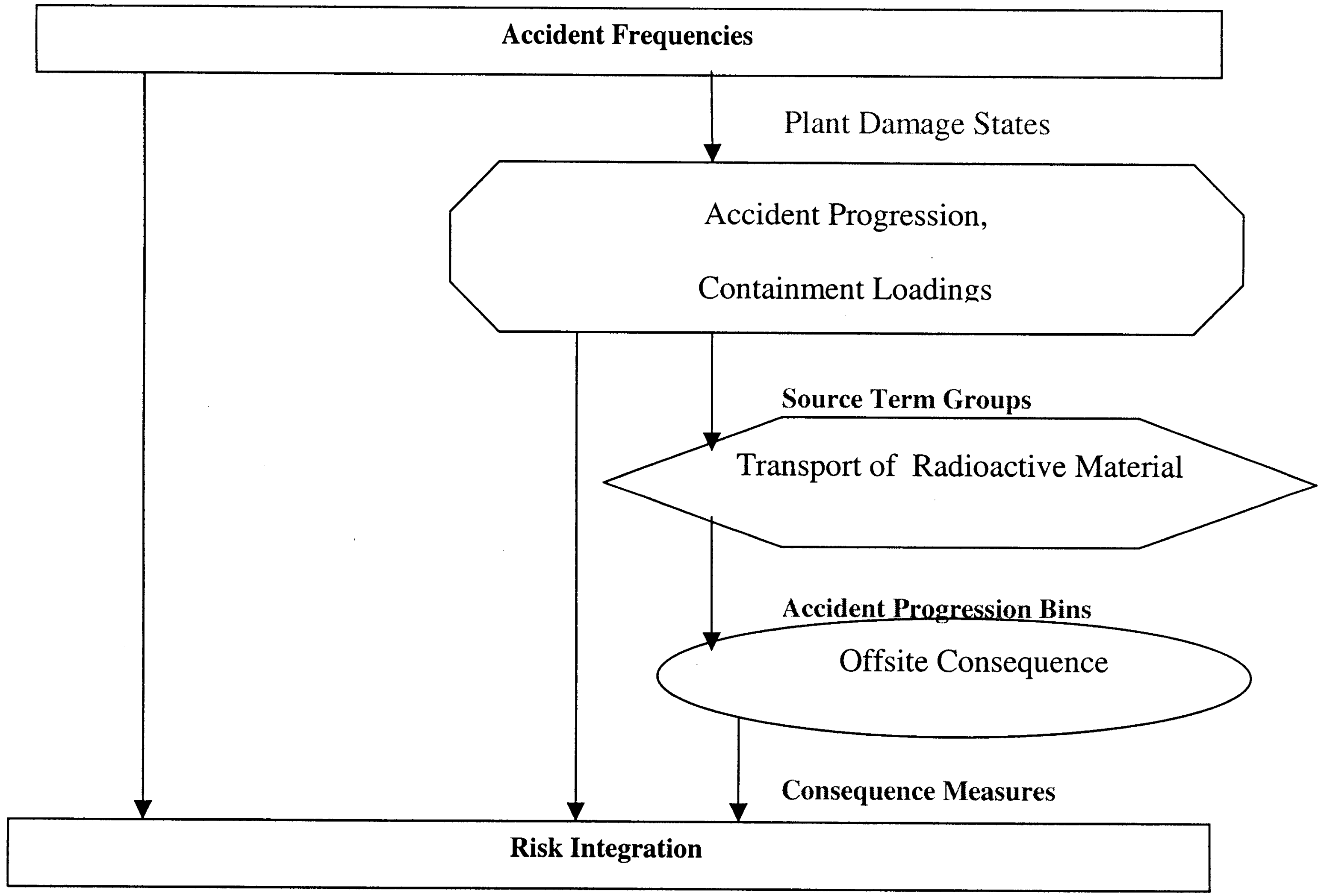


Figure 2. 1: Outline of a Probabilistic Risk Assessment Structure

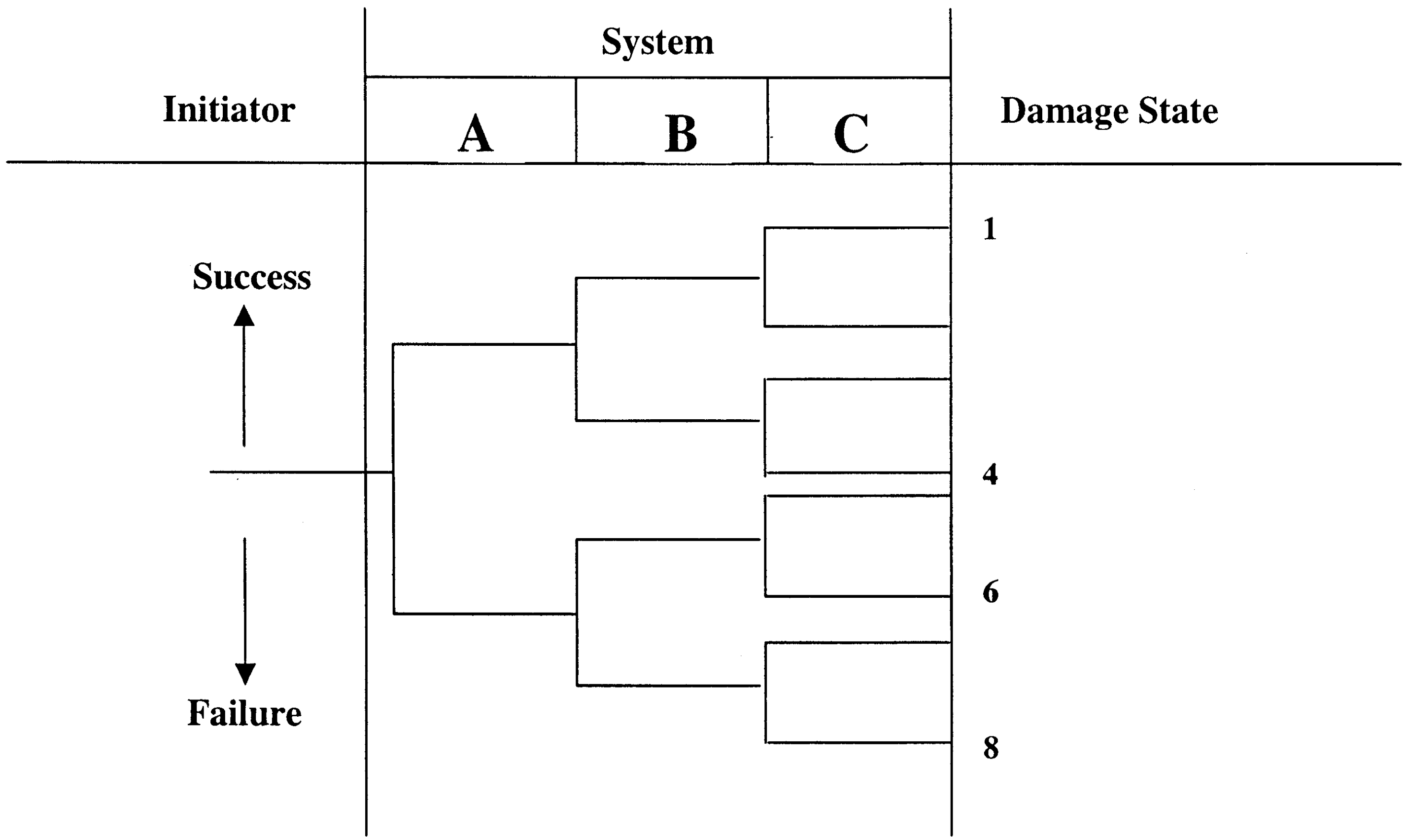


Figure 2. 2: Event Tree Illustrating the Organization of an Analysis

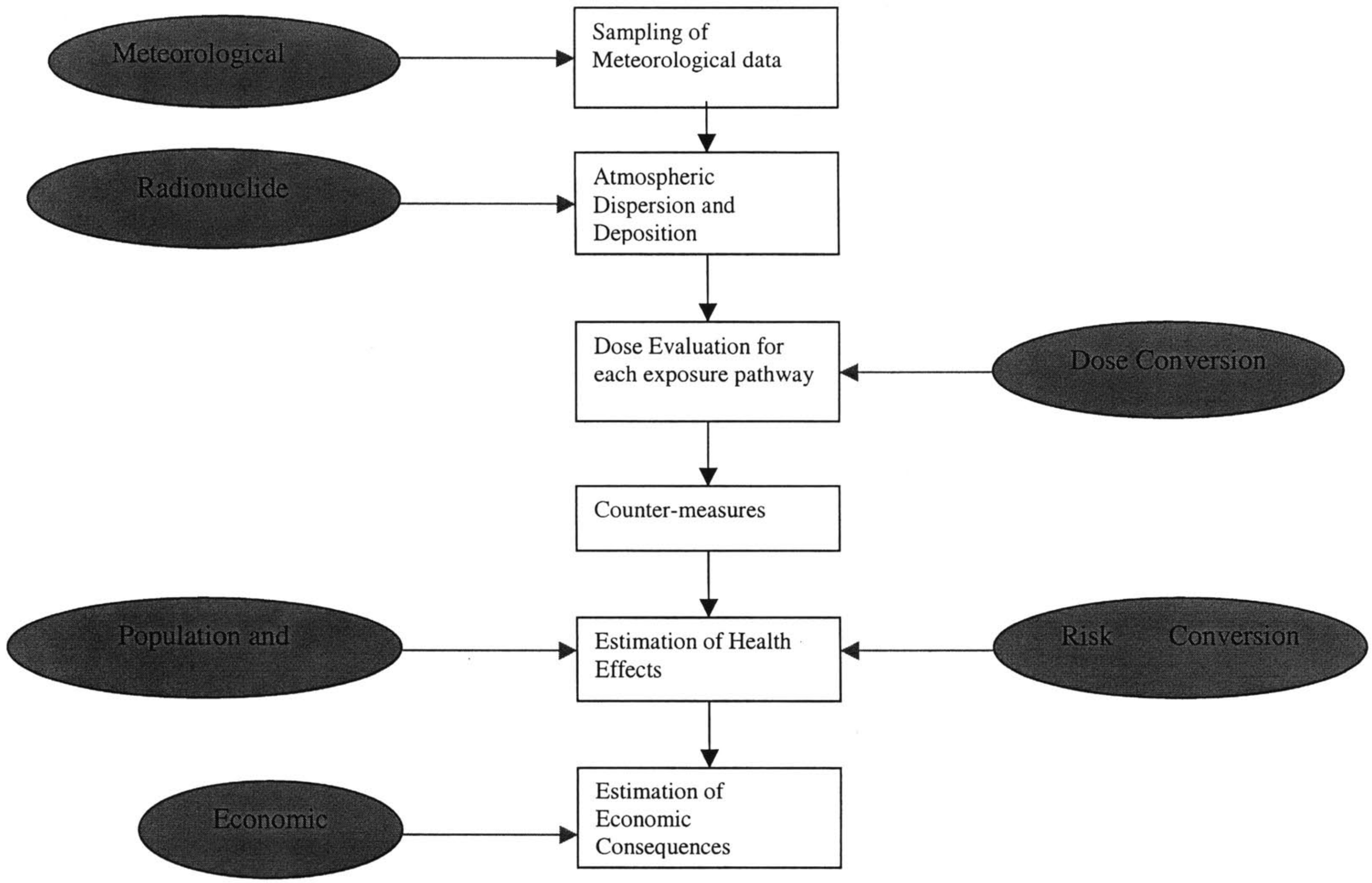


Figure 2. 3: Basic Elements of Probabilistic Consequence Analysis

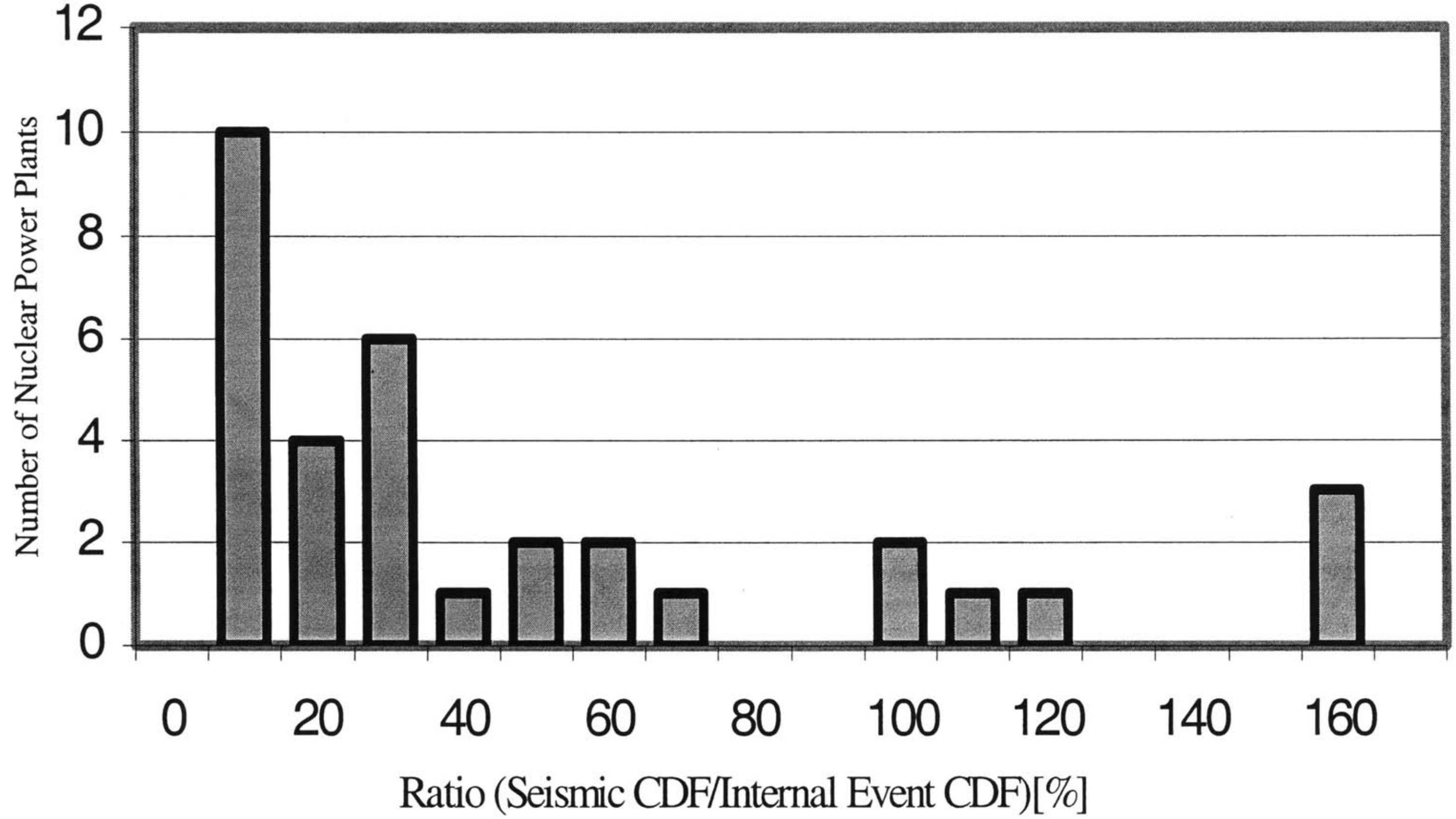


Figure 2. 4: National Nuclear Power Plant Frequency Distribution of Seismic to Internal Event CDF (Source:- EPRI)

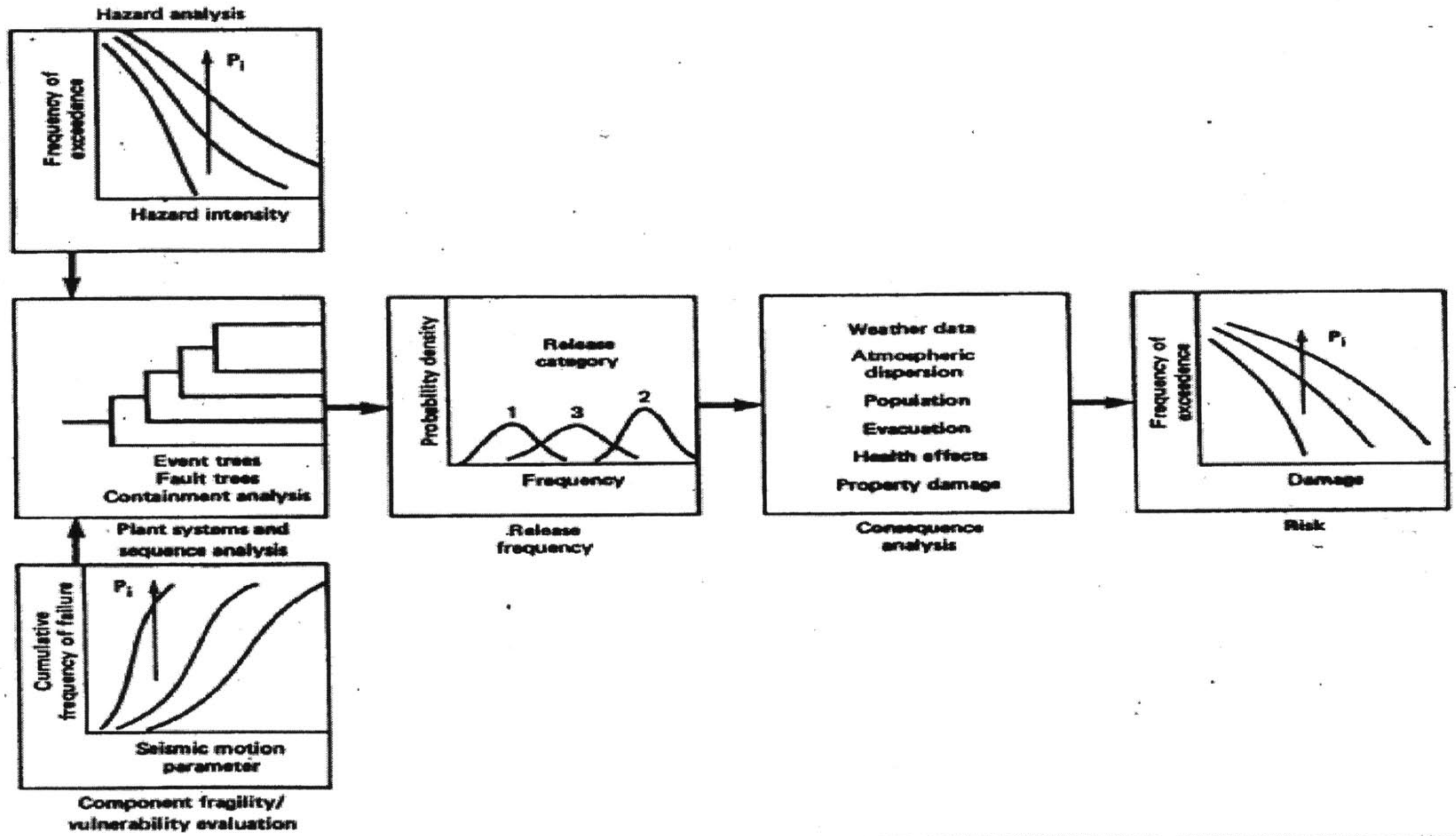


Figure 2. 5: Risk Assessment Methodology for External Events[5]

Chapter 3: Seismic Risk Methodology

3.1. Methodologies For Estimation of Direct Seismic Consequences

Earthquakes may result in multiple impacts. The major event may lead to one or more secondary events such as tsunamis, fire, and failure of dams or nuclear reactors. There may be many forms of losses from earthquakes such as economical loss, environmental loss, injuries and death, etc. In this initial phase of the project, risks are interpreted as the loss of lives and injuries. In contrast to a popular misunderstanding that earthquake hazard is a U.S. Pacific Coast regions' problem, earthquakes have been recorded in every state. Earthquake scientists and government agencies charged with planning for natural disasters are focusing on the seismic hazard facing various parts of the country.

Studies of risks from a seismic event have been done in the past, in the form of regional loss studies usually sponsored by either government agencies or the insurance industry. These studies evaluate damage and casualties by past earthquakes. They also take many quantitative measurements and they investigate qualitative effects from those earthquakes. These have been helpful to many seismologists, geologists, geophysicists, and other scientists and engineers, as well as public officials for prediction, preparation, and mitigation of future earthquakes. Some of the first of these earthquake loss estimation studies were performed in the early 1970's following the 1971 San Fernando earthquake. These studies cover some areas well and there is usually uniformity in the approaches used. Some areas are not covered well. These areas of loss estimation are still evolving and are either ignored or addressed in a qualitative manner. Most notable of these is indirect economic impact. More recently, there have been investments in earthquake loss estimation methodologies based on geographic information systems (GIS).

Loss estimations processes usually begin with potential hazards and inventory leading to the calculation of direct and induced physical damages and culminating in the estimation of losses and social impacts. Figure 3. 1 and Figure 3. 2 show components of an earthquake loss estimation.

3.2. Hazard U.S. (HAZUS) Methodology

The Federal Emergency Management Agency (FEMA) has developed a computer program, HAZUS, in cooperation with the National Institute of Building Sciences (NIBS) to estimate damage and loss from different magnitudes of earthquakes. The most recent version of the program, which was used for this project, is HAZUS '99. HAZUS is capable of using two separate geographic information systems (MapInfo® and ArcView®)[11] to map and display ground shaking, the pattern of building damage, and demographic information about a community. This GIS-based program is a tool used for estimating future losses from scenario earthquakes. Results are shown in terms of expected seismic consequences that include fatalities, morbidities, damaged buildings and bridges, and fires. The consequence of an earthquake is a combination of direct social losses, direct economic losses, and indirect economic losses. The consequence is also a function of earthquake occurrence probability, direct physical damages, and induced damages.

The overall framework of the methodology for HAZUS is similar to the methodology shown in Figure 3. 1. There are three different levels of analyses. The simplest type is the default data analysis. This level of analysis requires minimal effort by the user and the estimates are crude and used only as initial loss estimates to determine which questions warrant more detailed analyses. The next level of analysis is the user-supplied data analysis, which is the most commonly used level of analysis. This requires more extensive inventory data and effort by the user. It provides best estimates of earthquake damage/loss using the standardized methods of analysis included in the methodology. The most detailed analysis is accomplished through advanced data and models, incorporating results from engineering and economical studies that are not included within the methodology. But even with near perfect data, predictive methods are approximate and can be subject to significant uncertainties.

Various magnitudes of earthquake can be selected on HAZUS. The Richter Magnitude Scale is commonly used to define earthquake magnitudes. Developed by Charles F. Richter of the California Institute of Technology in 1935, the magnitude of an earthquake is determined from the logarithm of the amplitude of waves recorded by seismographs. An earthquake's intensity refers to the effect an earthquake has on the earth's surface. The most commonly used scale is the subjective Modified Mercalli Intensity Scale, which was developed in 1931 by the seismologists Harry Wood and Frank Neumann. The intensity scale consists of a series of certain key responses such as people awakening, movement of furniture, damage to chimneys, and finally total destruction. When the size and the location of a scenario earthquake are selected, HAZUS estimates the expected consequences based upon probabilistic information already built-in the program. It gives ranges of peak ground acceleration in terms of the gravitational acceleration due to gravity (g). Typically, the peak ground acceleration decreases away from the epicenter. But the peak ground acceleration is not just a function of distance away from the epicenter. Among other factors, it is a function of the ground conditions. The looser and softer grounds can enhance the motion of the ground, therefore giving higher peak ground acceleration than harder and firmer grounds. Table 3. 1 shows a relationship between the Richter Scale and peak ground acceleration. In considering damage and losses due to earthquakes and their methodologies for consequence estimates, there are areas of particular interest. These are fires following earthquakes, building damage including essential and residential buildings, roads and bridges damage, debris generated by earthquakes, and casualties. Of these, the area of most interest is a number of casualties. The consequence of an earthquake is a combination of direct social losses, direct economic losses, and indirect economic losses. The consequence is also a function of earthquake occurrence probability, direct physical damage, and induced damage. The following equation expresses how the losses and damage integrate to give overall consequence.

$$\text{Direct Seismic Consequence} = \left\langle \begin{bmatrix} DSL \\ DEL \\ IEL \end{bmatrix} \right\rangle = \sum_i (EQ)(DPD)(ID) \times \begin{bmatrix} DSL \\ DEL \\ IEL \end{bmatrix} = \Pr[EQ_i \cdot DPD_i \cdot ID_i] \times \begin{bmatrix} DSL \\ DEL \\ IEL \end{bmatrix}$$

Where,

DSL—Direct Social Losses such as casualties, displaced households, and short-term shelter needs,
DEL—Direct Economic Losses due to damaged buildings and lifeline systems,
IEL—Indirect Economic Losses, or long-term economic losses,
EQ—Earthquake,
DPD—Direct Physical Damage to general building stock, essential and high potential loss facilities (emergency response facilities, etc.), and lifeline systems (transportation, utility, etc.), and,
ID—Induced Damages such as inundation, fire, hazardous materials release, and debris.

The equation above basically provides an overview of how the three, losses are evaluated by HAZUS. The direct seismic consequences provided by HAZUS are conditional consequences. The probabilities of the earthquake (EQ), direct physical damage (DPD), and induced damage (ID) [of any particular magnitude of earthquake (i)] are factored into the various losses. Therefore, all three losses evaluated by HAZUS, are conditional on the occurrence of these three events (EQ, DPD, and ID).

HAZUS also makes casualty estimates at four different severity levels. The severity levels are defined as follows.

- Severity Level 1—Injuries will require medical attention but hospitalization is not needed.
- Severity Level 2—Injuries will require hospitalization but are not considered life threatening.
- Severity Level 3—Injuries will require hospitalization and can become life threatening if not promptly treated.
- Severity Level 4—Injuries are fatal.

HAZUS comes with some limitations in terms of uncertainties. These uncertainties come from incomplete scientific knowledge on earthquakes and how they affect structures. Approximations and simplifications made to complete analyses contribute partly to the uncertainties. Also accurate HAZUS results can usually be obtained when applied to a class of buildings or facilities, and not to a particular building or facility. The Eastern United States ground motion characteristics uncertainty is high, so conservative treatment of this uncertainty overestimates the losses in this area.

More detailed description and technical specifications of HAZUS are provided in the technical specification manual for the HAZUS program[11] and MIT-NSP-TR-005 (Analysis of Direct Seismic Risks and their Effects on Existing Emergency Response Plans Near a Nuclear Power Plant)[12].

Magnitude (Richter Scale)	Earthquake Effects (in United States)	Max PGA Range(g)	Class	Average Annually in United States
2.5 or less	Usually not felt, but can be recorded by seismograph.	0~0.05	Very Minor	247
2.5 to 5.4	Often felt, but only causes minor damage.	0.05~0.50	Minor~Moderate	1580
5.5 to 6.0	Slight damage to buildings and other structures.	0.50~0.80	Moderate	44
6.1 to 6.9	May cause a lot of damage in very populated areas.	0.80~1.68	Strong	6
7.0 to 7.9	Major earthquake. Serious damage.	1.68~3.80	Major	0.56
8.0 or greater	Great earthquake. Can totally destroy communities near the epicenter.	>3.80	Great	0.043

Table 3. 1: Earthquake Magnitudes

(Source: Earthquake Loss Estimation Methodology HAZUS99 Technical Manual)

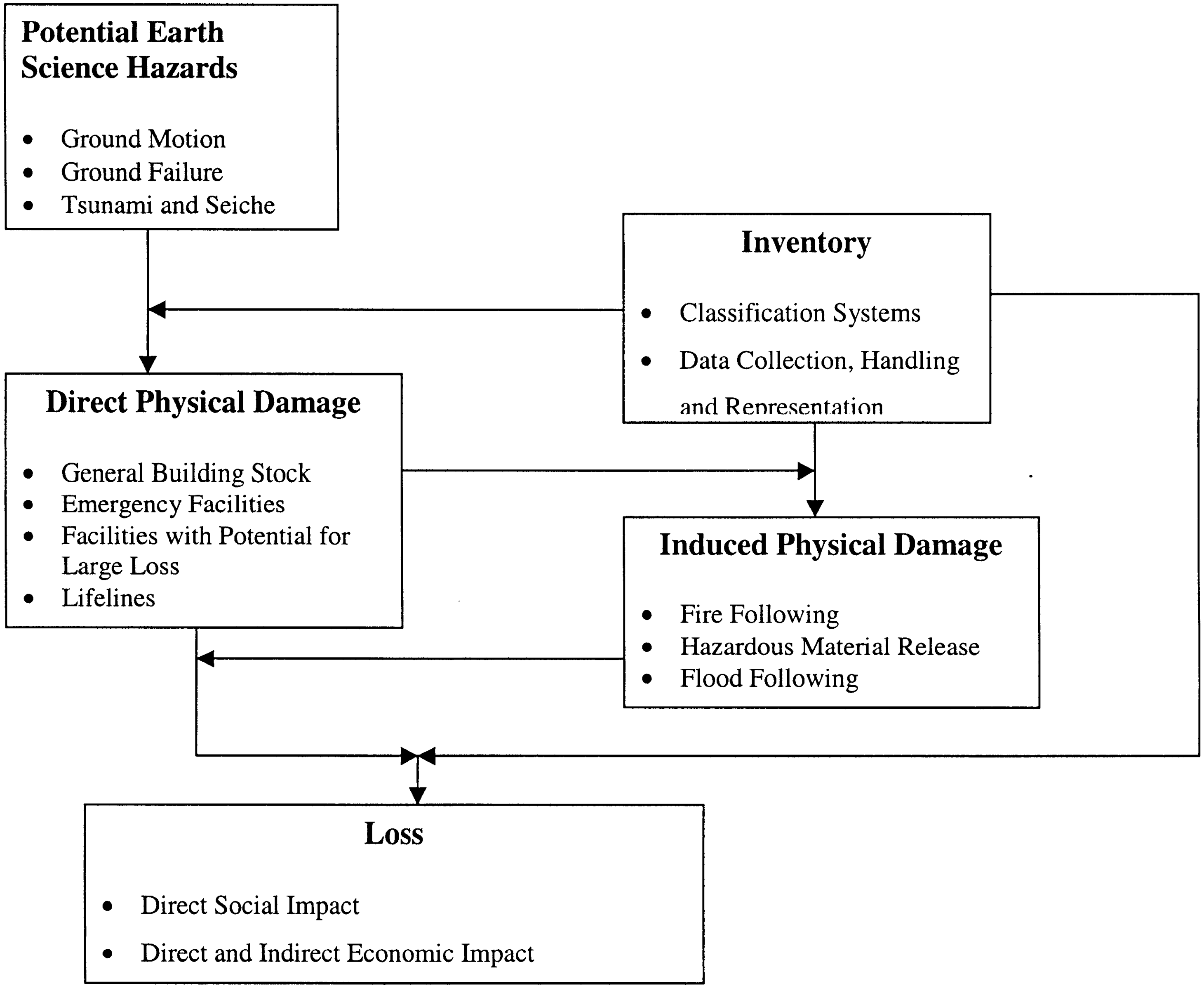


Figure 3. 1: Components of Earthquake Loss Estimation

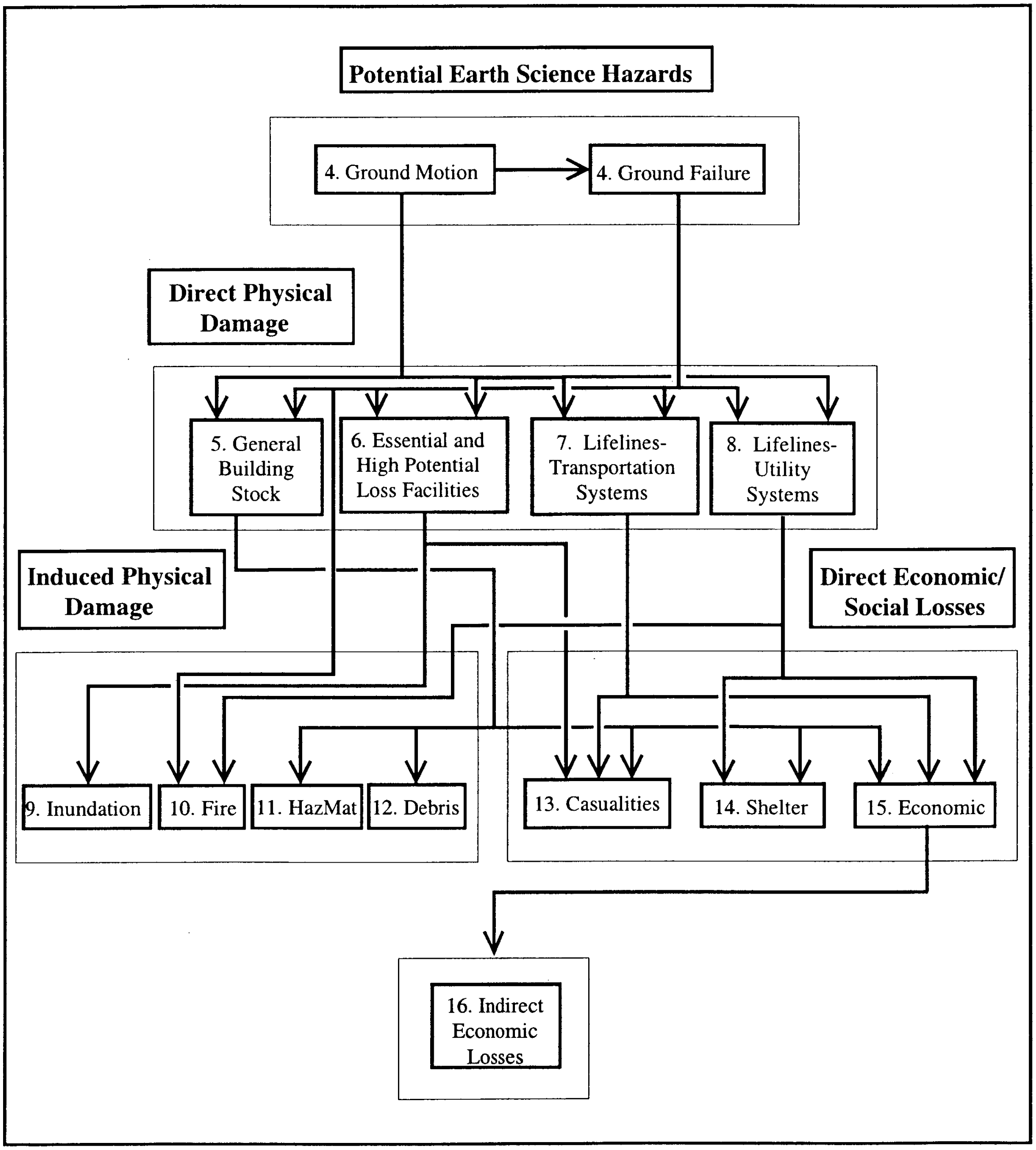


Figure 3. 2: Flowchart of the Earthquake Loss Estimation Methodology (HazMat: Hazardous Materials)

(Source: Earthquake Loss Estimation Methodology HAZUS99 Technical Manual)

B: Assessment and Results

Chapter 4: Case Study of Seabrook Power Station and the Greater Boston Area

4.1. Site Description of Seabrook Nuclear Power Station

The Seabrook power station is located on the western shore of Hampton in Rockingham County, in the town of Seabrook, New Hampshire, approximately eleven air miles south of Portsmouth, New Hampshire and two miles west of the Atlantic Ocean. It is located 40 miles south/southwest of the site is the center of the Boston metropolitan area. The site has an area of approximately 896 acres and is owned by Public Service of New Hampshire. Bounded on the north, east and south by marshland, access to the site is from the west via two roads, both entering from the U.S. Route 1. The site is located on a peninsula in the marsh composed of quartz diorite and includes quartzitic bedrock locally overlain, prior to construction, by a thin veneer of glacial and post glacial soils. All of the safety related structures are built on sound bedrock or on concrete fill or controlled backfill extending to sound bedrock. Only low to moderate seismic activity has been experienced in this region. The largest earthquake in the site area was the intensity VIII on the modified Mercalli (MM) scale event of 1755. Seismic events of various intensities are analyzed for the PRA and frequencies of occurrence and effects on plant safety systems are analyzed. Due to its location next to the ocean, the site is subject to an effect of land-water interaction that affects atmospheric dispersion and risk.

Two emergency planning zones (EPZ) have been designated for Seabrook. They are a 10-mile radius plume exposure pathway zone and a 50-mile radius ingestion exposure pathway zone. These are the areas for which planning is recommended to assure that prompt and effective actions can be taken to protect the public in case there is an accident. The people around the site live in several towns and cities located primarily to the north, south, and west of the site. The closest city that contains more than 25, 000 people is Portsmouth, New Hampshire (12 miles north / northeast). The 1983 resident population distribution within 50 miles is presented in Figure 4. 1. The population distribution has not changed much since then. In addition to the permanent residents, there is a transient population that comes to use the local beaches for recreation during the summer. Peak total transient population (a weekend in July) is estimated to be about 84,000 within 5 miles of the site and 117,000 within 10 miles of the site. A dog track with a peak capacity of 7,500 is located about 2.25 miles west of the site. There are several industrial firms, educational and medical facilities in the area. The nearest hospital is Amesbury Hospital about 5.5 miles southwest of the site.

The major highways in the vicinity include U.S. Route 1, Interstate 95 (I-95), Route 1A, and Route 51. Other roadways include I-95, the New Hampshire Turnpike passes 1.6 miles to the west of the site. It is the most traveled roadway in the area. Airports in the vicinity include Hampton Airport (small and privately owned), about 4.25 miles away and two small private airports in Newbury Park about 5 miles away. The Salisbury Branch of the Boston and Maine Railroad serves the area within 5 miles of the site.

4.2. Description of the Seabrook Power Plant

The Seabrook Station consists of one unit, which employs a four-loop pressurized water reactor (PWR) and auxiliary systems designed by Westinghouse Electric Corporation. The turbine-generators are supplied by General Electric Company and the balance of the plant is designed by United Engineers and Constructors, Incorporated. The unit is designed to produce 1, 150 MWe. The layout of the major structures on the site is shown in Figure 4. 2. The major structures are the containment structure, containment enclosure, primary auxiliary building, fuel storage building, control building, diesel generator building, turbine-generator building, emergency feedwater pumphouse, main steam and feedwater pipe chase, ocean intake and discharge structures, administration and service building, ultimate heat sink cooling tower, fire pumphouse, etc.

The primary auxiliary building houses most of the auxiliary systems for reactor coolant system (chemical and volume control, primary component cooling, safety injection, residual heat removal, and containment spray). The containment building (reinforced concrete structure lined with a welded carbon steel plate with a 4-foot thick concrete basement floor) completely encloses the reactor, the coolant system, steam generators, portions of engineered safety features systems, and supporting systems. It is designed to be able to withstand credible loads including environmental and abnormal loads. There is also the containment enclosure building, which encloses a small volume around the outside of the containment and entraps, filters, and discharges leakage from the containment.

4.3. Results of the Seabrook NPS PRA

The Seabrook NPS's PRA was used because of the convenience of access and its proximity a major city, Boston. At the beginning of the project, an older PRA (1983) was utilized. It had all three levels of PRA. However, a more recent Seabrook Station Probabilistic Safety Study (SSPSS-1999) was obtained. This PRA was, however, mainly a level 1 and level 2 PRA. Its more recent and more accurate results were incorporated with the 1983 PRA to obtain the results needed.

The key results of the current Seabrook Station Probabilistic Safety Study (SSPSS-1999) are quantitative estimates of risks. The annual average Core Damage Frequency (CDF) for the SB99 model – including internal and external accident initiating events (and subject to average maintenance) is **4.75×10^{-5} per reactor year**. The basis for the core damage risk can be understood by examining the significant risk contributors. Almost 73% of the CDF total is due to the Reactor Coolant Pump (RCP) seal Loss Of Coolant Accident (LOCA) sequences-from station blackout sequences and sequences with loss of Primary Component Cooling Water (PCC). Transients with loss of decay heat removal (loss of emergency feedwater (EFW), loss of startup feed pump (SUFP) and loss of feed-and-bleed cooling) account for 13% of the total while conventional LOCAs (excluding RCP seal LOCA) account for 7% of the total. Anticipated Transient Without Scram (ATWS) sequences account for 8% of the total CDF with the majority of the anticipated transient without scram (ATWS) risk arising from the failure of long term reactivity control. The ten most important event sequences that represent about 25% of the CDF total, are summarized below:

1. LOSPW – Loss of off-site power (LOSP) due to severe weather, with failure of the electric diesel generators (EDGs) resulting in station blackout (SBO);
2. L2SWA – Loss of both trains of service water (SW), starting with Train A;
3. L2SWB - Loss of both trains of service water (SW), starting with Train B;
4. FCRCC – fire in the control room, with failure of the PCC system, resulting in RCP seal LOCA;
5. FTBLP – Fire in the turbine building, causing LOSP, with independent failure of the EDGs, and failure to recover alternating current (AC) power;
6. E7T – Earthquake induced station blackout (peak ground acceleration of 0.7g);
7. L2SWA-RCP – Same as #2, except for the size of the RCP seal LOCA;
8. L2SWB-RCP – same as #3, except for the size of the RCP seal LOCA;
9. ASEIS – Earthquake-associated damage to control rod assemblies, resulting in an ATWS (because of the size of the earthquake, operator actions are assumed to fail, which results in loss of long-term reactivity control);
10. LOSPW-EFW – This is a SBO sequence, similar to # 1, except that the turbine driven emergency feedwater (EFW) pump has failed. Therefore, time to core damage is short and no credit is given for recovery actions.

4.3.1. Initiating Event Contribution to Core Damage Frequency

Accident sequences that lead to core damage can be classified into internal event and external event initiated event sequence groups. From Figure 4. 3 and Figure 4. 4, it is seen that, based upon mean values, external events make a significant contribution to the calculated CDF, dominated by those from seismic, fire, and transportation events. This relative contribution is due partly to the relatively high levels of uncertainty for external event frequencies. Internal event-initiated accidents account for 57.4% [**2.62 x 10^{-5}**] of the CDF while external initiated events account for 42.6% [**1.94 x 10^{-5}**] of total CDF. Of the external initiating events, seismic events account for more than half of the external event risks. [Note: General transients include LOSP, loss of support systems (e.g. loss of a train of PCC), and other transients such as turbine trip, loss of feedwater.]

External event hazards include natural events (e.g. seismic events) and hazards initiating inside the plant involving materials introduced from outside (e.g. fire). External events have the ability to affect what are otherwise redundant and independent systems/trains. Hazards that cannot cause significant damage to the plant or that are extremely low in frequency were eliminated from our analysis. Also, for event sequences for which the annual CDF is less than ~1E-7 (~-1E-8 for CDF with containment impact) and the event does not have some unique effect not covered by another initiating event, the event was screened out from our analysis. Lastly, for the hazards that remained, a best estimate analysis was performed (in some cases) and, if possible, the risk contributions of the event sequences were screened again.

4.3.2. Seabrook NPS Seismic Hazards Analysis

A brief summary of the analysis of seismic risks for Seabrook NPS is presented in this section. The seismic analysis was performed using PRA methods. A summary of the major steps follows:

- Hazard Analysis - determination of the frequency of ground motions of various magnitudes at the site. A Seabrook site-specific hazard curve was developed for the SSPSA (Seabrook Station Probabilistic Safety Analysis).
- Fragility Analysis - determination of the seismically initiated ground acceleration at which plant structures and components are predicted to fail. Failures of structures and components having a median capacity of surviving a peak acceleration of 2.0g or greater were screened out of the analysis, as being negligible.
- Seismic Quantification - combination of hazard and fragility to yield initiating event frequencies and fragility values at discrete hazard levels in order to yield conditional system failure probabilities.
- Plant Model Assembly - integration of seismic initiators and seismic-initiated component failures with random hardware failures and maintenance unavailabilities in the plant event tree models.

Seabrook systems and components, which are essential to the prevention or mitigation of consequences of severe accidents, are designed to enable the facility to withstand the effects of natural forces including earthquakes. The plant should be able to withstand the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE). The structural design criteria were based upon respective peak horizontal ground accelerations of all seismic structure values of 0.25g for the SSE and of 0.125g for the OBE. The key component failures have one of two general plant effects: to cause a plant upset condition (an initiating event) or to cause failure of safety systems that are required to respond to the initiating event. These results can be used together with the estimated annual frequency of occurrence of various ground motion levels to determine the frequency of seismic-induced plant initiating events. These initiators along with the conditional failure probability of safety systems are used in the plant model to determine the probability of core damage and radionuclide release.

The initiating event frequencies were calculated by combining the mean hazard frequency at a discrete value of peak ground acceleration with the conditional probability of failure of the component (e.g. reactor internals) at that acceleration. This analysis resulted in 16 seismic initiators being identified – 9 general transient Initiating Events (IEs), 6 large LOCA IEs, and a single ATWS IE (ASEIS). The ASEIS initiator frequency is the sum of the ATWS IE frequencies for all the acceleration bins considered in the analysis. The ATWS IEs were combined because this initiating event is a core damage event itself. Table 4. 1 lists these seismically induced IEs and the values of their frequencies, their CDF (annual), and their percentage of total seismic initiator contribution.

Event trees used for internal events analyses are also used to model the plant response to seismic initiators. Event trees that model the courses of scenarios initiated by the above events are those of general transients (GT), ATWS and large LOCA. Component failures or failures of operator actions resulting in the systems being inoperable are represented as the top events in these trees and which would affect the scenarios were considered. Following a seismic event it is possible for mitigating equipment and actions to fail due to the earthquake's effects.

4.3.3. Release Category Frequency

Each accident sequence involving core damage will result in a certain release of radionuclides to the environment. This may range from a large release for early containment failures to a very small release for accident sequences where the containment remains intact. For Seabrook's reactor, more than 90% of the core damage sequences end with an intact containment or a "late" release – one occurring more than 24 hours after the event. The large, early release frequency (LERF) is an important risk measure because it is the release type that may occur before prompt emergency planning actions like evacuation can be completed. So it is an important factor of risk from the plant following a seismic event. The mean LERF total = **5.08 x $\mathbf{10^{-08}}$** per reactor-year (0.1% of release) for Seabrook station. The most important contributors to LERF are containment bypass and containment isolation failure rather than catastrophic failure of the containment structure (because as is mentioned earlier, the containment has a high median acceleration capacity of 7.8g, which makes it able to withstand most plausible seismic events). Listed below are the different release categories and their definitions. Table 4. 2 lists the release categories, their failure modes and frequencies per year.

Containment Intact – Given a core melt, the containment will remain intact in the long term and the release to the environment is restricted to the containment leakage limited by Technical Specifications, with essentially no public health effects.

Large Late Release – Given a core melt, the containment will eventually overpressurize and fail structurally. The time to containment failure is long (>24 hours) because of the large volume in the containment and strength of the containment.

Small Early Release – Given a core melt, containment is isolated except for a single 3-inch diameter opening, a larger initial leakage results. This size opening is too small to relieve the pressure buildup in containment, so eventually the containment also fails due to overpressurization. This results in a very small potential for early health effects.

Large Early Release – Given a core melt, a large opening exists with the potential for early as well as latent health effects.

4.3.4. Source Term Analysis

The estimated Seabrook Station source terms are shown in Table 4. 3. It lists the timing and release fractions for 16 accident event sequences. All releases are assumed to be non-buoyant ground level releases, which do not exhibit any plume rise. Each Release Category includes a realistic and conservative estimate of the source term magnitude. The tools used to evaluate source terms were developed over a time (1982 to 1987) during which the understanding of severe accident phenomenon and the analytical tools expanded, providing more realistic results. Three general methods were used in the SSPSA and subsequent studies to generate the currently available source term estimates. These results may contain conservative errors and biases. The three methods used to generate the source term estimates are the following:

1. **MARCH/CORRAL – Using WASH-1400 Methodology**

The MARCH/CORRAL codes used in the SSPSA are essentially unchanged from the codes developed and used in preparing the initial PRA, the Reactor Safety Study (WASH-1400). These calculate a source term that is conservative, based upon more recent knowledge. This methodology was used to calculate the "conservative" set of Seabrook source terms. The calculation involved either using the codes with Seabrook-specific input or adopting WASH-1400 source terms directly.

2. **MARCH/CORRAL with Manual Enhancements**

Based upon the understanding in 1982-83 of the conservatism in the source terms calculated by MARCH/CORRAL, the release fractions were manually adjusted based on expert opinion. These release fraction factors were subjectively estimated to account for in-vessel and ex-vessel radioactive material depletion processes that were either not taken into account or were under-estimated in MARCH/CORRAL. This method was the basis used to calculate a set of source term values in the SSPSA. The basis for all SSPSA source terms is the material released from the damaged core and which is potentially available to be released from the containment. The radioactive inventory supplied to CORRAL is based directly on WASH-1400 values, as described in Appendix H of the SSPSA. The CORRAL code model for Seabrook included the enclosure structure. The CORRAL code requires knowledge of this inventory as well as the release timing in order to evaluate the effects of the various material depletion mechanisms. The SSPSA used 3 basic core release tables, which are categorized according to event sequences as:·

- Transients and Small LOCAs
- Large LOCA with Steam Explosion at Vessel Failure (Oxidation Release).
- Large LOCA without Steam Explosion

3. **MAAP (Modular Accident Analysis Program)**

The MAAP code accounts explicitly for source term reductions based upon the current best estimate understanding of severe accident phenomena. It was used to generate revised source term estimates by running Seabrook specific models as shown in Table 4. 3

4.3.5. Accident Consequence Analysis

The end product of a PRA is a consequence analysis. The objective of this analysis is to estimate the number of health and economic effects affecting the population surrounding the power plant due to hypothetical radioactive atmospheric releases. A computer code called CRACIT (Calculation of Reactor Accident Consequences Including Trajectory) is used for this purpose at Seabrook Nuclear Power Plant. There are several other computer codes that can also be used (NURAC, TIRION, ALICE and CRAC) to calculate health effect consequences for each release category using random samples of meteorological conditions. Each consequence (e.g. health effect) set has a magnitude frequency of occurrence associated with it. A conditional frequency distribution of health and economic effects is developed and then combined with the occurrence frequency of each release category to comprise a statement of risk. This frequency is obtained by multiplying the frequency of occurrence of a radioactive material release category by the frequency of the accident event consequence calculated from meteorological scenario sampling. After the respective frequencies of all consequence sets have been calculated, they are combined to yield an overall frequency distribution of consequence versus probability. Several factors (including release timing and magnitude, atmospheric conditions, and spatial and temporal population distribution) can cause the effects to vary substantially for any radioactive material release category. The calculation of consequences is further influenced by the techniques used to simulate the above phenomena and to estimate doses to population and individuals. Hourly meteorological data are used to determine the position and radioactive material concentration of the radioactive plume for each simulated accident. They are selected to represent the spectrum of possible meteorological conditions. Meteorological data are gathered primarily from a 209-foot (63.7 meters) meteorological tower at the plant site. Data used included wind speed and wind direction for each stability class. Upper and lower level data and data from satellite stations are also used. Then population distribution and evacuation speed data are coupled with the radioactive material distribution to estimate doses and damages that are used to calculate the consequences.

Three population and evacuation scenarios are used. They are winter weekday, summer weekday and summer weekend day (worst case scenario). Out of 8,760 hours in the year, 6,384 hours used the winter weekday scenario, 1,973 hours used the summer weekday scenario and the remaining 403 hours use the summer weekend scenario, as modeled in CRACIT. In order realistically to represent the evacuation plans for accidents, evacuation trajectories simulated in CRACIT follow the routes and timing estimated for emergency for emergency planning purposes and from traffic studies conducted by transportation experts. Three sets of evacuation and population data were used for an evacuation zone consisting of the area within 10 miles (16 km) of the plant. Important factors in evacuation include travel distances, speeds and delay times (which are very important parameter in determining early fatality risk if delay extends the time of departure beyond time of release). Delays could also result from late notification of authorities and people, adverse weather, communications failures, etc. Sheltering is another way of mitigating the effects of releases but which was not used in this PRA.

Factors omitted from the SSPSA include the effect of the earthquake upon efforts to reduce radiological consequences. These efforts include evacuation, sheltering and medical help. All such efforts are likely to be hindered by an earthquake, particularly strong ones. Evacuation would be hindered by damage to and debris upon transportation infrastructure. Sheltering would be hindered by damage to structures and by the possibility of subsequent damage, caused by aftershocks. Medical assistance would be hindered by damage to medical facilities, heavy overall demands upon available medical resources, and by difficulty in having medical personnel reach radiologically affected people.

The radiation doses are computed for early (short time following release) and chronic exposures. Early dose is assumed to arise from plume shine (dose received as the radionuclide plume passes over the receptor), inhalation of radioactive material and dose received from material deposited on the ground (groundshine). It is also assumed in the PRA that residents beyond ten miles would be relocated but not immediately following a release. Health effects estimated are acute fatalities, cancer fatalities, thyroid cancer fatality, and whole body man-rem. Economic damages calculated include costs of evacuation and relocation, the costs of interaction and decontamination of land, and costs of interdiction of crops and other farm products. The estimated losses due to a seismic release from Seabrook as calculated from the PRA are included in Table 4. 5. However, the PRA does not take other expected economic losses that are related to a release from a power plant into consideration. Examples and estimates of such losses due to an accident at Seabrook NPS are presented in Figure 4. 5 in 2001 dollars.

Early fatalities are mainly due to damage the bone marrow, lung and gastrointestinal tract; deaths due to radiation thyroiditis and prenatal exposure are expected to be relatively small. The number of early fatalities is calculated by comparing the doses to bone marrow, to lung and to gastrointestinal tract. The probability of fatality from the bone marrow irradiation dominates the corresponding probabilities for lung and gastrointestinal tract, at all distances. Injuries are due to exposure to sublethal doses. Extensive exposure would result in respiratory impairment (SD_{50}[1] is about 4500 rads), gastrointestinal morbidity (the threshold is about 1000 rads and reaches the 100% level at 2500 rads), thyroid morbidity (includes hypothyroidism and radiation thyroiditis), sterility, congenital malfunctions and growth retardation, cataracts as well as prodromal vomiting. Late somatic effects are limited to latent cancer fatalities and morbidities plus benign thyroid nodules.

Accident sequences that lead to similar releases of radioactive material are grouped into one of a set of release categories. The release categories used are based upon different horizontal peak ground acceleration (PGA) levels of earthquakes and the risk curves are graphs of consequences vs. frequency of exceedance of these consequences at discrete values of peak ground acceleration PGA. These frequency-consequence (F-C) curves (sometimes referred to as complementary cumulative distribution functions (CCDF)) were obtained for early fatalities and latent cancer fatalities. See Figure 4. 7 and Figure 4. 8 for the risk curves for early fatality and latent cancer respectively. Each curve represents the risk curve for a conditional earthquake of discrete horizontal peak ground acceleration shown. These curves show the predicted probability [per reactor-year (frequency)] with which an accident will cause fatalities greater than the corresponding number of fatalities on the x-axis.

---

[1] SD_{50} is defined as the sublethal dose that is expected to cause a clinical response in 50% of the exposed population.

These frequency values obtained were then combined with the seismic event probabilities to obtaining the conditional probabilities of exceedance of a consequence a seismic event has occurred. A discretized mean value frequency table at accelerations between 0.1g and 2.0g, which was used in obtaining seismic hazard curves, is provided below in Table 4. 4 and Figure 4. 6. This was obtained by Seabrook NPS from historical earthquake data analysis. The CCDFs themselves can be used as a measure of public or societal risk. However, in order to make them understandable to the general public and to make comparisons with the expected fatalities from the direct effects of earthquakes, mean estimates of fatalities need to be derived from these risk curves for earthquakes of various PGA values. Using the initiating event vector and plant, containment, and site matrices provided by the PRA, a simple matrix multiplication operation is performed to integrate the various matrices and obtain a vector of annual consequence risks. Expected (in a statistical sense) number of early fatalities (with a 10-mile evacuation), latent fatalities and injuries per year are derived from this. The results are presented in Table 4. 5 and Figure 4. 9. Seabrook NPS did a study to investigate and improve emergency planning[13]. In this study, the plant considered the effects of different evacuation distances and sheltering, for the emergency planning zone (EPZ), on the fatality consequences. This was done for the overall risk from the plant. Using the different release categories, average factors of difference between different evacuation differences was obtained and this was iterated backwards for the seismically induced risk at the NPP. It was from this that the different levels of fatalities at different evacuation distances presented in Table 4. 5 were derived. More detailed information about the Seabrook NPS PRA can be found in some references[14],[15],[16].

4.4. HAZUS Loss Estimation for Greater Boston Area

In New England, natural disasters that are prevalent include floods, hurricanes, severe winter storms, ice jams, wildfires, nor'easters, and even the stray tornado. Earthquake hazards are not typically associated with common natural hazards of New England states. However, there is a history of large earthquakes in New England dating back to the earthquake that took place off the coast of New Hampshire and Massachusetts in 1727, Cape Ann earthquake of 1755 at an estimated magnitude of 6.0, and many others. While there was hardly any measuring device, it was recorded that the earthquakes shattered brick buildings, toppled chimneys, and caused other types of damage. Compared to the nineteenth century records, there are more reported earthquakes in the twentieth century. This increase is mainly due to a larger population density increases the chances of a small event being felt, and the installation of seismographs capable of recording minor tremors that normally are not felt. The bedrock east of the Rocky Mountains is colder and harder than that to the west, and transmits seismic waves very well which makes earthquakes which reduces the attenuation of earthquake waves New England area. It is now accepted and understood the possibility of a large earthquakes occurring in New England by geologists and seismologists, and they consider the New England area to have a moderate seismic risk. The study region selected was the greater Boston area that includes Seabrook Nuclear Power Plant in Seabrook, New Hampshire, and southern parts of the State of Maine. Boston is in Suffolk County, Massachusetts, and Seabrook Nuclear Power Plant in located in the eastern part of Rockingham County, New Hampshire. Figure 4. 10 shows an outline of the study region. The information on this study region is as follows:

- 12 counties in Massachusetts, New Hampshire and Maine
- Total Area: 7,042 mi^2 (18,237 km^2)

- Total Population: 4,611,600 people (1990 Census Bureau Data)
- Total Households: 1,734,000 households
- Total Buildings: 1,107,887 buildings
- 1,062,585 residential buildings
- Total Bridges: 3,948 bridges.

Results obtained from the HAZUS program is displayed in Table 4.6. It presents the expected casualties, economic damage and structural damage at different sizes of earthquakes centered at both Seabrook and Boston. The earthquakes were centered in both areas to show the difference expected depending on whether a seismic event originated in a large metropolitan like Boston or a small town like Seabrook.

Initiating Event	Initiator Frequency (per year)	CDF(per year)	% Contribution to CDF due to Seismic Events
General Transients		**6.30E-06**	**60**
E1T – Seismic 0.1G Transient Event	4.70E-07	2.00E-07	1.9
E2T – Seismic 0.2G Transient Event	3.67E-04	4.30E-07	4.1
E3T – Seismic 0.3G Transient Event	1.02E-04	5.80E-07	5.5
E4T – Seismic 0.4G Transient Event	4.35E-05	6.80E-07	6.5
E5T – Seismic 0.5G Transient Event	1.92E-05	7.90E-07	7.6
E7T – Seismic 0.7G Transient Event	1.64E-05	2.60E-06	24.6
E10T – Seismic 1.0G Transient Event	2.18E-06	9.00E-07	8.5
E14T – Seismic 1.4G Transient Event	1.48E-07	1.40E-07	1.4
E20T – Seismic 2.0G Transient Event	6.20E-09	6.00E-09	0.1
LLOCA		**1.30E-06**	**12.3**
E4L- Seismic 0.5G LLOCA	3.54E-08	7.50E-10	< 0.1
E5L – Seismic 0.5G LLOCA	1.16E-07	7.50E-09	0.1
E7L – Seismic 0.7G LLOCA	1.00E-06	2.50E-07	2.3
E10L – Seismic 1.0G LLOCA	9.00E-07	5.30E-07	5
E14L – Seismic 1.4G LLOCA	3.86E-07	3.80E-07	3.6
E20L – Seismic 2.0G LLOCA	1.27E-07	1.30E-07	1.2
ATWS ASEIS (combined ATWS IE for 0.3g to 2.0g Seismic Events)	2.91E-07	**2.90E-06**	**27.7**
TOTAL SEISMIC INITIATORS		1.05E-05	100

Table 4. 1: Seismic Initiating Event Contributions to Core Damage Frequency Total

Release Classification	Containment Failure Mode		Total Frequency (per year)
Large Early Release Frequency (LERF)			5.09E-08
	Overpressurization from direct containment heating		1.04E-09
	Steam/hydrogen explosion		
	Containment isolation failure		1.20E-08
	Induced steam generator tube rupture (SGTR) due to high RCS temperature and pressure without steam generator coolant makeup		9.90E-10
	SGTR, early core melt with stem line bypass		3.23E-08
	Containment bypass- direct release through-wall pipe break		4.55E-09
Small Early Release Frequency (SERF)			2.99E-06
	Small leak (type A) progressing to large, late failure		1.44E-07
	Small leak (type B) does not increase in size		9.58E-07
	Small leak (type B) with recovery to prevent large, late release		1.77E-06
	SGTR late core melt with steam line bypass		1.14E-07
	Containment bypass - submerged release		7.56E-09
Large Late Release Frequency (LLATE)			1.23E-05
	Late containment overpressurization with dry containment		7.78E-06
	Late containment overpressurization with wet containment		4.54E-06
	Late basemat melt through 9' reactor cavity basement		1.05E-08
Containment Intact	Intact, containment design basis leak rate		3.03E-05
Total			4.57E-05

Table 4. 2: Containment Radioactive Release Categories and Occurrence from Seabrook Nuclear Power Plant PRA

Source Term Event Sequence ID	Puff	Start Time (Hrs)	Duration(Hrs)	Fraction of Core Inventory Released						Release Category
				Kr-Xe	Cs-Rb	Te	Ba-Sr	Ru	La	
S1A - Large, Early Release w/DCH or Steam Explosion/ Hydrogen Burn										
S1A-C	1	3.5	0.01	9.4E-01	2.3E-01	2.4E-01	3.3E-03	4.1E-01	1.0E-04	LERF-S1A
S1A-R	1	16.6	0.2	9.4E-01	7.5E-01	3.9E-01	9.3E-02	4.6E-01	2.8E-03	
S2 - Small, Early Release - Early Penetration Failure or Small Pre-existing Leak										
S2-C	1	2.2	2	3.0E-02	2.3E-02	4.2E-03	2.8E-03	8.0E-04	8.0E-05	SERF-S2A
	2	4.2	4	7.0E-02	4.8E-02	3.9E-02	5.5E-03	3.4E-03	5.0E-04	SERF-S2B
	3	8.2	18	2.3E-02	1.3E-01	1.5E-01	1.4E-02	1.1E-02	1.9E-03	SERF-S2R
	4	26.2	4	8.8E-01	1.1E-01	1.3E-01	1.2E-02	9.8E-03	1.7E-03	(a)
S2-R	1	2.2	12	1.5E-01	4.0E-03	7.0E-04	5.0E-04	2.0E-04	2.0E-05	
	2	14.2	8	2.0E-01	7.0E-03	8.0E-04	8.0E-04	6.0E-04	1.0E-04	
	3	22.2	56	6.5E-01	2.0E-03	2.0E-03	2.0E-04	1.0E-04	2.0E-05	
S3A - Large, Late Release - Dry Containment, Late Overpressurization Failure or Basemat Melt-through										
S3A-C	1	28	1	1.0E+00	1.5E-02	1.9E-02	1.6E-03	1.5E-03	3.0E-04	LATE-S3A,
S3A-R	1	89	0	1.0E+00	1.0E-03	2.0E-03	1.0E-05	1.0E-05	1.0E-05	LATE-S4 (b)
S3B - Large, Late Release - Wet Containment, Late Overpressurization Failure										
S3B-C	1	22	1	1.0E+00	2.6E-02	4.9E-03	3.3E-03	1.0E-03	1.0E-04	LATE-S3B
S3B-R	1	53	1	7.0E-01	9.0E-04	2.0E-04	1.0E-04	3.0E-05	3.0E-06	
S5 - Intact Containment										
S5-C	1	4.3	24	1.4E-02	5.0E-07	9.0E-08	6.0E-08	2.0E-08	2.0E-09	INTACT-S5
S5-R										
S6 - Large, Early Release - Containment Isolation Failure or Large Pre-existing Leak										
S6-C	1	1.7	1	1.5E-01	1.1E-01	2.0E-02	1.4E-02	4.1E-03	4.0E-04	LERF-S6
	2	2.7	4	4.2E-01	1.9E-01	6.3E-02	2.2E-02	9.0E-03	1.0E-03	
	3	6.7	10	3.2E-01	1.3E-01	3.2E-01	1.1E-02	2.0E-02	3.8E-03	
S6-R	1	4	2	2.0E-01	4.0E-03	9.0E-05	3.0E-04	2.0E-05	2.0E-05	
	2	6	4	3.0E-01	5.0E-03	1.0E-04	3.0E-04	3.0E-05	3.0E-05	
	3	10	10	5.0E-01	1.0E-03	9.0E-05	2.0E-05	1.0E-05	1.0E-05	
S7A - Bypass - ISGTR or RHR Unflooded Pipe Break (Interfacing LOCA)										
S7A-C	1	2.5	0.5	9.0E-01	5.0E-01	3.0E-01	6.0E-02	2.0E-02	4.0E-03	LERF -S7S
S7A-R										LERF-S7I
										LERF-S7V
S7B - Bypass - SGT with Safety Valve Open or RHR Flooded Pump Seal (interfacing LOCA)										
S7B-C	1	8.5	7	9.0E-01	5.0E-02	3.0E-02	6.0E-03	2.0E-03	4.0E-04	SERF-S7S
S7B-R	1	8.5	7	9.0E-01	5.0E-04	3.0E-04	6.0E-05	2.0E-05	4.0E-06	SERF-S7V

Table 4. 3: Seabrook Source Term

Discrete Seismic Acceleration Level (g)		Mean Annual Bin Frequency (1/yr)
Midpoint Value (g)	Bin Boundaries (g)	
0.1	0.05 to 0.15	4.70E-03
0.2	0.15 to 0.25	3.67E-04
0.3	0.25 to 0.35	1.02E-04
0.4	0.35 to 0.45	4.38E-05
0.5	0.45 to 0.55	1.97E-05
0.7	0.55 to 0.85	1.88E-05
1.0	0.85 to 1.15	3.75E-06
1.4	1.15 to 1.65	6.64E-07
2.0	>1.65	1.48E-07

Table 4. 4: Seabrook Site Discrete Mean Peak Earthquake Acceleration Frequency Distribution

PGA (g)	0.2	0.3	0.4	0.5	0.7	1.0
Nuclear Early Fatality with 10 Mile evacuation	4.63E-04	3.70E-03	3.88E-02	4.23E-01	1.97E+00	2.98E+00
Nuclear Early Fatality without evacuation	3.03E-03	2.42E-02	2.54E-01	2.76E+00	1.29E+01	1.95E+01
Nuclear Early Fatality with 2 Mile evacuation & 10 mile sheltering	5.53E-04	4.42E-03	4.63E-02	5.05E-01	2.35E+00	3.57E+00
Latent Cancer Fatality with 10 Mile evacuation	3.80E+00	1.39E+01	5.96E+01	1.45E+02	3.51E+02	5.06E+02
Nuclear Injuries (# or person)	2.04E-03	9.65E-03	6.80E-02	5.50E-01	2.44E+00	3.70E+00
Nuclear Economic Damage ($)	1.86E+06	6.85E+06	2.96E+07	7.38E+07	1.82E+08	2.61E+08

Table 4. 5: Consequence Analysis Results from Seabrook NPS PRA

Maximum PGA(g)	Centered in Boston			Centered in Seabrook		
	0.54	0.94	1.46	0.53	0.92	1.44
Severity 1	18,414 (0.146%)	80,638 (0.638%)	198,493 (1.57%)	2,952 (0.023%)	21,058 (0.167%)	78,217 (0.619%)
Severity 2	3,077 (0.024%)	14,758 (0.117%)	37,665 (0.298%)	416 (0.003%)	3,492 (0.028%)	14,003 (0.1%)
Severity 3	354 (0.003%)	2,003 (0.016%)	5,340 (0.042%)	35 (3E-6)	425 (0.003%)	1,896 (0.015%)
Severity 4	328 (0.003%)	1,786 (0.014%)	4,749 (0.038%)	32 (3E-6)	371 (0.003%)	1,647 (0.013%)
Prompt Fatalities	328	1,786	4,749	32	371	1,647
Hospital Functionality	0.77	0.55	0.34	0.92	0.72	0.46
Total Fatalities(1)	409	2,687	8,273	35	490	2,671
Total Injuries(2)	22,118	96,478	237,974	3,400	24,856	93,092
Direct Economic Loss for Building(3)	29,175	80,782	166,316	5,475	26,213	75,087
Direct Economic loss for Transportation(4)	158	730	2,007	57	393	1,415
Direct Economic loss for Utilities(4)	434	972	1,942	137	445	1,138
Total direct economic loss(5)	29,769	82,517	170,265	5,674	27,050	77,641

(1) Total Fatalities=[severity 4]+[severity 3]x(1- Hospital Functionality)

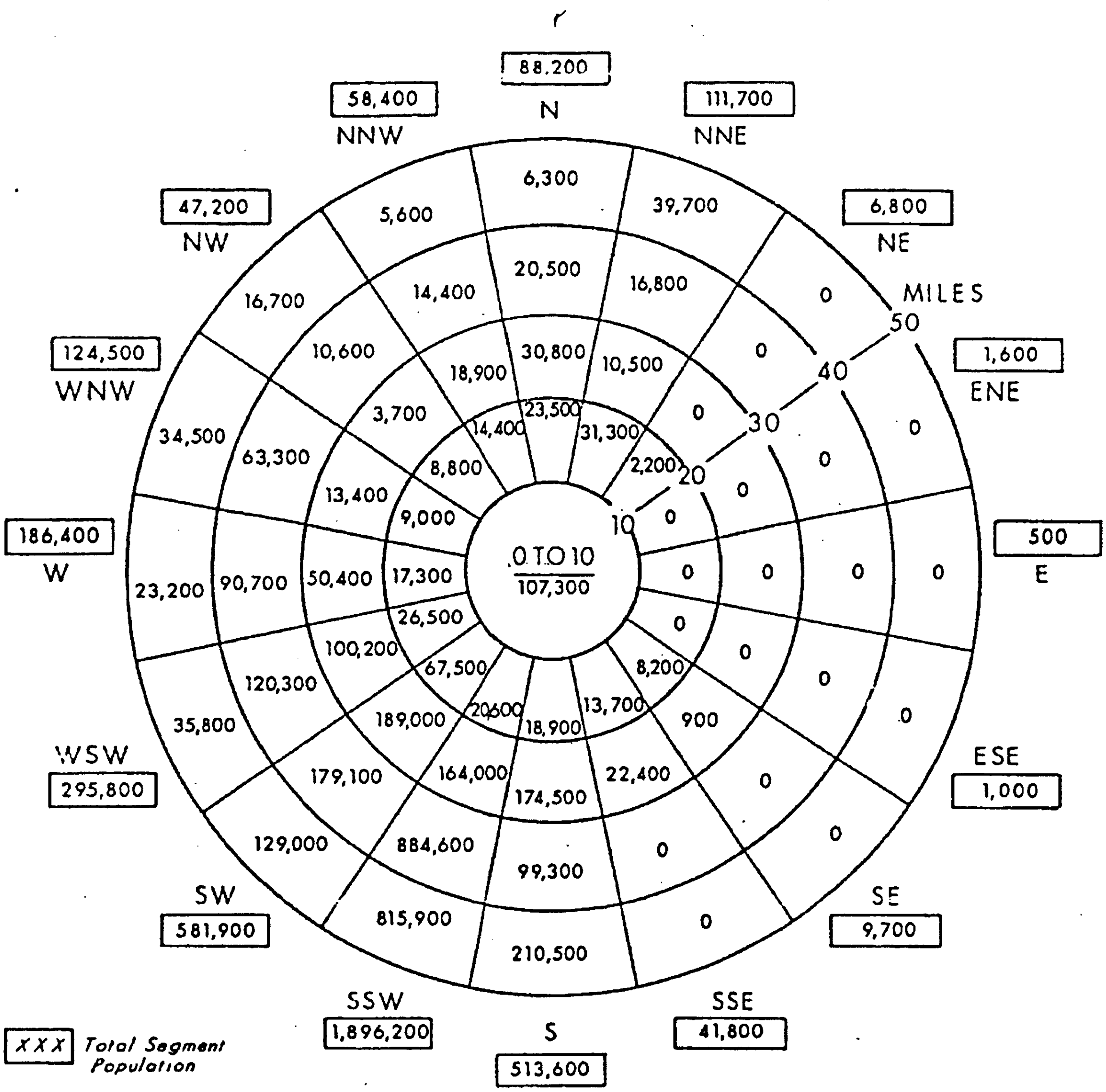
(2) Total Injuries=[severity 1]+[severity 2]+[severity 3]x [Hospital Functionality]

(3) Direct economic loss for building includes Property Damage (Capital Stock) Losses and Business Interruption (Income) Losses. Property Damage (Capital Stock) Losses includes

- Building Repair and Replacement Costs
- Building Contents Losses
- Building Inventory Losses
- Business Interruption (Income) Losses in turn includes
- Relocation Expenses
- Loss of Proprietors' Income
- Rental Income Losses

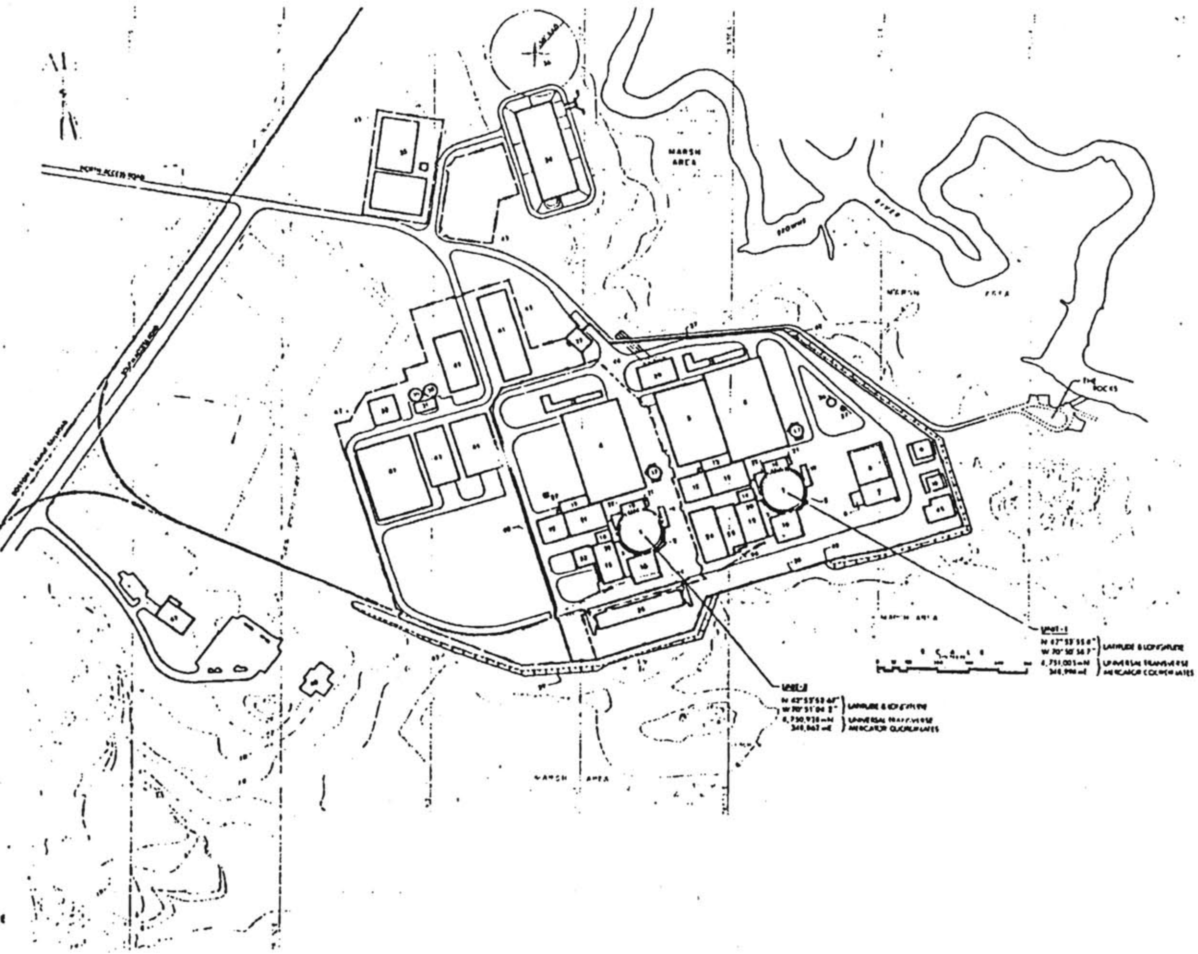
(4) Direct Economic loss for Transportation includes the cost of damage to lifelines, but no attempt is made to estimate losses due to interruption of customer service, alternative supply services, etc.

Table 4. 6: Estimate Of Direct Effects Of Earthquakes Of Varying Magnitudes, Centered In Boston And Seabrook, Respectively



POPULATION TOTALS			
RING, MILES	RING POPULATION	TOTAL MILES	CUMULATIVE POPULATION
0-10	107,300	0-10	107,300
10-20	261,900	0-20	369,200
20-30	778,700	0-30	1,147,900
30-40	1,499,600	0-40	2,647,500
40-50	1,317,800	0-50	3,965,300

Figure 4. 1: 1983 Resident Population



PRINCIPLE STATION STRUCTURES

1 CONTAINMENT STRUCTURE
2 CONTAINMENT ENCLOSURE AREA
3 CONTAINMENT ENCLOSURE VENTILATION AREA
4 TURBINE BUILDING AND HEATER BAY
5 ADMINISTRATION AND SERVICE BUILDING
6 CIRCULATING WATER PUMPHOUSE
7 SERVICE WATER PUMPHOUSE
8 ELECTRICAL CONTROL ROOM
9 INTAKE TRANSITION STRUCTURE
10 DISCHARGE TRANSITION STRUCTURE
11 CONTROL BUILDING
12 DIESEL GENERATOR BUILDING
13 NON-ESSENTIAL SWITCHGEAR ROOM
14 THE SPRAY EQUIPMENT VAULT
15 PRIMARY AUXILIARY BUILDING
16 FUEL STORAGE BUILDING
17 CONDENSATE STORAGE TANK
18 EMERGENCY FEEDWATER PUMPHOUSE
19 STEAM & FEEDWATER PIPE CHASE (EAST)
20 STEAM & FEEDWATER PIPE CHASE (WEST)
21 PRE ACTION VALVE BUILDING
22 PERSONNEL HATCH AREA
23 TANK FARM AREA
24 WASTE PROCESSING BUILDING
25 SERVICE WATER COOLING TOWER
26 DEMINERALIZED WATER STORAGE TANK
27 CONTROL BUILDING MAKE-UP AIR INTAKE STRUCTURE
28 TRANSFORMER AREA
29 SWITCHING STATION (SUBSTATION)
30 FIRE PROTECTION TANKS
31 FIRE PUMP HOUSE
32 FUEL OIL STORAGE TANK
33 GUARD HOUSE
34 SETTLING BASIN
35 SEWAGE TREATMENT & LAGOON AREA
36 METEOROLOGICAL TOWER
37 RETAINING WALL
38 SEAWALL
39 RIPRAP
40 EDUCATION CENTER
41 MISC NON-OPERATIONAL STRUCTURES
42 PERMANENT SECURITY FENCE FOR OPERATING UNITS 1 AND 2
43 SUPPLEMENTARY FENCE FOR PROTECTION OF ANCILLARY PLANT STRUCTURES
44 TEMPORARY SECURITY FENCE ISOLATING OPERATING UNIT 1 DURING CONSTRUCTION OF UNIT 2
45 CHLORINATION BUILDING
46 STEAM GENERATOR BLOWDOWN BUILDING
47 TRAINING CENTER

Figure 4. 2: Seabrook NPS Layout

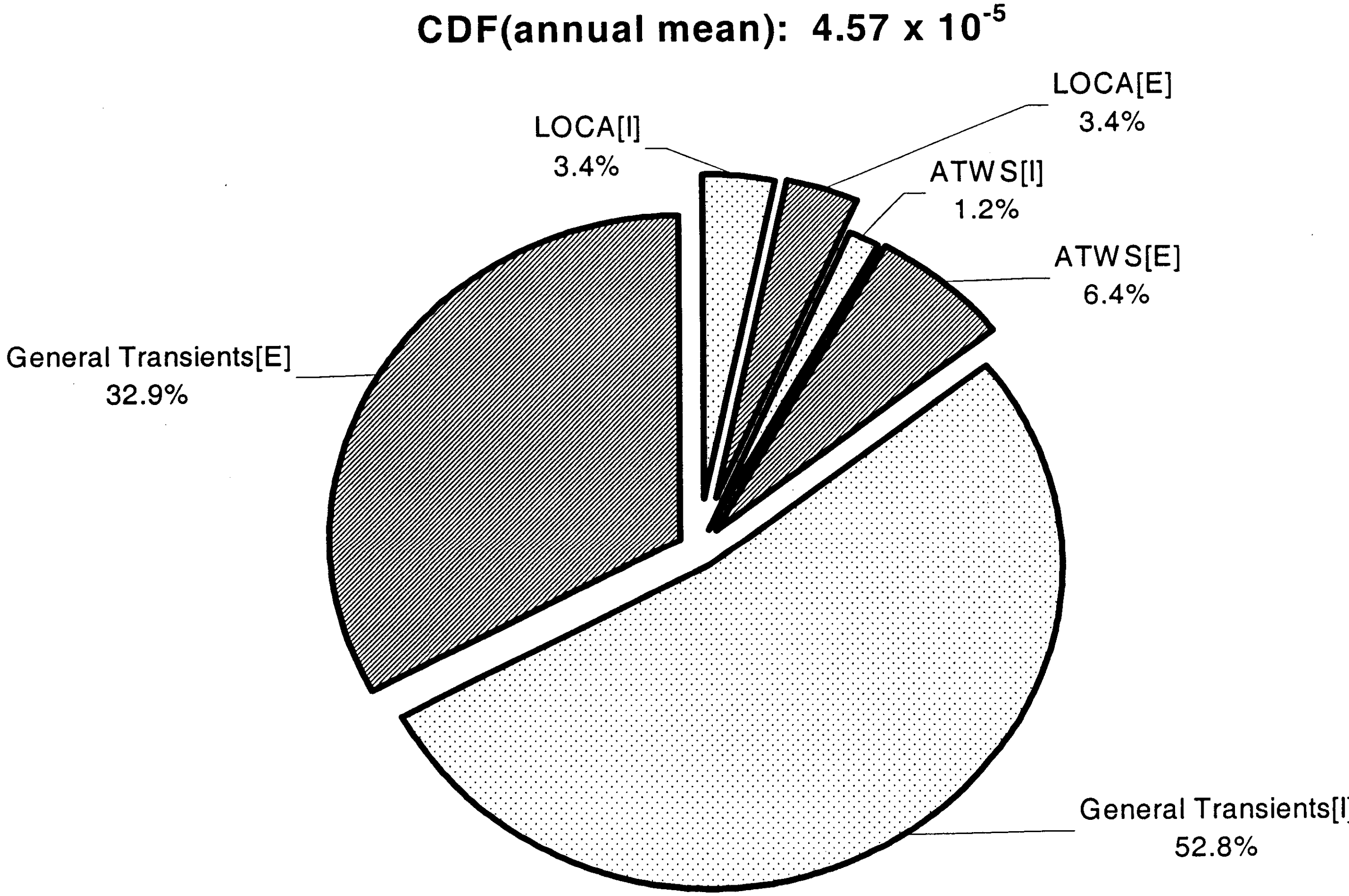


Figure 4. 3: Accidents Leading to Core Damage Grouped by Class of Accident and by Internal and External Initiating Event Designated by [I] and [E] Respectively [Data Source:SSPSS-1999]

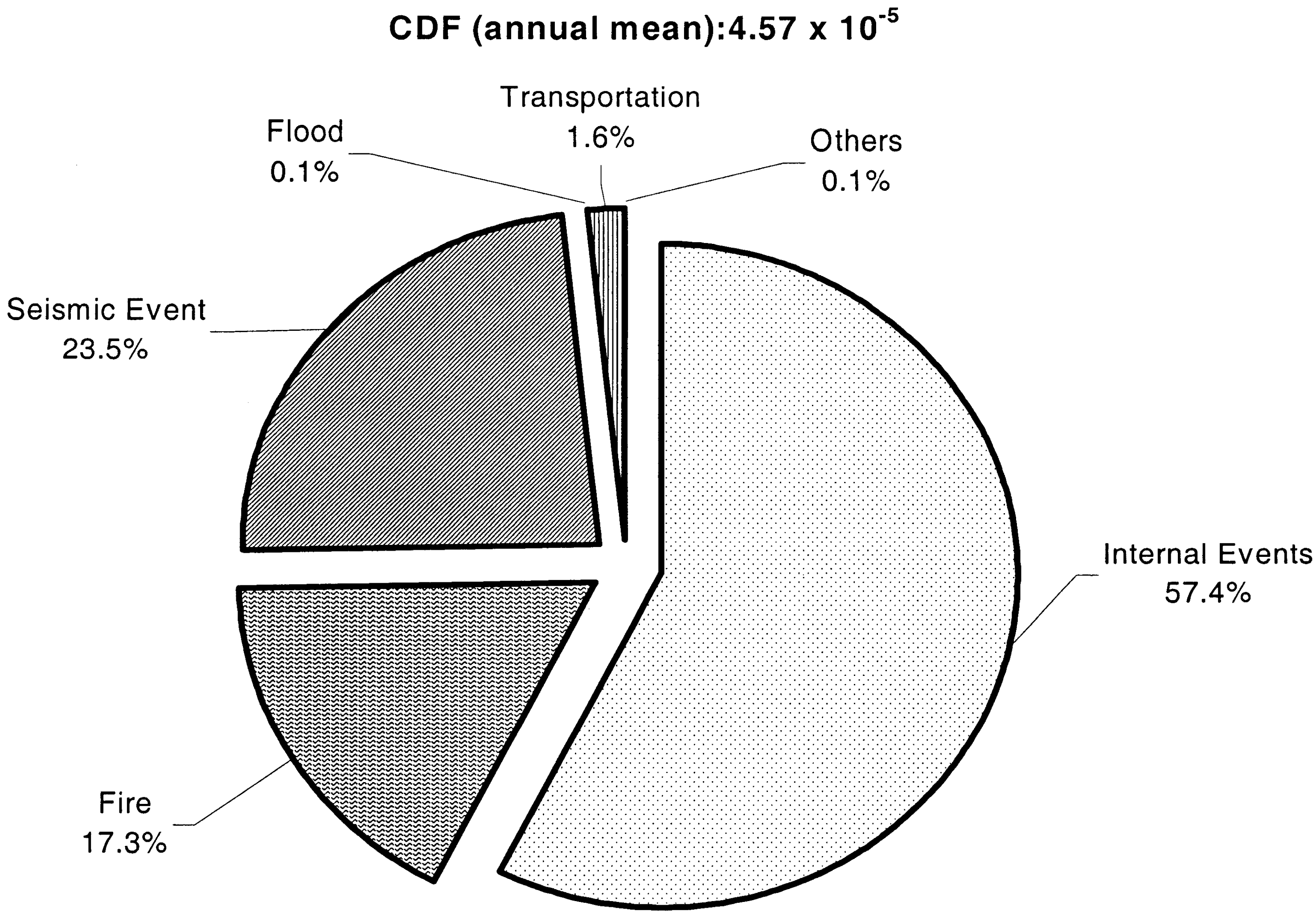


Figure 4. 4: CDF Contribution From Internal and External Initiating Events [Data Source: SSPSS-1999]

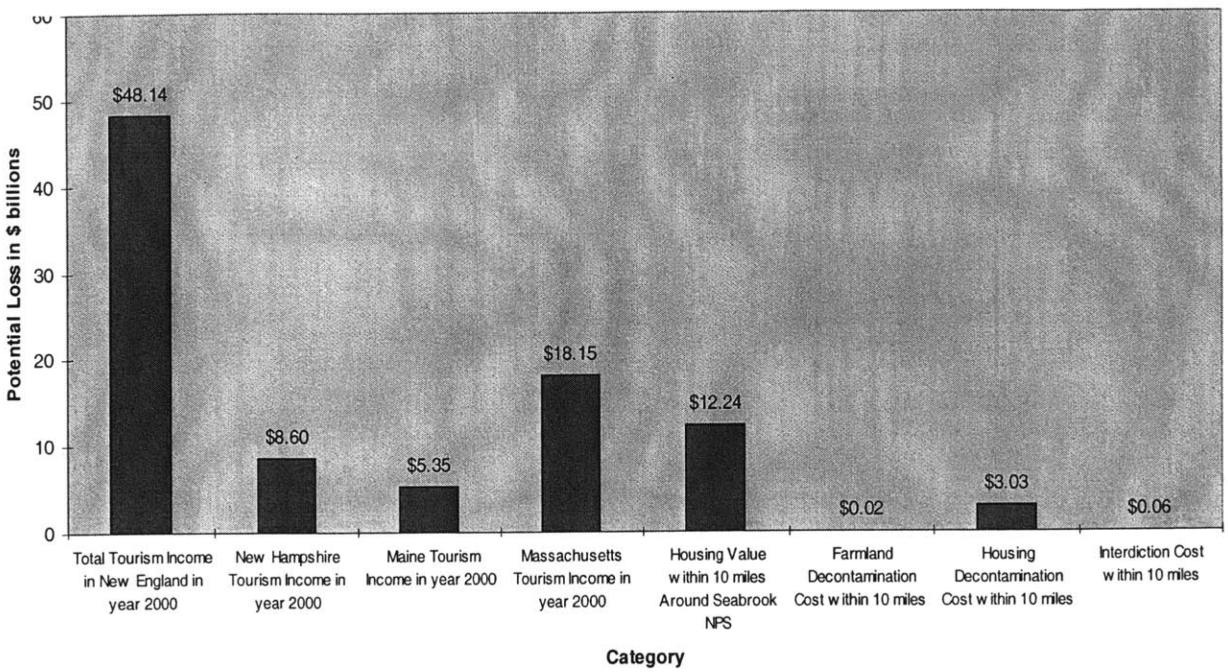


Figure 4. 5: Potential Economic Losses Due Reactor Accident at Seabrook NPS
[Source: Yingli Zhu, MIT]

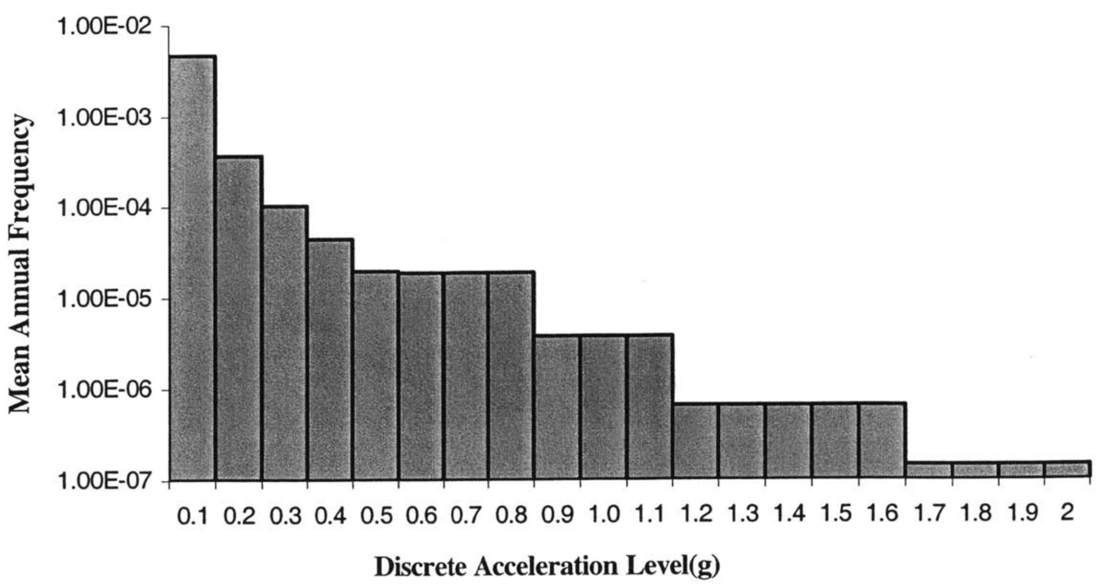


Figure 4. 6: Seabrook Site Discrete Mean Peak Earthquake Acceleration Frequency Distribution

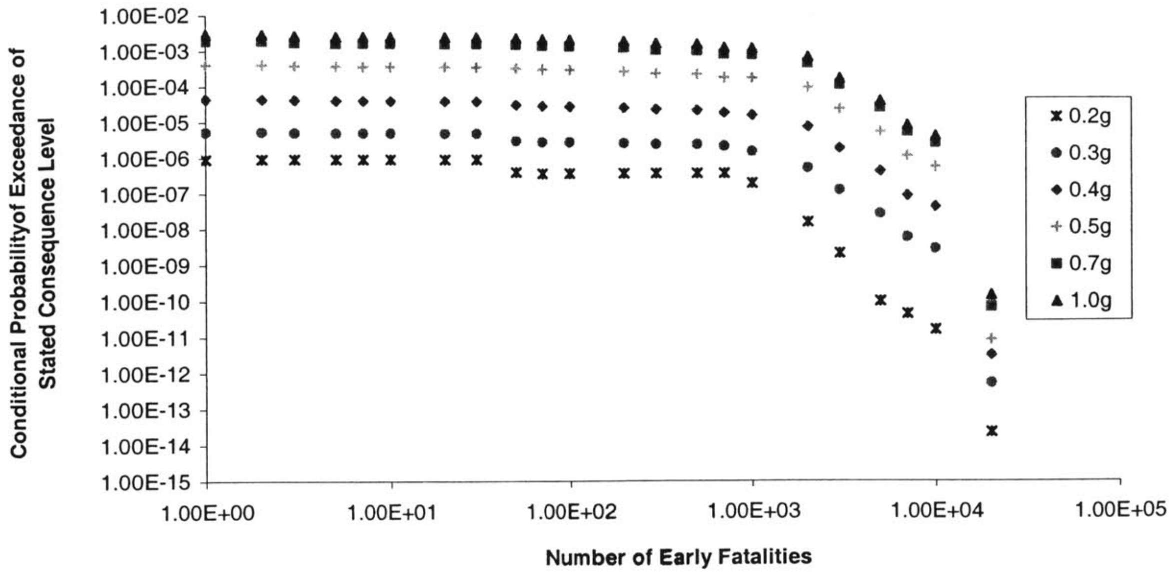


Figure 4. 7: Seabrook NPS Conditional Mean Risk of Early Fatality for Seismic Events of Different Horizontal Peak Ground Accelerations(g)

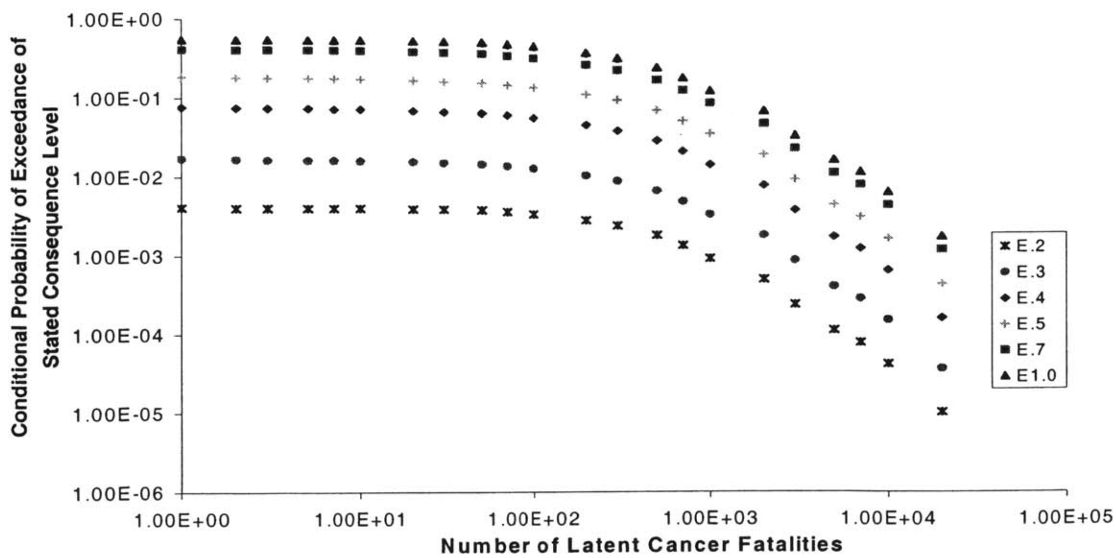


Figure 4. 8: Seabrook NPS Conditional Mean Risk of Latent Cancer Fatality for Seismic Events of Different Horizontal Peak Ground Accelerations (g)

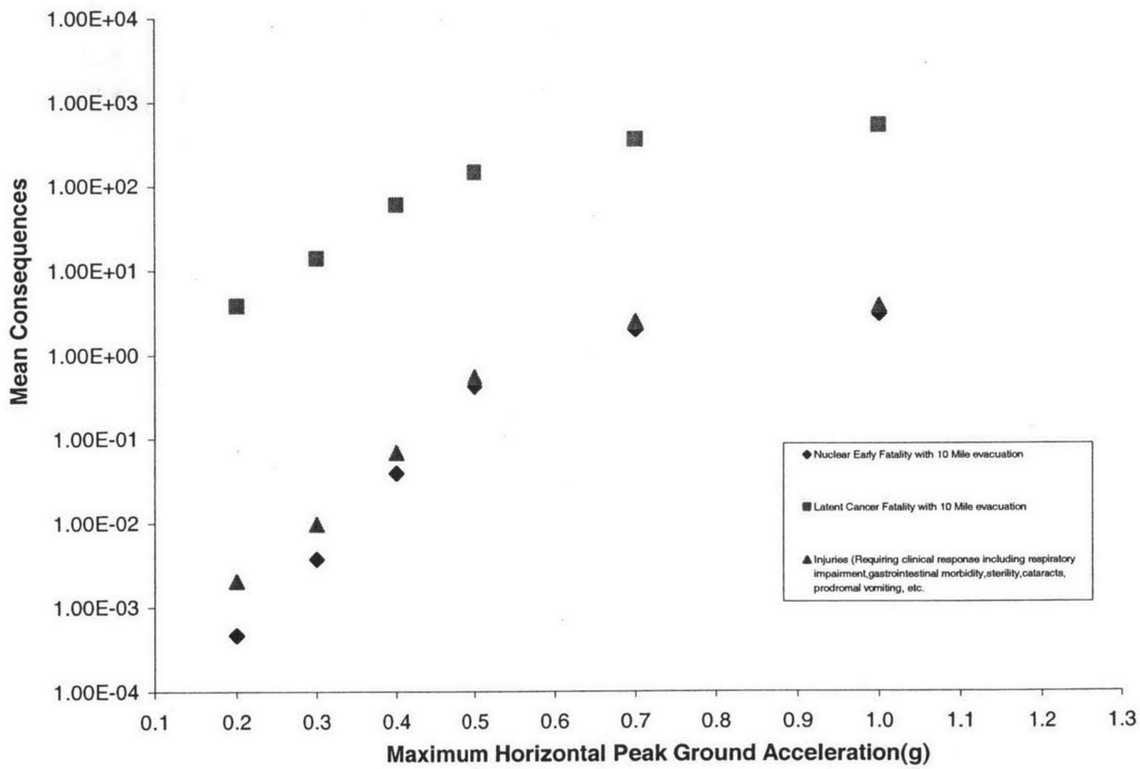


Figure 4. 9: Conditional Mean Risk from Seabrook NPS

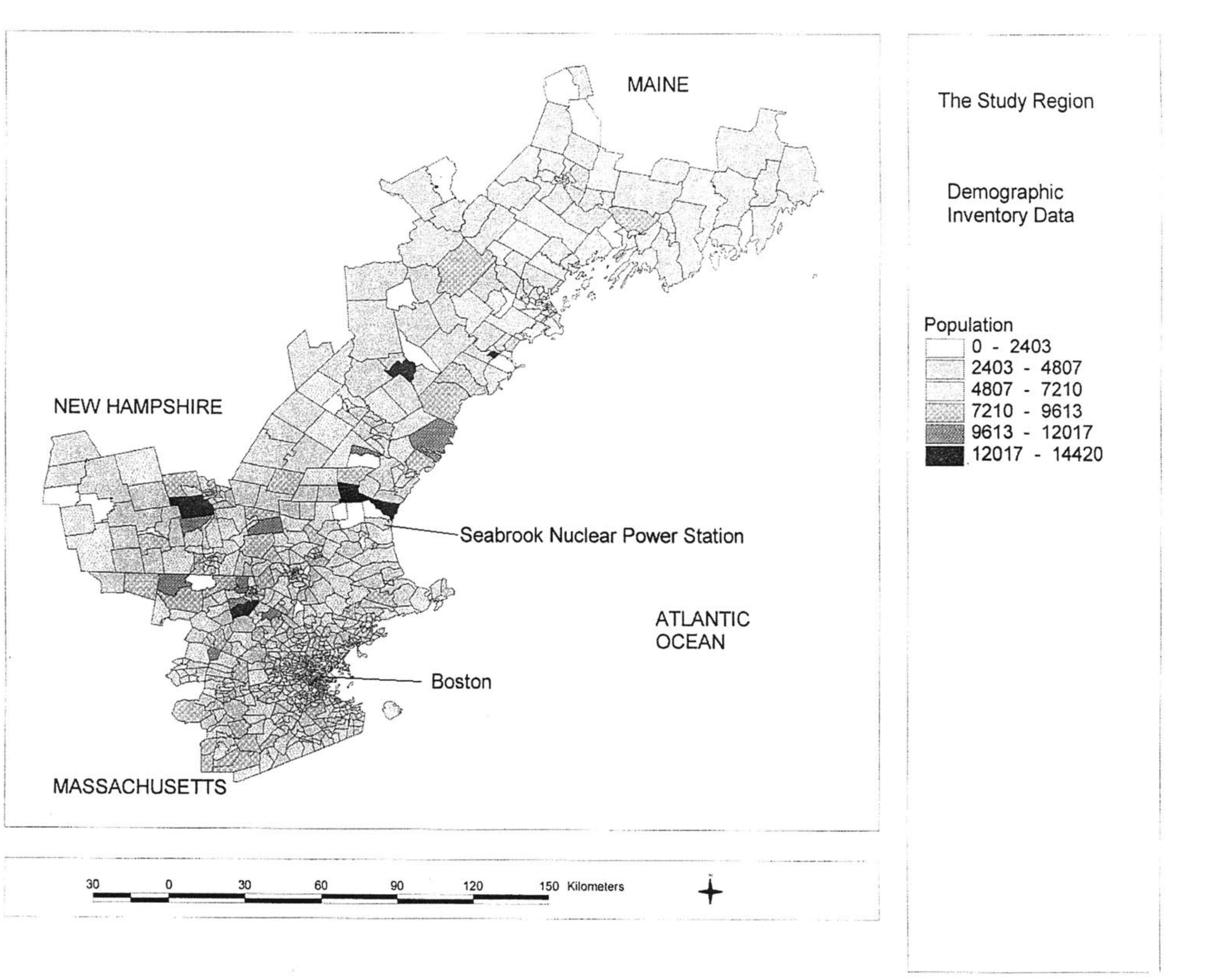


Figure 4. 10: HAZUS Study Region Population Density Map

C: Policy Analysis

Chapter 5: Quantitative Public Safety Assessment

In the following chapter, results gotten from the previous chapters will be used to illustrate comparison bases for the consequences from direct seismic events and seismically induced nuclear risks. Acceptability criteria to be used for quantitative goals are then derived using these comparisons.

5.1. Acceptability Criteria

As shown in the results from the previous chapter, seismic nuclear risk can be a significant contributor to overall nuclear risks posed by a NPP. It is therefore important to keep nuclear power plant-related seismic consequences small. We recognize, however, that risks are associated with the benefits and costs of any technological endeavor. Risks cannot be totally eliminated but can be reduced and managed. Risk management is a socio-political as well as technical problem. Due to a lack of consensus as to what aspects of risks are important, creating a workable quantitative risk management framework is a major task. This follows for risks from seismic events in any region with or without a nuclear power plant. Hence, safety goals (or policies) are needed as parts of risk management. Any proposed safety goal must state a degree of protection that is required and must be formulated in such a way that is useful in a practical and regulatory sense. In setting such goals, one has to be careful not to go to the extreme limits in setting safety goals. These extremes are: (1) goals of zero risk, which would ensure safety to the public at the cost of sacrificing other goals such as development and (2) goals where too much weight is placed on the development of a particular technology, even at the cost of high risk to the public.

With the above in mind, the next task in the project was to formulate an acceptability criterion for seismic nuclear risks that would provide sufficient protection for the public that is likely to be exposed to these risks. Safety goals have been used in the past to show the acceptability criteria for many industries. For the nuclear industry, the United States Nuclear Regulatory Commission (USNRC) has a safety policy statement[17]. The objective of the policy statement is to 'establish goals that broadly define an acceptable level of radiological risk' and expresses the NRC's views on the level of risks to public health and safety that the industry should strive for in its nuclear power. The NRC has two qualitative goals (not limits) supported by two quantitative objectives based on the principle that nuclear risks should not be a significant addition to other societal risks.

Qualitative Goals:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life or health.
- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The intent of these goals is to require a level of safety such that individuals working and living near nuclear power plants will not be concerned about their proximity to the plants. To determine the achievement of the safety goals, the following quantitative objectives are used:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1

percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

PRA is the methodology used in determining whether these above goals (quantitative) are being met or not, by various NPPs. However, because of the complexity and uncertainties of performing a level 3 PRA where public health risks are evaluated, subsidiary objectives, which achieve the same intent as the quantitative health objectives, are useful in making regulatory decisions. They include a CDF of less than 1 in 10,000 per year of reactor operation and LERF of less than 1 in 100,000 years.

The above goals are implemented by stating acceptable criteria by which the plants can operate. The Commission believes that a 0.1 percent ratio reflects both qualitative goals and that it is low enough to expect people in the proximity of nuclear power plants not to have special concerns. It is intended that by using results from the Seabrook NPS seismic PRA, HAZUS seismic risk results and other resources, acceptability criteria, in the spirit of the USNRC safety goals, will be developed for seismically induced risks from nuclear plants.

5.2. Comparison Basis

Determining what risk is small enough would require delving into existing controversy concerning the specification in numerical terms of "How safe is safe enough?" The USNRC utilized a comparison mode, i.e. comparing the risk from NPPs to total mortality risk of the entire U.S. population. This appears to be very conservative to some observers. One could easily argue that a more appropriate basis for comparison would be risk from non-nuclear accidents or alternative electric generating technologies. The conservatism of the level of safety achieved by a proposed safety goal/policy would depend on what risk is chosen for comparison. Any choice made is bound to be based on a value judgment and open to debate.

Earthquakes that are strong enough to damage a reactor are likely to cause much damage in the region around the reactor itself. The possibility of increment of the seismic risk caused by NPPs cannot be reduced to zero. Considering that the seismic events themselves give rise to these incremental nuclear seismic risks, it is only reasonable to determine how small these risks can be made by comparing average background seismic risks and the incremental seismic risks due to nuclear power plants. By attempting to do this, some problems arise on how these risks can be compared. Some physical units are required to quantify the risks to be compared. Some quantities that have been, and are still used to quantify consequences include:

- Number of Fatalities (immediate and delayed)
- Number of injuries
- Number of illnesses
- Lost workdays
- Money loss in property damage and general suffering

- Reduction of life expectancy

The factors and terms included in each consequence above could differ depending on which methodology is used to analyze them. For our case, there are some differences in the definitions of the above (see chapter 4). Interpretation of consequences such as lives and injuries in terms of a single unit is also not an easy task as different studies and experts differ on how this should be done. When comparing the nuclear seismic and direct seismic risks, one has to decide on what bases they are to be compared. There are various forms of comparison that can be done that would be useful for setting a safety policy. Some bases of comparison are outlined next.

5.2.1. Spatial Basis for Comparison of Risks

One basis that can be used for risk comparison is spatial. One of the results obtained from the direct seismic risk analysis is an epicenter distance vs. magnitude of earthquake relationship. Figure 5. 1 shows this in graphical form. The graph shows how far the epicenter of an earthquake (of various magnitudes) has to be from Seabrook NPP so that Seabrook will have a PGA of 0.25g (its SSE). There are two different confidence radii curves shown for uncertainty. The median and 97% confidence level curves. If one picks an earthquake with a magnitude 7.9, for example, the corresponding distance on the 97% confidence level curve is approximately 135 km. This mean that the modelers are stating that to get a 0.25 g PGA at Seabrook, the greatest distance that the epicenter of an earthquake with magnitude 7.9 can be from Seabrook is 135 km. They are stating this with 97% assurance (certainty). If the median curve was used the median curve was used instead, the result gotten comes with a 50% certainty. As can be seen from this figure, an earthquake that is strong enough to damage the NPP could be located at a large distance from the plant. Thus, it would affect large areas. As the magnitude of the earthquake grows, the affected area size and its direct consequences also grow steadily. This is inversely proportional to the frequency of the earthquake, which decreases. In contrast, strong earthquakes can only cause so much of the plant's radionuclide inventory to be released, as this is a finite quantity. What we get is the conditional number of casualties affected by a large earthquake not increasing as the earthquake magnitude grows, rather the conditional probability of the release grows, while the frequency of the earthquake decreases. Therefore, the size of the areas affected by nuclear consequences will not grow steadily like the areas affected by direct consequences will grow, for large earthquakes. Three spatial bases considered for comparison of direct and NPP related seismic risks include:

1. Region Affected by radiation effects
2. Political region that encompasses most of the direct and nuclear effects
3. The nation

5.2.1.1. Risk Comparison Within Region of Radiation Effects

From Figure 4.10, it is seen that the study zone is sized to encompass the radiation effects of the Seabrook NPP. The emergency planning zone (EPZ) used by the NRC for latent cancer evacuation is 50 miles and this was used for radiation effects zone. The estimated mean risks for earthquakes of different magnitudes within that zone are shown in Figure 5. 2. These earthquakes are centered in Seabrook where majority of the radiation effects will take place. A 10-mile EPZ is used for the early fatality because the radiation beyond 10 miles of the site is not enough to cause prompt fatalities. Represented in the figure are the expected fatalities from a seismically (different PGAs) induced nuclear accident (without and without evacuation) and the expected fatalities from the occurrence of an earthquake. The earthquakes are centered in Seabrook. It can be seen that for strong earthquakes, the prompt direct consequences are much greater than the prompt nuclear consequences. The prompt direct effects are greater by factors ranging from about 100 to 500 than the nuclear risks (even without evacuation) from the same sized earthquakes. The latent nuclear consequences are greater than the direct consequences at lower PGA levels but become less than the direct consequences at higher PGA levels. The latent fatalities are not immediate and can take up to 40 years to take effect and there is some considerable uncertainty in its evaluation. HAZUS does not provide a similar analysis in considering possible future deaths from the direct earthquake consequences. Therefore, comparing the direct consequences to the latent fatality consequences from the NPP is not fair. Following sections will consider means of comparing the fatalities equivalently.

5.2.1.2. Risk Comparison Within the Zones Encompassing Most of the Direct Consequences

From Figure 5. 1, it is observed that the size of the zone in which earthquakes are strong enough to affect the NPP is in the order of the New England region. For this comparison in the New England region, the earthquake was also centered in the center of a major city, Boston, to see the difference in direct effect consequences based on the origin of the earthquake. Table 5. 1 (all the terms in this table are defined in Chapters 3 and 4) shows the magnitudes of earthquakes in terms of PGA and its consequences. It also shows the respective PGA expected at Seabrook when the earthquake originates at Boston. A comparison of prompt fatality risks for the region is displayed in the graph of Figure 5. 3. The figure shows the expected prompt fatalities from the direct effect of an earthquake and a seismically induced nuclear accident. Two results are shown for the direct effect of earthquakes. One set of result is based on an earthquake that is centered in Seabrook while the other result is based on an earthquake centered in Boston. The nuclear risks from the NPP are based on earthquakes originating at Seabrook. As can be expected, the fatalities for the earthquake centered in Boston are much larger than when centered in Seabrook. Table 5.1 shows that an earthquake centered in Boston need to have a very large magnitude to have any effect on the plant in Seabrook. An earthquake originating in Boston with a maximum horizontal PGA of 1.185g will lead to a maximum horizontal PGA of 2.443 (which is less the SSE of the Seabrook NPP) at Seabrook.

5.2.1.3. Risk Comparison on a National Basis

In order to compare seismic risks on a national basis, a seismic PRA for each NPP in the U.S. has to be performed. A level 3 PRA is needed so that their risks can be combined. The needed resources to perform this analysis were not available. Therefore, we treated the Seabrook NPP as a typical plant and its risks as an average. Subsequently, to obtain the total risks from NPPs, the risks from Seabrook NPP was multiplied by 103, which is the number of nuclear power reactors in the United States. From this, the PGA dependent and total fatality frequencies are obtained. These are summarized in Table 5. 2. It will be noted, from the table, that the higher fatality total value of 0.0396 fatalities annually (without evacuation) is much smaller compared to the national mean direct fatality of about 8.2, which was obtained from over the past 90 years.

5.2.2. Comparison of Different Types of Risks

In the United States, primary safety regulatory emphasis is placed upon fatalities. There are also other types of consequences due to NPP and direct earthquakes that are of interest and used for regulation. They include injuries, economic damage, morbidities, genetic effects, etc. In this section, comparison of various forms of fatalities, injuries and financial losses will be examined. As mentioned in previous chapters, the bases of comparison are not perfectly consistent as the methodologies used have slight differences in the way some consequences are measured.

5.2.2.1. Fatalities

There are prompt fatalities and delayed ones. The delayed fatalities are associated with the nuclear related risks due to accumulation of radiation that can lead to cancer or some form of genetic defect in the future. Fatalities due to direct and nuclear seismic accidents in the Seabrook area are presented in Figure 5. 2 and it shows both prompt and latent fatalities.

The science of determining the latent cancer fatalities from a radioactive release from a NPP is not perfect and is a work in progress. Hence, there is a significant uncertainty in the result shown. To get a more consistent basis of comparison with the direct consequences, we looked for means of standardizing our units used for fatalities.

5.2.2.2. Standardization of Risk Unit

For the comparisons being carried, it is important that the consequences being compared are as alike as possible. This means that consequences like early fatalities, in the scenarios being compared should not have totally different meanings. Also, there are some consequences like latent cancer fatality and genetic effects that can be due to nuclear seismic risk but do not exist for direct seismic risk. One of the means that have been used in the past is to attempt to create a standard unit, like monetary units, for all consequences. These were considered to make risk benefit analysis possible in previous studies. However, the possibility of pricing fatalities, sufferings, and other deleterious consequences has always been elusive. One looks at the results presented in the previous sections and has reason to be concerned regarding the comparison made in fatalities. Latent cancer fatalities are not immediate and should not be just counted as an early fatality. And since the people die, they cannot be counted as injuries either. To make this comparison analysis better, there is a need to standardize the units for the fatalities so that the results are more objective. What method should be used standardization turns out to be another bone of contention.

Two methodologies for standardizing the fatalities were used to see the difference obtained in our results and mainly to present to the panel of experts later to decide which one they found more objective. Both methodologies are suggestions. They are:

A) Life Shortening

In this part, the Life Years Lost (LYL) methodology is utilized. The LYL is the number of years a person would have lived in the absence of death. According to the U.S. census data, the average American is a 35 years old individual. This average person's life expectancy is about 40 to 45 years. This average age is assumed for people who had prompt fatality. For latent cancer, cancers due to radiation release from a plant are expected to develop from 10 to 50 years after exposure to the people. The affected individuals are expected to lose an average of about 20 years. The fatalities derived are converted to LYL unit to compare. The results from using this methodology are presented in Table 5. 3 and Figure 5. 4. Using this methodology does not show any major change in the way the results are presented using fatality as the unit. There is a reduction in the difference between the latent and direct consequences though.

B) The Risk Conversion Factor Methodology[18]

Dan Litai of MIT used this methodology in his PhD thesis (1980) when he proposed a new approach for risk comparison. Risk Conversion Factor (RCF) is a societal risk- response based ratio. If there were two distributions that represented two risk-types that were exactly similar with respect to all factors, then the ratio of these two distributions would provide how much people are more aversive to one risk than the other. The ratio of these two distributions is also referred to as a RISK-CONVERSION-FACTOR (RCF). The ratio of these two distributions would also give us a measure for subsequent comparisons of similar risks. For practical purposes, point values of RCFs are required and derived from the resulting distributions. Good approximations of RCFs may also be derived by simply dividing the median or mean values of the component distributions. For nuclear energy, using results obtained from WASH-1400, the RCF between early and delayed risk was calculated to be 30. Therefore, an equivalent one-category representation may be carried out by multiplying the early fatality by 30 and adding to the delayed fatalities, or by dividing the delayed fatalities by 30 and adding it to the early fatalities. The results for these are shown in Table 5. 4 and Figure 5. 5. The total early fatality in Table 5.4 was evaluated by dividing the latent fatality by the RCF of 30 and adding the result to the early fatality (without evacuation) from the Seabrook NPS PRA. Figure 5.5 shows that by using this methodology, the direct fatalities are large compared to the overall fatalities nuclear seismic related fatalities.

5.2.2.3. Injuries

Chapter 4 discussed the injuries from both direct and nuclear related consequences. The comparison is shown in Figure 5. 6. Once again the, injuries from the direct seismic event is larger than the injuries from the nuclear seismic related event. The injuries from the nuclear related event is not as easily determined as fatalities; thus the estimates shown have some subjectivity and increased uncertainty. There is also uncertainty associated with the injuries evaluated by HAZUS.

5.2.2.4. Economic Impacts

The financial losses incurred after an accident are difficult to estimate. One could easily forget or choose not to include costs that might turn out to be relevant. In the case of very large earthquakes, economic damage can be large enough to be of serious concern. This has been observed from the various earthquakes that have occurred in the past. The economic damage as estimated by HAZUS is mainly concerned with the value of destroyed infrastructure and the costs of repair. However, for Seabrook NPP, the estimated economic impact is concerned mainly with losses due to condemned property, the costs of decontamination and the costs of evacuation. Both can be seen to lacking some completeness. Some economic effects not considered by either model include the cost of health effects and economic multiplier effect (like effect on employment in one area due to loss of jobs in another that is close). For the NPP, there is the opportunity cost of lost revenue while it is shut down and buying from other sources to meet its obligations. Comparison between direct economic consequences and nuclear seismic-related economic consequences is shown in Figure 5. 7. The figure shows a comparison of economic loss expected, in Seabrook Area, after earthquakes of different peak ground accelerations occur and the economic loss expected due to the radionuclide release from the Seabrook NPP.

A closer approximation of economic damage expected from Seabrook NPP is analyzed by evaluating losses due to reduced market values for clean property in the vicinity of a radiation release, losses of markets for goods suspected of contamination and deterred tourism. Figure 4.12 outlines the value of housing in the vicinity of the Seabrook NPS and the annual market value of tourism in the New England States. The values in the figure present an extreme of total release; however it is assumed that a reasonable large fraction of these values could be lost in a NPP accident regardless of the actual amount of radioactive material released.

5.2.3. Comparison To All other Risks

A plausible safety comparison basis for the establishing a safety policy for NPP-related seismic risks is to require minimizing the risks compared to other general risks faced by the population (similar to the USNRC Safety Goals). To get a general idea of what kind of risks are faced by the general populace of the country, Table 5. 5 presents the fatalities in the country in 1998[19] by different means. It is noted that direct earthquake fatality risks are small compared to other accidental, non-accidental and disease risks. And as seen in previous sections, the nuclear power plant-related seismic risks are much smaller than the direct earthquake risks. So compared to the overall risks, NPP-related seismic risks contribute very little to incremental risks.

5.3. Mitigative Actions for Nuclear Power Plant-Related Seismic Risks (ALARA)

Towards the end of mitigating hazards in the U.S., the government has usually required investments to either reduce or eliminate the risks of some activities. Some mitigation actions that can be taken for earthquakes affecting nuclear power plants include:

- Structural backfits to plants
- Equipment of potential affected population with protective gear against radiation.
- Provision of potassium iodide pills to prevent thyroid cancer
- Provision of better communication for and between residents
- Provision of temporary shelter
- Structural backfits to homes in plant vicinity
- En-mass evacuation of resident (most likely by airlifts)

The costs of averting some consequences were analyzed by Yingli Zhu of MIT and presented in Figure 5. 8. Shown are costs of averting thyroid nodules, fatalities and injuries from pre-accident preventive actions of providing potassium iodide (KI), filters to reduce internal doses due to inhalation of radioactive materials and respirators for the previous purpose. The preventive methods prove to be costly and place equivalent valuations for human life above what society has required in other instances. Seismic retrofits of NPPs are not very cheap either. They are expected to be more expensive than the methods we have investigated. Required mitigative actions in the U.S. arise from the principle of ALARA where these actions are to be taken if possible at low marginal cost.

When considering the ALARA principle, the NRC uses a \$2000 per person-rem conversion factor, limiting its scope solely to health effects. This is based on a relatively simple logic in which the dollar per person-rem conversion factor is defined as the product of the dollar value of the health detriment and a risk coefficient that establishes the probability of health effects as a result of low doses of radiation. NRC used a probability of 7×10^{-4} per rem (in agreement with figures from the NCRP and ICRP) that includes allowances for fatal cancers, non-fatal cancers, and severe genetic effects. \$3 million was adopted as the dollar valuation of health detriment. Multiplying these two gives the \$2000 conversion factor. According to the National Academy of Sciences Committee on Biological Effects of Ionizing Radiation (BEIR) report, 1 man-rem of exposure to the whole body induces 1.8E-4 fatalities. For the ALARA standard, this implies spending about \$11 million per fatality averted. The results of mitigative costs can be seen to be greater than this.

5.4. Safety Acceptability Criteria Proposal for Nuclear Power Plant-Related Seismic Risks

The purpose of all the analyses and comparisons presented in the previous section is to suggest means by which acceptability criteria to be used for safety policy can be developed. Developing acceptability criteria is not only a technical task but involves much socio-political decision-making also. The decision makers have to decide what they believe is safe enough for the population in the vicinity of the nuclear plants. Based on the analysis of Seabrook NPS's PRA, the following acceptability criteria were suggested for a safety policy for nuclear seismic risks.

5.4.1. Formulation of Acceptability Criteria for Nuclear Power Plants

This is exactly like the USNRC safety goals. This is an overall integrated type of goal that depends on the comparison of the overall risk from a NPP to some predetermined risk level. This would require no special criteria or policy for the NPP-related seismic risks but considers the overall risk of the NPP. The seismic initiated risks are considered with other risks in the NPP and the NPP decision makers can decide where they will choose to reduce their risks to meet the safety goal risks. The safety goal risk is derived from already existing and established risks e.g. a fraction of the risk from other industries or alternative plants risks or general risks faced by the population.

5.4.2. Multi-Part Acceptability Criteria for NPP

As mentioned previously, requiring an overall integrated or event–specific goal on their own offers some disadvantages. These disadvantages could be reduced if both were used in conjunction. This means additional or supplemental criteria could be required for seismic risks (or any major external event risk that poses a major threat to the NPP). For seismic risks, it would be required that

$$\left[\frac{\text{NPP Seismic Risks }(\alpha)}{\text{Other Seismic Risks }(\alpha)}\right] < \text{L}$$

Other risks are those of the context in which nuclear power plant risks are to be assessed (the direct societal seismic risks is used in this case). α is the confidence level at which contrasted risks are to be evaluated (the choice of the value of α is a social decision). L is a value selected by the regulatory authority. As the magnitude of the earthquake increases, it is expected that the nuclear power plant-related seismic risks will reach an asymptotic (as the maximum radioactive releases are attained) and the direct seismic risks will grow monotonically because the affected area increases and there is also an increase in damage density. This means that the ratio

$$\left[\frac{\text{NPP Seismic Risks}}{\text{Other Seismic Risks}}\right]$$

will decrease as the magnitude of the earthquake increases.

However, to deal with the uncertainties involved in the results gotten, it will be required that both the NPP and other risks should be evaluated:

- On a consistent basis.
- Including the same factors of uncertainty, such as PGA, attenuation, ground motion frequency, structural response and consequence factors and units.
- Propagating uncertainties to provide distribution functions for consequences.

Even though this acceptability criterion was formulated with seismic risks in mind, one can envision it being extended to other external events like Hurricane, Floods, Tornadoes, etc. This proposal makes sure that NPPs do not contribute much incremental risk to the risks already faced by the public. The regulating entity can also ensure that there is a more complete structure of safety for NPP whereby overall safety of the plant and safety with regards to external events are kept to a small amount. While the first part of the proposal is already in place and is used to ensure that the plants do not add any unnecessary incremental risks to society, the second part prevents unnecessary regulatory requirements on the part of the regulators in demanding high marginal cost response from the NPPs, in the name of defense-in-depth. NPPs located in regions with high frequency of earthquakes will appreciate this proposal.

5.5. Review of Proposal By Panel of Experts

To complete this project, a panel of experts in NPP risks and policy formulation was assembled for a one-day workshop at MIT. The results of the analyses performed and proposals were then presented to them for their review. A summary of the proceedings of the workshop is attached in Appendix A. The conclusion of their review was that while they could understand the comprehensive comparison that was the backbone of the proposals, they did not think this was necessary in the United States. One main argument against the comparison analysis was that the two kinds of risk being compared were not fundamentally different (like comparing apples and oranges), i.e. while the risks from the NPP have some benefits, the risks from the direct earthquakes do not. This argument, however, is overlooking the fact that this mode of comparison presents a good and understandable means of communicating the risks to the public. These comparisons put the risks from the nuclear power plant in perspective with the external event that caused them.

An acceptability criterion which was suggested at the workshop and agreed upon by all participants was the Three-Region approach, which is presented next.

5.5.1. Three-Region Approach to Safety Policy Proposal

This approach is basically illustrated in Figure 5. 9[20]. There are three regions which express quantitative risks limits. The bottom region is the acceptable region and plants whose risks lie in this region are considered acceptable. NPPs whose risks fall in the top region are considered unacceptable and would be required to improve their risk status so that they fall in the middle region. In the middle region, NPPs are accepted based on the ALARA principle whereby increment in risk reduction is balanced against its costs to see if it is worth undergoing. Each individual plant is expected to meet these criteria. The seismic nuclear risk can be checked with these criteria when the risks from a nuclear power plant are categorized in terms of the initiating events. The risk from each initiating event is placed in the different regions they fall into. Then, the initiating event that contributes most to the total risk, can be seen. From this, one can attempt a risk reduction, should it be justified in cost-benefit terms. This way the NPP-related seismic risk can be reduced or left as it is depending on where it falls on the region table.

The value of the two boundaries that separate the regions is very important in the formulation and proposal of this policy. These values are based on societal decisions of what risk levels are acceptable and not acceptable. The decision makers such as the USNRC choose these boundary values. For example the current safety goal (0.1 percent of the sum of prompt fatality risks from other accidents) in place could be used as the lower region boundary value. Any NPP that reaches or falls below this goal is considered safe enough. For the upper boundary, the decision makers can determine what level of risk would not be an acceptable incremental risk to the society. How the decision makers come to this conclusion will be left to their discretion; a possible value of risk would be one at which the benefit gained from the NPP is not worth the risk it poses. Any NPP with risk greater than this will be considered unsafe and required to reduce its risk to fall in between both boundaries. The NPPs with risk between the upper and lower boundaries are considered acceptable.

The quantitative risk limits units used could be the CDF, LERF or consequences. All three can be derived from a PRA. It is up to the decision maker to choose which he/she prefers. The utilization of CDF and LERF reduces uncertainties in risks and ensures coherence in the evaluation of the risks, while the level 3 consequences would be more appropriate means of measuring risk and would easier to understand by the public. The quantity of risk limit for the two boundaries is to be determined by the regulators who use this approach.

Max PGA (g) in Boston	PGA at Seabrook (g)	Level 3	Level 4	Hospital Functionality	Fatalities
0.3461	0.0386	14	13	73.83	16.6638
0.4313	0.0546	80	77	64.88	105.096
0.537	0.0866	344	318	54.47	474.6232
0.9357	0.1849	1814	1625	34.05	2821.333
1.185	0.2443	7439	7131	21.03	13005.58

Table 5. 1: Direct Consequences from Earthquake Centered at Boston

Maximum Horizontal PGA	0.2g	0.3g	0.4g	0.5g	0.7g	1.0g	Total
With Evacuation	1.75E-05	3.89E-05	1.75E-04	8.58E-04	3.81E-03	1.15E-03	6.05E-03
Without Evacuation	1.14E-04	2.54E-04	1.14E-03	5.61E-03	2.49E-02	7.54E-03	3.96E-02

Fatality at a PGA * Frequency of Occurrence of Earthquake With that PGA * 103 NPPs in the U.S.A

Total = Σ National Fatality at each PGA level

Table 5. 2: National NPP-Related Seismic Consequences (Prompt Fatality)

Maximum Horizontal PGA	0.2g	0.3g	0.4g	0.5g	0.7g	1.0g
Early Fatality (with evacuation)	4.63E-04	3.70E-03	3.88E-02	4.23E-01	1.97E+00	2.98E+00
Latent Fatality	3.80E+00	1.39E+01	5.96E+01	1.45E+02	3.51E+02	5.06E+02
Early Fatality LYL#	1.94E-02	1.55E-01	1.63E+00	1.78E+01	8.26E+01	1.25E+02
Latent Fatality LYL##	7.60E+01	2.77E+02	1.19E+03	2.90E+03	7.02E+03	1.01E+04

 # Early Fatality*Average Years Lost per Life (42yrs)

Latent Cancer fatality*Average Years Lost per Life (20yrs)

Table 5. 3: Seabrook NPP Life Years Lost from Early and Latent Fatality

Maximum Horizontal PGA	0.2g	0.3g	0.4g	0.5g	0.7g	1.0g
Early Fatality (with evacuation)	4.63E-04	3.70E-03	3.88E-02	4.23E-01	1.97E+00	2.98E+00
Latent Fatality	3.80E+00	1.39E+01	5.96E+01	1.45E+02	3.51E+02	5.06E+02
Weighted Latent Fatality with RCF (1/30)	1.27E-01	4.62E-01	1.99E+00	4.83E+00	1.17E+01	1.69E+01
Weighted Total Early Fatality	1.27E-01	4.66E-01	2.03E+00	5.25E+00	1.37E+01	1.98E+01

Table 5. 4: Seabrook NPP Combined Prompt and Weighted Latent Fatality Risk

Fatalities in the U.S.A. (1998)

	1998
All Causes	2338100
Accidental Deaths	93200
Diseases	1952100
*Other Causes of Prompt Fatalities	292800

*Include Ill-defined deaths, unspecified cause, infant death, Natural Causes, suicides, Homicides

Non-Accidental Deaths (1998)

Heart Disease	724300
Cancer	538900
Pneumonia and Influenza	94800
Hypertension	14200
Suicide	29300
Homicide	17400

Accidental Deaths (1998)

Tornadoes**	52
Floods**	94
Hurricanes**	23
Lightning**	60
Motor Vehicle	41800
Water Transport	621
Railroad	1008
Air Transport	672
Falls	16100
Fires	3000
Electric Current	482*
Firearms	800
Drowning	4200
Poison [solid, liquid]	9300
Poison (Gas, Vapor)	600
Earthquake (Mean)	8

*1992
** Mean Fatality over past 11 years

Table 5. 5: U.S. Fatalities from Various Sources

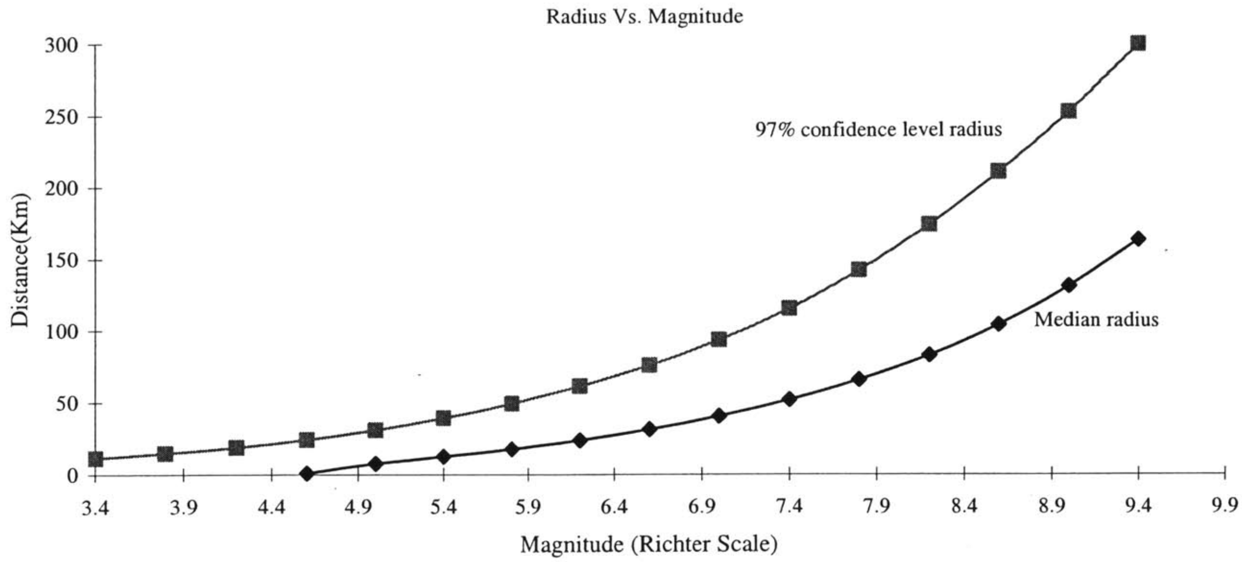


Figure 5. 1: Epicenter Distance Such That Seabrook NPS PGA is 0.25g

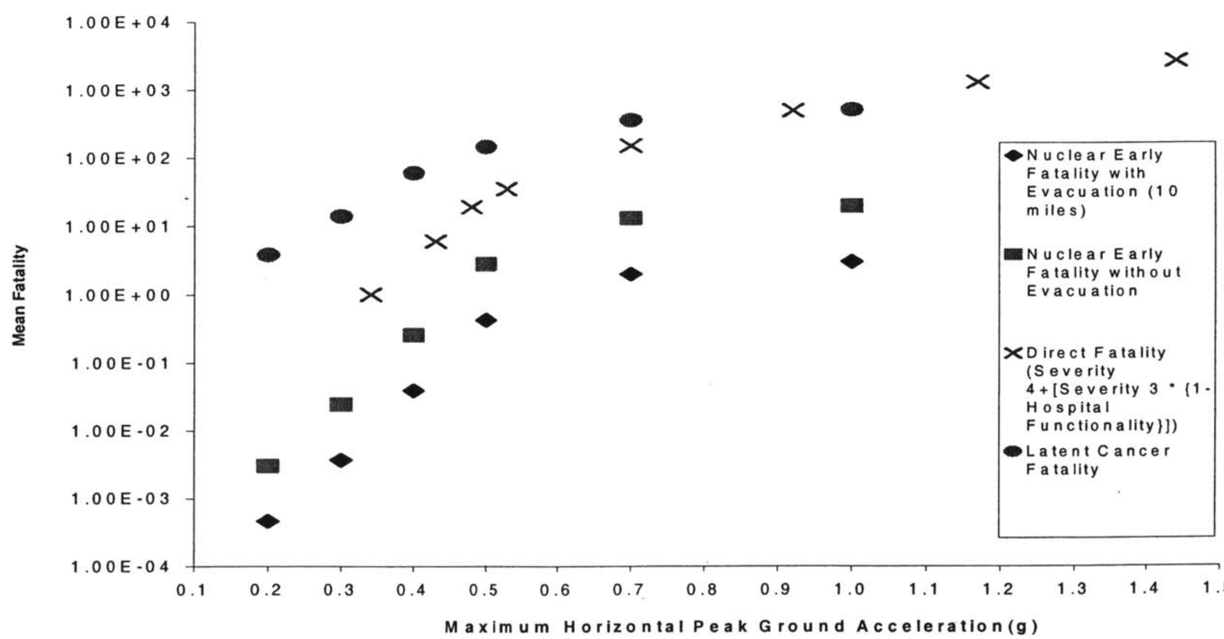


Figure 5. 2: Comparison of Conditional Prompt Fatalities for Earthquakes Located At Seabrook

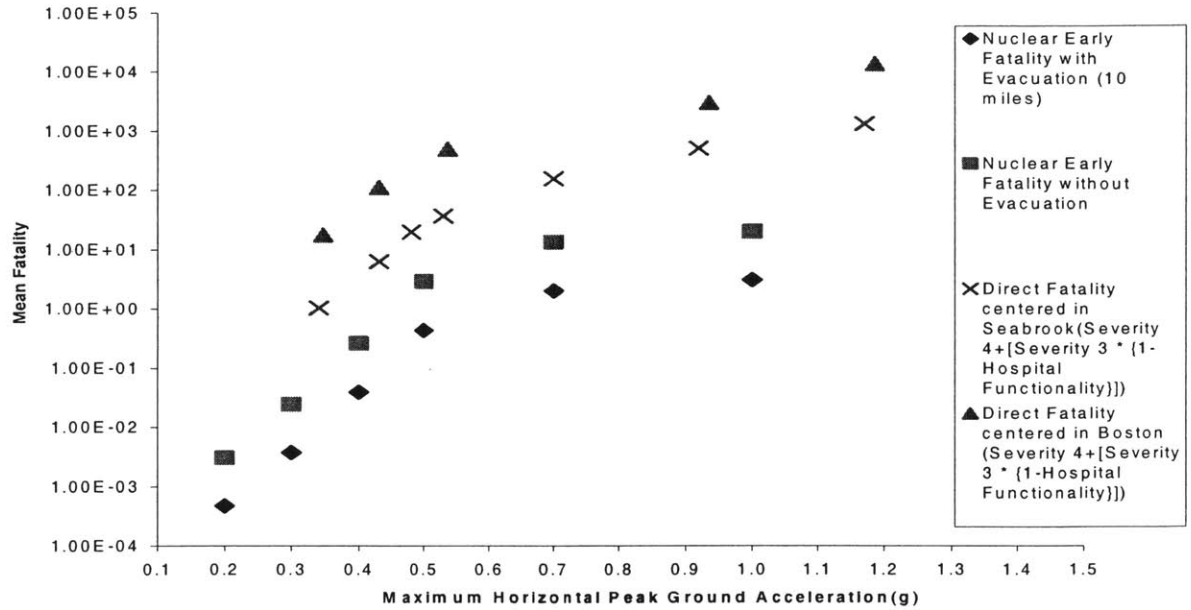


Figure 5. 3: Comparison of Conditional Prompt Fatalities For Earthquakes Located At New England Study Region (Boston and Seabrook)

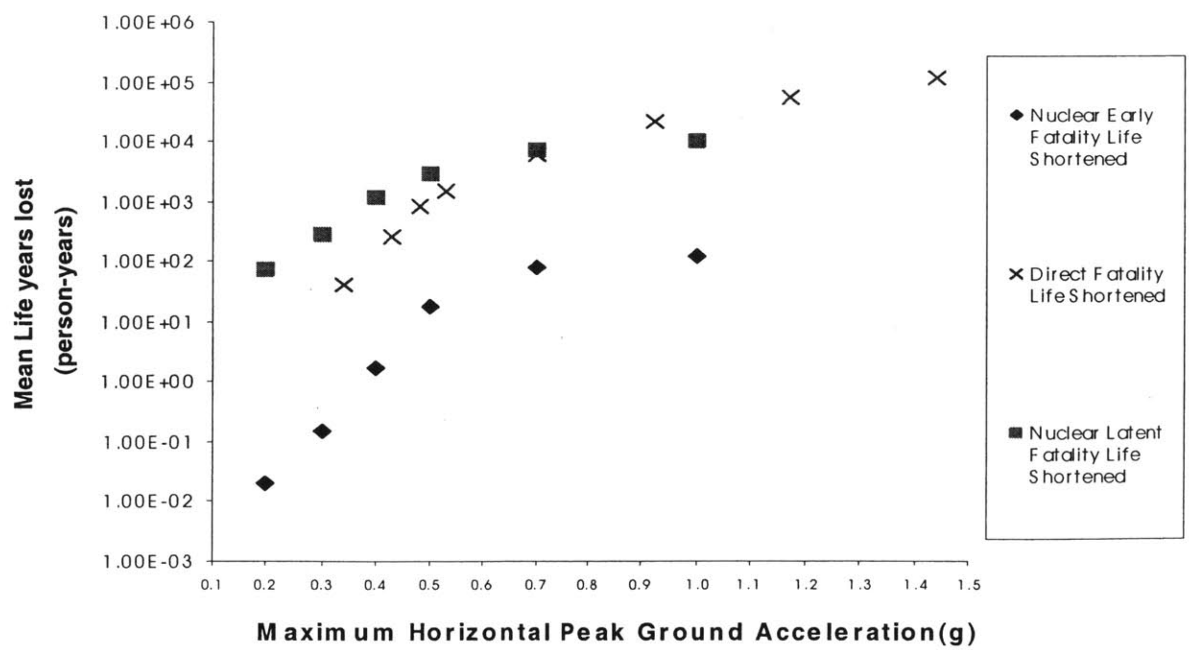


Figure 5. 4: Comparison of Conditional Fatalities using Life Years Lost Methodology

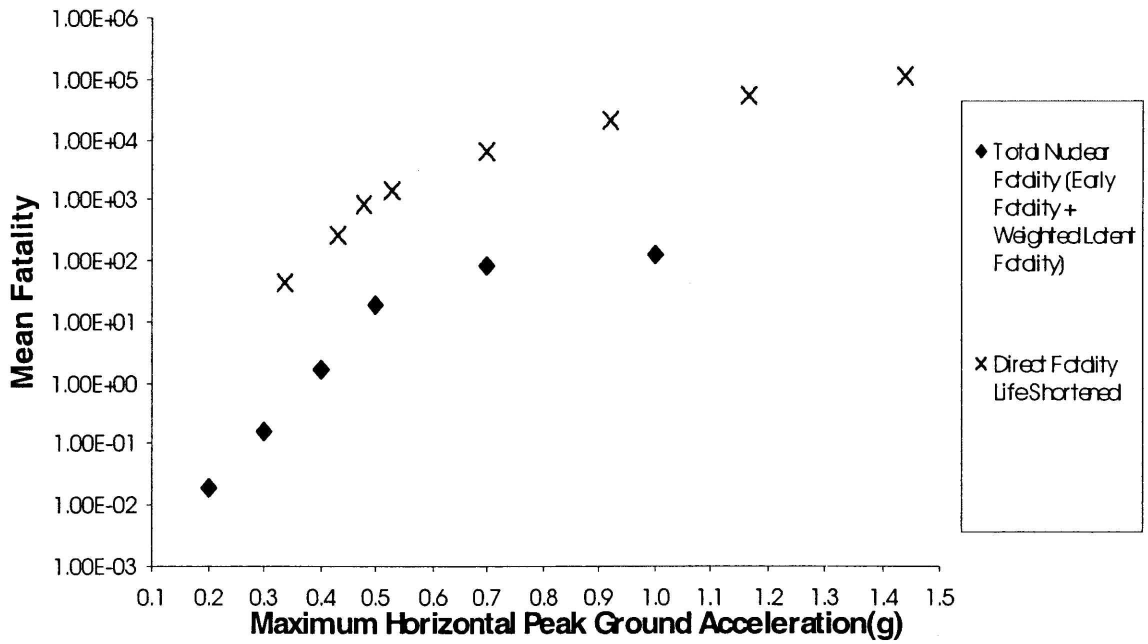


Figure 5. 5: Comparison of Fatalities using Risk Conversion Factor of 30 from Litai

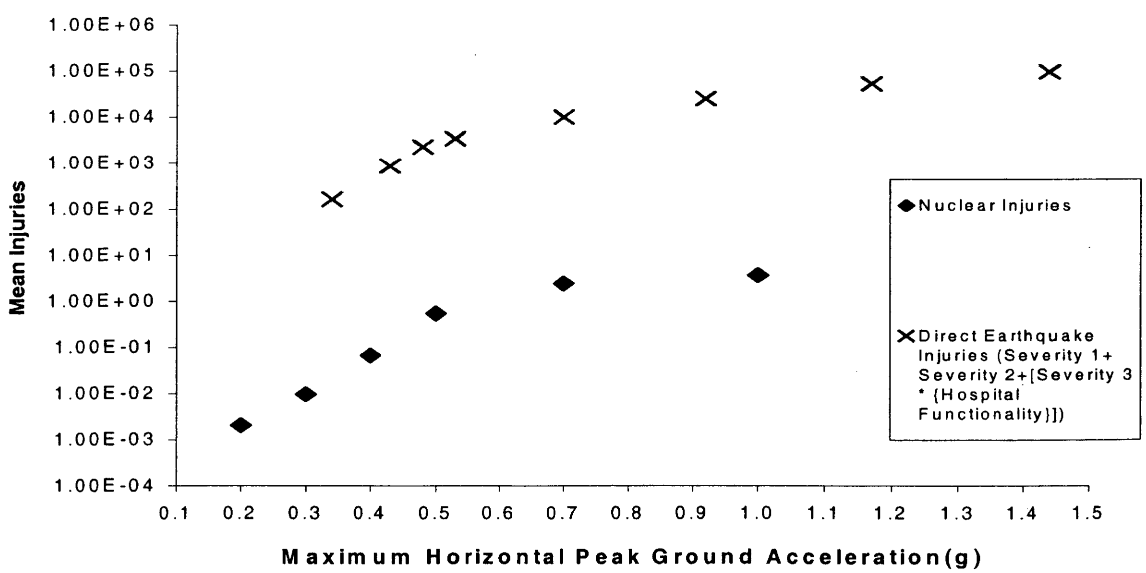


Figure 5. 6: Comparison of Injuries

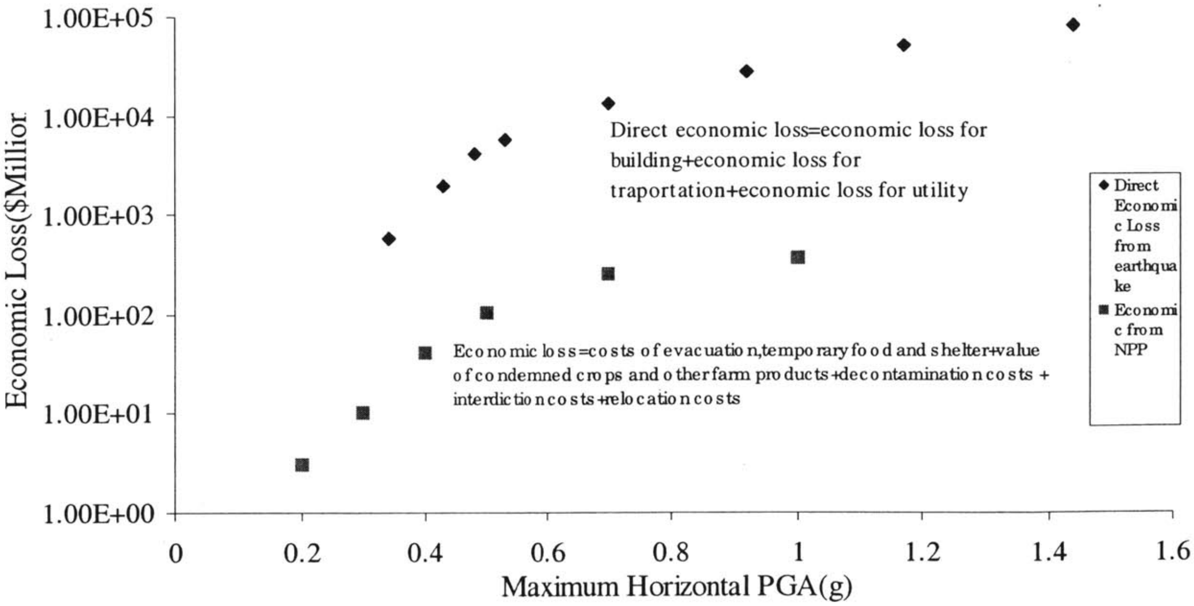


Figure 5. 7: Comparison of Direct Economic Loss From Earthquake effect, on Seabrook Area, to Economic Loss from the Seabrook NPP

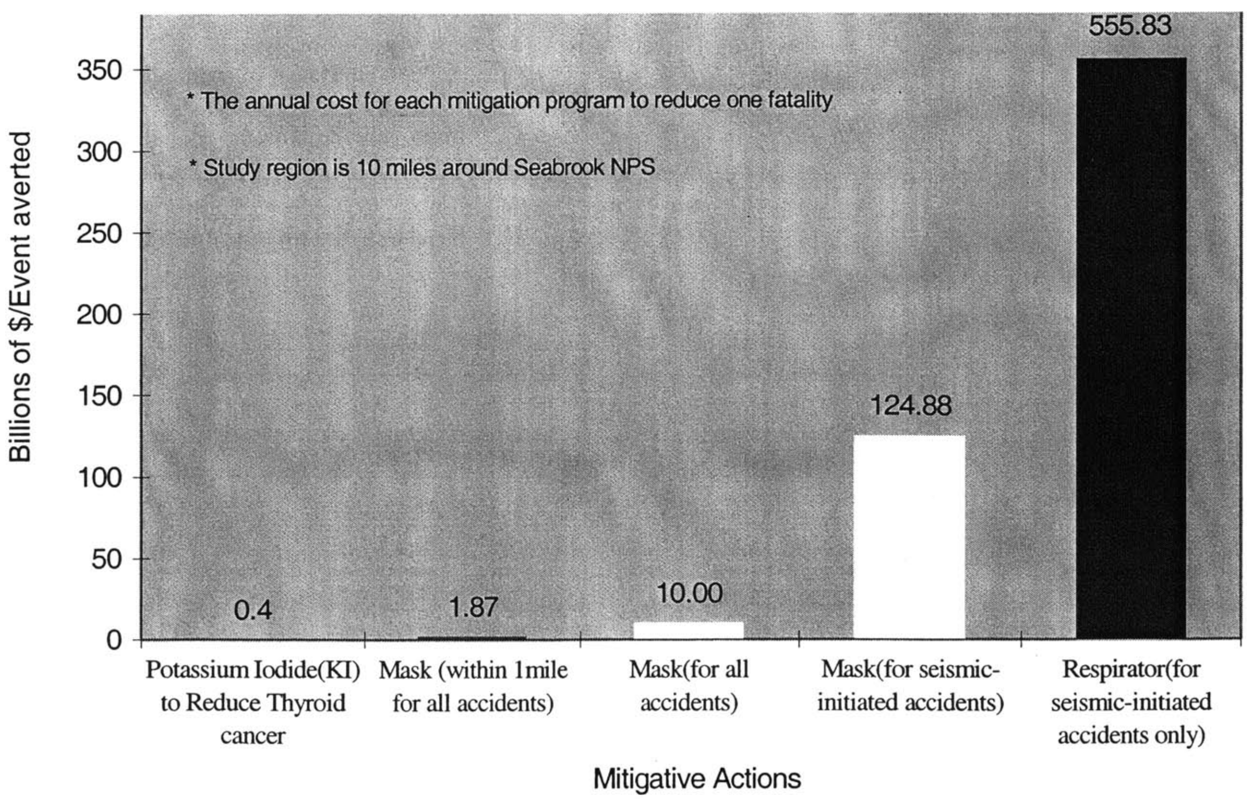


Figure 5. 8: Billions of Dollars/Event Averted Via Mitigation

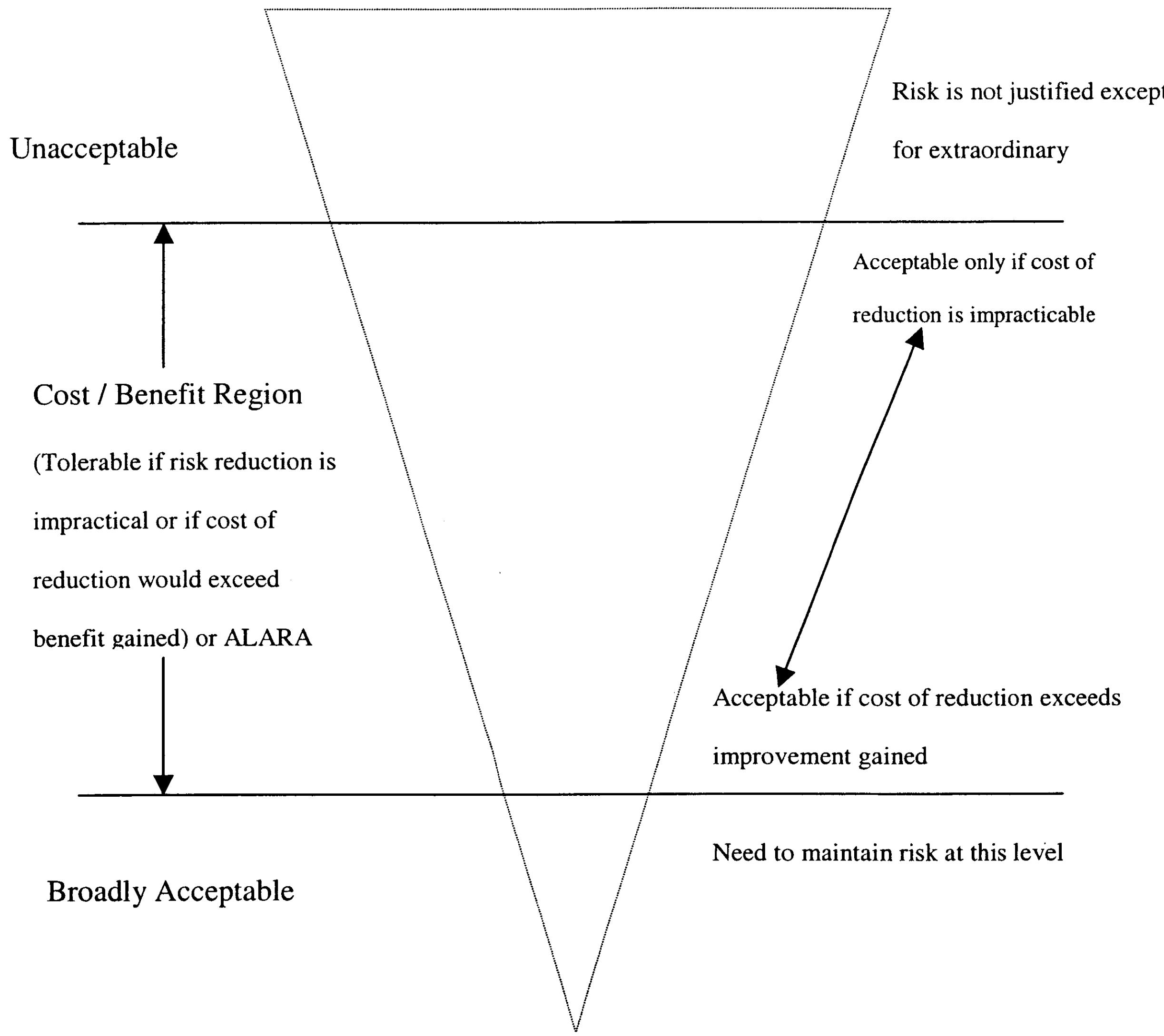


Figure 5. 9: Levels of Risk[20]

Chapter 6: Qualitative Analysis of Nuclear Seismic Public Safety Policy Related Issues

This chapter analyses the qualitative aspects involved in public safety policy. Using insights gained from the risk studies in previous chapters, suggestions are made for improving public safety in the vicinity of NPPs.

6.1. Stakeholders

In the protection of the health and safety of the public, there are many stakeholders. If an accident occurs, the public is best protected when the response by all stakeholders is fully integrated. Listed below are major stakeholders, in the U.S., involved in the protection of the public in the case of a seismically induced nuclear accident.

Federal Emergency Management Agency (FEMA): FEMA coordinates all Federal planning involving offsite impact of radiological emergencies. It also takes the lead in assessing offsite radiological emergency response plans and preparedness, determines the adequacy of implementing the plans and communicates its decisions to the NRC. FEMA also reviews state and local preparedness through regional assistance committees (RACs) for each standard federal region and will work with state and local governments on the resolution of deficiencies on a continuous basis.

USNRC: The USNRC, a creation of the Atomic Energy Act of 1974, reviews those FEMA findings and determines, in conjunction with its onsite findings, the overall state of emergency preparedness. NRC also makes radiological health and safety decisions in the issuance of licenses and oversees the continued operation of plants including taking enforcement actions as notices of violations, civil penalties and shutdown of plants.

State and Local Governments: Policies concerning safety and planning involve several levels of government: Federal, State and local governments, including counties, townships and villages. Local government jurisdictions are found within the 10-mile plume exposure pathway Emergency Planning Zone that affects early fatalities after an accident. Plans and response mechanisms by local governments are particularly needed and important for the 10-mile EPZ. However, the 50-mile EPZ for the ingestion exposure pathway may encompass one or several states as in our case study of the New England Region. Policies and planning on this level are best handled by State governments, with support from the local and Federal Governments. Making safety policies for the public depends on the population in the area of accident; hence planning and implementation of policies are better done on a State and local level, as long as resources are available, rather than on a Federal level. Also, local governments handle issues like housing planning and allocation of land; therefore, they play a major role in any proposed policy for public safety.

The Federal Government: As mentioned previously, the federal government plays an important role in the protection of the public. FEMA and the USNRC are federal Agencies. The Federal government also controls much money that will be needed for any implementation of any safety policy. Also, all NPP are federally regulated and under the federal government jurisdiction.

The Nuclear Industry: These include the NPPs and utilities that run them. They are primarily responsible for planning and implementing emergency measures within their site boundaries. They are also responsible for accident assessment including evaluating any potential risk to the public health and safety, both onsite and offsite, and making timely recommendations to State and local governments concerning protective measures.

The Public: These include the people who will be likely affected should an accident occur at the NPP. Any policy or plan enacted will not succeed without the cooperation of the public it is meant to protect. The public usually needs to be consulted or communicated with about policies in place or to be enacted for their protection.

6.2. Insights from Turkey Point NPS's Experience of Hurricane Andrew

An experience at a nuclear power plant relevant to the situation that would be encountered at a post-earthquake NPP is that of the Turkey Point Nuclear Generating Station (which consists of 4 units, Units 1 and 2 are fossil (oil) fueled (430 MW(e)) and units 3 and 4 are nuclear fueled (760 MW(e) PWRs) in 1992. In order to get a good idea of what could go wrong and actions that could be taken at a nuclear power plant when affected by an external event like an earthquake, an investigation was launched into previous relevant incidents from the past. A probable event in the US is occurrence of a strong hurricane. The most recent of such hurricane that presented a threat to a nuclear power plant was Hurricane Andrew[21], which crossed the coast of Florida in August 1992. Major hurricanes, particularly those rated in the highest categories 4 or 5, are of greatest concern to nuclear power plant. Category 4 or 5 hurricanes, on the average, strike the U.S. every six years. Andrew was a category 4 hurricane, one of the strongest to reach the United States.

On August 14, 1992, the National Hurricane Center noted a developing tropical storm and started plotting its position on a chart. On August 21, preparations started at the Turkey Point Nuclear Generating Station, using the Emergency Plan Implementing Procedure (EPIP), even though Andrew was still a tropical storm about 800 miles (1300 km) away. Activities undertaken included removing equipment from outside areas, tying down equipment; and preparing for the expected storm surge of ocean water. By noon on Sunday, August 24, the staff was ordered to begin shutting down units 3 and 4 at 6pm. Turkey point procedures required both units to be at least in Mode 4 (hot shutdown), two hours before the onset of the hurricane winds, which could prevent personnel from working outside and securing the plant. The plant was kept in mode 4 rather than mode 5 (cold shutdown) in order to retain the steam-driven auxiliary FW pumps as immediate backup method of removing decay heat.

Volunteers remained at the plant during the hurricane. They had been given time prior to the hurricane to ensure their families and homes safety. Operating crews took simulator training on scenarios likely to occur during the hurricane. Operators were also placed in each EDG control centers (Class I buildings) because they might not be accessible during the height of the storm. Technical Support and Operational Support centers were also established. By midnight, preparations for the storm were complete. Hurricane Andrew passed over the Turkey Point site in a westerly direction on August 24, 1992 with sustained wind speeds of 145 mph (233km/hr) and gusts of at least 175 mph (282 km/h). People within 10-mile (16 km) EPZ were already evacuated.

The lessons learned from the hurricane are as follows. The simulator training helped the operators to be more alert to likely plant transients. More careful supply preparation is needed because emergency supplies, such as food and water, ran out faster than anticipated. Not only the emergency supplies, but also equipment for debris removal off the roads were in shortage. The amount of time and effort spent on meeting the physical and emotional needs of the plant staff and their families far exceeded anything anticipated prior to the storm. Assistance received from the St. Lucie NPP made recovery much faster and more efficient. Offsite communication was lost; additional planning for restoration of communications (e.g. use of temporary satellite communications system provided by the NRC) could have speeded recovery had it been installed onsite before storm. Cellular telephones proved not to be as reliable as expected – due to heavy traffic on circuits and the limited ranges of individual telephones. Section 50.54(x) of Title 10 of the Code of Federal Regulations gave flexibility needed to adapt to the situations encountered. The provisions of this regulation allow senior personnel to take reasonable action that departs from a license condition or technical specification in an emergency when this action is immediately needed to protect the plant, the public health and safety and when no action consistent with license conditions and technical specifications that can give adequate or equivalent protection is immediately apparent.

Hurricanes, unlike earthquakes, can have advance warnings that help the plant, as well as the community, prepare for the disaster. But even with hurricanes, some damage and losses are inevitable. The important risk factor associated with hurricanes, earthquakes, and all other natural hazards alike, is how to reduce these damage and losses by properly planning and mitigation of foreseeable consequences.

6.3. Existing Policies

There are several policies and plans in place on Federal, State and local governments level already. They include NUREG-0396, NUREG-0654, State Emergency Operation Plans, and Radiological Emergency Response Plans, just to name a few. The purpose of this section is not to review everything in existence already; rather to look for means by which these policies can be strengthened or improved using insights gained from the risk study of NPPs.

Some basic principles[22] of policies regarding emergency preparedness and response planning include:

1. Plans should be adjusted to behavior of people rather than vice-versa.

2. Policies for emergency planning should be considered a process and not a product. There should be continuous updating, hazard assessment, public education, training and evaluation.

3. Policies should be based on scenarios that are as practical as possible.

4. Everyday plans used for normal emergencies are not enough for use in major disasters because factors such as social environment and communication loads are altered.

5. Policies concerning planning are not the same as policies for management. Planning involves tasks such as educational activities, reducing unknowns in a problematic situation, and evoking actions, while management includes tasks as warning, search and rescue and casualty care.

There are a number of weaknesses that can be identified in local emergency policies that involve violation of some of the principles listed above. Many local planning efforts involving outdated, non-exercised and command-and-control model plans that make erroneous expectations about group behaviors continue to be documented in post-event response studies[23],[24]. Some of the response plans studied are:

Seabrook NPS Radiological Emergency Response Plan

Seabrook NPS has a plan in case of an accident that is presented in a brochure so that the information is made available to the communities surrounding the Seabrook area. This plan can also be used in other emergencies such as floods, fires, hurricanes, tornadoes, or toxic chemical spills. It discusses emergency contact numbers, sirens and emergency alert systems (EAS), shelter, evacuation, pets and livestock, and information on Seabrook station and radiation.

State Emergency Operation Plan

The New Hampshire and Massachusetts Offices of Emergency Management have plans called the State Emergency Operation Plan, also know as the State EOP that is used in case of a disaster. The State EOP describes the basic mechanisms and structures that the state can employ in potential and/or actual emergency situations. The State EOP is not hazard specific. Its primary purpose is to initiate, coordinate, and sustain an effective State response to disasters and emergency situations. It is designed to identify planning assumptions, assess hazard potentials, and develop policies; establish a concept of operations built on an interagency coordination in order to facilitate an timely and effective State response; assign specific functional responsibilities to appropriate State departments and agencies; and coordinate actions necessary to respond to an emergency and coordinate the links between local governments, Federal response, neighboring States and the Province of Quebec, Canada.

New Hampshire Radiological Emergency Response Plan

The only hazard specific emergency response plan, other than terrorism, is the radiological emergency response plan (RERP). The New Hampshire Office of Emergency Management has developed their RERP in coordination with other state agencies and in accordance with the planning guidelines specified in NUREG-0654/FEMA-REP-1. Other States have their own RERP also. It describes the plan and emergency response capabilities needed in case of a radiological emergency at commercial nuclear power plants in or near New Hampshire. The RERP is site specific so that it describes plan for individual communities around potential hazardous areas. For example, a complete RERP for the town of Seabrook consists of Volume

20 (Seabrook Station Local Radiological Emergency Response Plan) and Volume 35 (Seabrook Plan Information and Implementing Procedures) of the NHRERP. Some descriptions contained in the RERP include:

- Classification of nuclear emergencies using the Emergency Classification Levels (ECL) outlined in Appendix 1 to NUREG-0654/FEMA-REP-1, Rev. 1.
- Methods used to notify the local EPZ and host community emergency response organizations, local officials, private organizations and the public, in the event of a nuclear emergency.
- Means to be employed to assess the offsite consequences of an onsite accident.
- Protective actions to be implemented by the emergency response organization and the public.
- Means for controlling radiological exposure of emergency workers involved in protective response activities.
- Exercises and drills to be conducted to evaluate major portions of the offsite emergency response capability.
- Responsibilities for development, review, update, and distribution of the plan and its associated procedures.

In the event of an actual or potential radiological release accident, there are several ways to protect the pubic. The protective actions can differ depending on various factors such as time, demographics, wind direction and velocity, and weather conditions. After these factors are taken into consideration, a protective action will be implemented to minimize direct exposure within the Plume Exposure Pathway EPZ and minimize indirect exposure within the Ingestion Exposure Pathway EPZ.

6.5. Possible Changes to Protective Actions Due to Seismic Nature of Nuclear Accident

6.5.1. Communication

As can be seen from the sample case of Turkey Point NPP, communication was one of the aspects of the preparation for the hurricane that did not function as wanted. Communications is used here to refer to the planning stage communication, which includes education of the public in advance and warning systems and communications after an emergency including use of radio, EAS and telephony devices.

6.5.1.1. Pre-Accident Communication

Public Awareness and Education

Information is a very valuable asset in the prevention of casualties before and after an accident. Making the public around a NPP aware of systems and plans in place for their safety in case of an accident will make a big difference in the way the public reacts when the incident does occur. Considering that if the incident is pre-empted by a seismic event, people might not act rationally, it makes it the more important to make sure that there is enough information available to them to reduce such irrational behaviors. Safety workshops at schools and workplaces, handing out of brochures and fliers, and periodic PSAs on radios and local TVs are some of such events that could be carried out. This is one area in which the utilities, which own the NPPs, could assist the local government. The importance of such information cannot be overstated. Depending on how it is handled, it can make other protective actions like evacuation and sheltering more efficient. A study to quantify how much such prior information reduces fatalities in the case of an earthquake would be helpful but is not part of the work done here.

Early Warning Systems

There are circumstances where early earthquake warning can be provided. The possibility of such a warning would only, realistically, be of assistance to regions, which are distant (about 100km) from the main fault source of the seismic hazard. A warning signal from a seismic detector close to the fault rapture could reach a distant community some seconds before severe shaking begins. This is due to the fact that the travel time of electromagnetic waves is negligible when compared with that of seismic waves. Seconds might seem like too little time but it might just be enough to perform some emergency preparation tasks like pulling a switch at a NPP, and starting evacuation procedures. Another major issue concerning the prediction of earthquakes and its potential efficiency is the reaction of the public that receives the warning.

There are a number of warning systems to warn major metropolitan areas. A demonstration showpiece is the Seismic Alert System (SAS) of Mexico City, sponsored since 1989 by the city authorities. The system consists of units for the seismic detection; telecommunications; central control, and radio warning. The detector system consists of series of digital strong-motion field stations, each of which monitors activities within 100km and detects and estimates the magnitude of an earthquake within 10 seconds of its onset. The central control unit in Mexico City is sent a warning if estimated magnitude of 6 or more is detected. The decision to broadcast an early warning is taken by the central control unit after receiving data from all the stations. General alerts are only sounded when at least 2 field stations estimate a magnitude at or above the threshold of 6. The performance of the SAS was demonstrated in the 7.3 magnitude earthquake of 14th September 1995, which was epicentered about 300km south of Mexico City. The distance allowed a 72 seconds waiting time from the time of the broadcast to the time of arrival of strong shaking. This was sufficient time for notification of emergency military and civilian response centers and orderly evacuation of schools in which an alarm system had been fitted.

6.5.1.2. During and Post-Accident Communication

Many instruments and communication devices are in place for communication during an incident and after one to communicate between the public and those in charge of mitigation. A list of such devices is listed in NUREG-0654. The experience from Turkey Point shows how tests and simulations of scenarios with these equipments are necessary. There also needs to be diversity in the form of communication devices available and some such as satellite phones that cannot be affected by floods or earthquakes should be included. Communication after a large event like an earthquake is almost as important as communication before one. The means by which the public in the vicinity of a NPP can receive information and instructions is something worth investing in by the utility. It could nothing more than a battery-operated radio to pick up radio signal for announcements after such an event.

Currently, nuclear power plants have various levels of emergency alerts. At every alert level, the plant has to inform the local government, which in turn decides whether to alert the public, or not. The highest alert is when a breach of containment occurs and there is a large release of radioactive materials into the environment. The power station informs the local government which in turns warns the residents. The notification of an emergency is accomplished through several different ways. Some of the means of alerting and warning the public include: a siren sounded aloud with steady tone lasting three to five minutes, a message on TTY (teletypewriter) for those who have registered with emergency management officials, alerting of ocean boaters by the U.S. Coast Guard and sirens, special announcements on local emergency alert system radio stations and broadcasts from loudspeakers on emergency. It could be argued that the transition between the release of information from the plant to the local government and the time it takes to make a decision is wasted time that could have been used to warn residents and give them more time to react to an emergency. It is being proposed here that power plants should be given the authority to inform the public directly. The plants should be allowed to give warning signals directly to the public. An emergency alert system of their own can also be put into place in order to increase successful alerting of the emergency. This could improve the amount of time that would be available for the public to evacuate or seek shelter. However, any action taken by the Nuclear Power Plant should not be an independent action apart from the governing entity. The plans should not only be complementary but they should also be redundant. Redundant procedures carried out by both the designated state agencies and the power station, with good communication, would ensure maximum success probability of the procedures.

6.5.2. Sheltering

The purpose of sheltering after a radiological accident is to provide protection from radionuclides that are released from the accident. When shelter-in-place is ordered, people should go indoors immediately and close all doors and windows. All window fans, air conditioners, clothes dryers, kitchen and bath exhaust fans, and any other sources of outside air should be turned off to limit plume from entering the building. Heavier construction materials or increased layers of building material increase the amount of protection from exposure to radiation. Therefore, shelter should be sought in the lowest level of the building such as in basement, away from windows. Sheltering can reduce both external and thyroid radiation doses. Detailed instructions on what to do when sheltering-in-place are listed in the "Expedient Sheltering in Place: An Evaluation for the Chemical Stockpile Emergency Preparedness Program" report.

However, in our scenario, sheltering-in-place or sheltering people in reception centers after evacuation cannot be solely counted upon because some of the buildings relied upon for sheltering might not survive the earthquake that leads to the radiation release. It is proposed here that there needs to be some scale of retrofitting for potential shelters in the vicinity of NPPs. The question would be who would incur the cost of these retrofits. Recent seismic retrofits in the Bay Area of California averaged around 25% of the house values. Attempting to seismically retrofit all homes in the vicinity of a NPP would be an undertaking of huge proportions. Lawmakers are unlikely to accept this especially for a low frequency event in parts of the U.S. However, areas designated as receptions centers and important lifeline buildings like hospitals, fire departments, and electrical substations can be retrofitted to be able to withstand possible large earthquakes. The ability of utilities to help in the area of sheltering would be require a cost benefit analysis on the part of the utilities to analyze whether it is worth the money that would be invested in the action.

6.5.3. Evacuation

Evacuation is the controlled relocation of a population from an area of known danger or unacceptable risk to a safer area. If evacuation is determined to be an effective protective action against radiological exposure, State officials are responsible for ordering the evacuation. When evacuation is ordered, instructions will be broadcasted over the Emergency Alert System. Evacuation is accomplished by means of automobile, if highways and bridges are accessible, or by walking, if necessary. Evacuees should drive away from the affected area to the reception center along designated routes. Emergency buses would be available (provided by the State Department of Transportation) if evacuation is ordered. The buses will transport people to reception centers. Information on bus routes is usually available through the Emergency Alert System radio stations and brochures.

Evacuation has its drawbacks like its costliness, ability to keep track of population and possibility of contamination during the process of evacuation. In our scenario, there is the added problem that the earthquake which led to the radiation release is strong enough to cause damage to the roads and thereby impeding effective evacuation of the area. Other means of evacuating the public like airlifting and evacuating by water, if next to a body of water, should be considered. The costs involved would make these alternatives unattractive to policy makers.

6.6. Other Policies and Protective Actions

Apart from the major ones listed above, other actions and policies in place to protect the public in the vicinity of power plants include the distribution of Potassium Iodide (KI) to the public. This reduces the potential latent fatalities from Thyroid Cancer if there is a release. The EPZ for early fatalities is 10 miles and that for latent fatalities is 50 miles. The bigger the EPZ, the slower the emergency responses would be and this would be detrimental to those nearest to the plant. According to the risk study of Seabrook NPS, using an EPZ of 2 miles with sheltering beyond would result in almost the same fatalities as having an EPZ of 10 miles. If this smaller EPZ is accepted, it is conceivable to propose that NPPs should be required to own all land within 2 to 3 miles radius of the plant and not permit any residential or industrial houses within this boundary. This would reduce the number of people likely to be affected by a release from the plant. A radiation monitoring system can also be implemented so that the path of radionuclides can be tracked. This would assist in determining which sectors around the power station need to be evacuated. However, these should be positioned in such a way that a possible earthquake would not affect them.

Policies by local and State governments can be better coordinated and the utilities can be permitted a more active role policymaking and implementation role. Afterall, the end result of all the policies is to ensure the safety of public that could be potentially affected. While not advocating a self-regulated industry (a possible policy alternative that would run into much problems), there needs to be more active involvement by the NPPs in the policy making for public safety. As mentioned earlier, seismic (with or without a consequent release from a NPP) safety policies are apathetic and the NPPs have much less to worry about than the various branches of the government and would be more sensitive to policies put in place for public safety than the governments would be. Areas involving direction and control, public education and warning, planning for coordination of medical facilities, accident assessment, and contamination monitoring need to be improved and re-assessed to make plans for them better representative simulations of real events of which seismic nuclear release is one of them.

D: Conclusion

Chapter 7: Conclusion

Earthquakes are recorded in every state of the U.S. and have great potential to cause much damage resulting in multiple disasters. Consequences from earthquakes include economical loss, environmental loss, injuries and death, etc. They are also common in other countries such as Japan, Taiwan and Turkey and have been the focus of numerous studies. However, because of the low probability of large earthquakes in some regions, it can be difficult to make it a national priority.

Earthquakes also occur in regions, which are locations for NPPs in the U.S. and internationally and there are some concerns that NPPs will pose additional risks, to the public in their vicinity, to the risks of the seismic events. Using Seabrook Nuclear Power Station and New England Region as sample cases, tools for risk assessment and loss estimations (PRA and HAZUS) were used to determine the consequences to the public for different, low probability, seismic events that can affect the NPP. This showed that, due to the defense-in-depth approach and good engineering, the consequences from NPP are less than the earthquake consequences by factors up to 500. The only exception is in the case of latent fatalities from NPPs. Using the results, some comparative analyses were made and used to propose quantitative safety policies/goals that could be used to monitor and discourage the increase in the risks from NPPs due to seismic events. Comparative analyses of the direct earthquake risks and the seismically induced nuclear risks are credible because they present a good means of communicating the risks posed to the public. The public can easily understand these comparative analyses and make societal decisions about these risks.

Results were presented to a panel of experts and their inputs were noted. Majority of them did not believe it was necessary to formulate separate safety policies specifically for seismically induced nuclear risks but that it should be kept in context in an overall safety goal for NPPs. However, this reaction from the experts could be attributed to the fact that seismic nuclear risks have never been much of a problem in the U.S. For a country like Japan with a high frequency of earthquakes, densely populated NPP regions, and reliance on nuclear energy, having a separate safety policy for seismically induced nuclear risks could be considered.

There are policies and emergency plans in place already in case of an accidental release of radioactive materials from NPPs or other sources. Plans are carried out under the jurisdictions of the Federal, State and local governments, with the utilities playing minimal support role. The plans in place at plants and for their surrounding population have not had the chance to be implemented in a real live event. However, from events like the Turkey Point experience with Hurricane Andrew (with advance warning), it can be noted that there are still some loopholes in existing plans such as communication and transportation after a major event. There has to be more cooperation amongst the different levels of government and the NPPs should be allowed a more active role in the policy and plan development for the safety of the public in their vicinity. Considering the scenario of seismically induced nuclear accident, much of the plans in place would not function as planned if this incident occurred. Even though it is a low probability event, it has the potential for high consequences. Therefore, it should not be neglected. Any changes made to existing policies to counteract possible adverse effect of having this event only help in strengthening the policies in place already and would assure an improved safety for the public. One of the principle of emergency plans listed previously was that policies should be based on scenarios that are as practical as possible; radiological emergency response agencies should, in their policy development process, consider possible consequences resulting from direct seismic affecting protective actions and formulate a revised plan for radiological emergency response actions, if possible. The risk studies of NPPs provide insights into how earthquakes would affect a plant and analyze what incremental risk can be expected from the plants and how to mitigate them effectively.

The conclusion of Phase One of this project elaborates the need for further study in the combined areas of seismic and nuclear risk analysis. There are still uncertainties in the models, data and results obtained. There is also a difference in the way results are presented from each section. NPPs usually present their risks results from Level 1 PRAs, in the form of CDFs and LERFs, these days. However, there is a need for more and current Level 3 PRAs to be conducted in the use for public safety analysis. The level 3 PRA results are easier to understand and communicate to the public. This makes it easier for the public to feel assured that NPPs do not pose unnecessary risks to them.

References

[1] USNRC, *Reactor Safety Study,* WASH-1400 (NUREG 75/014), Washington, D.C., October 1975,

[2] National Academy of Sciences, *Estimation Losses from Future Earthquakes (Panel report/Technical Background*, FEMA-176/177 (Earthquake Hazards Reduction Series), Washington DC, 1989.

[3] National Institute of Building Sciences, *Assessment of the State-of-the Art Earthquake Loss Estimation Methodologies,* FEMA-249, Washington, D.C., June 1994.

[4] USNRC, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, NUREG-1150, Vol. 1 & 2, Washington, D.C., June 1989.

[5] Hickmann, J. et al., *PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants*, NUREG/CR-2300, ANS and IEEE, Vol. 2, Rev. 1, Washington, D.C., April 5, 1982.

[6] IAEA, *Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Levels 1-3)*, Safety Series, Bernan Associates, Vienna, 1992, 1995, 1996.

[7] Fullwood, R.R., *Probabilistic Safety Assessment in the Chemical and Nuclear Industries*, Butterworth-Heinemann Press, NY, 1999.

[8] Parry, G.W., "The Characterization of Uncertainty in Probabilistic Risk Assessments of Complex Systems," Reliability Engineering and Safety Systems, 54 (1996), 119-126.

[9] Apostolakis, G.E., "A Commentary on Model Uncertainty," In Proc. Workshop I in Advanced Topics in Risk and Reliability Analysis, Model Uncertainty: Its Characterization and Quantification, NUREG/CP-0138, Washington, D.C., October 1994.

[10] Budnitz, R.J., "Current Status of Methodologies for Seismic Probabilistic Safety Analysis," Reliability Engineering & System Safety, Volume 62, Issues 1-2, 1998, 71-88.

[11] Federal Emergency Management Agency, *Earthquake Loss Estimation Methodology HAZUS99 Technical Manual*, FEMA, Washington, D.C., 1999.

[12] Kim, J.J. et al, *Analysis of Direct Seismic Risks and their Effects on Existing Emergency Response Plans Near a Nuclear Power Plant,* Nuclear Systems Enhanced Performance Program, MIT-NSP-TR-005, MIT, Cambridge, MA, October 2001,.

[13] Pickard, Lowe and Garrick, Inc., *Seabrook Station Risk Management and Emergency Planning Study*, prepared for Public Service Company of New Hampshire and Yankee Atomic Electric Company, PLG-0432, Seabrook, NH, December 1985.

[14] Pickard, Lowe and Garrick, Inc., *Seabrook Station Probabilistic Safety Assessment (SSPSA),* prepared for Public Service Company of New Hampshire and Yankee Atomic Electric Company, PLG-0300, Seabrook, NH, December 1983.

[15] Kiper, K.L., et al, *Individual Plant Examination – Report for Seabrook Station*, (IPE), Seabrook, NH, March 1991.

[16] North Atlantic Service Corporation, *Seabrook Station Probabilistic Safety Study (SSPSS-1999)*, Seabrook, NH, May 2000.

[17] USNRC, *Goals for the Operations of Nuclear Power Plants: Policy Statement*, 10 CFR Part 50, 51 FR 30028, Washington, D.C., August 1986.

[18] Litai, D., *A Risk Comparison Methodology for the Assessment of Acceptable Risk*, MIT Ph.D. Thesis, MIT, Cambridge, MA, 1980.

[19] National Safety Council, *Injury Facts*, 2000 Edition, Itasca, IL, 2000.

[20] Rimington, J.D., "Risk and The Regulator: Puzzles and Predicaments", Trans IChemE, Vol 73, Part B, August 1995, 173-181.

[21] USNRC & INPO, *Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20-30, 1992*, NUREG-1474, Washington, D.C., March 1993.

[22] Wenger, D. & Derek. T., *Emergency Preparedness and Response Planning: An Overview of Seismic Policy Related Issues*, Preliminary Paper #121, 1987.

[23] Berke, P & Beatley, T., *Planning for Earthquakes*, John Hopkins University Press, Baltimore, 1992.

[24] Alesch, D.J. & Petak, W.J., *The Politics and Economics of Earthquake Hazard Mitigation*, University of Colorado, Denver, CO, 1986.

[25] United States Geological Survey, National Earthquake Information Center, *The Richter Magnitude Scale*, World Data Center for Seismology, Denver, CO.

[26] United States Geological Survey, National Earthquake Information Center, *The Modified Mercalli Intensity Scale*, World Data Center for Seismology, Denver, CO.

[27] D.W. Whitehead (Editor), *Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1*, prepared for the U.S. NRC, NUREG/CR-6143, SAND93-2440, Vol. 1, Washington, D.C., July 1995.

[28] Consolidated Edison Company of New York, Inc., *Indian Point Probabilistic Safety Study*, Power Authority of the State of New York, , NY, 1982.

[29] Woo. G., *The Mathematics of Natural Catastrophes*, Imperial College Press, London, 1999.

[30] USNRC & EUR, *Probabilistic Accident Consequence Uncertainty Analysis*, Vol. 1 & 2, NUREG/CR 6523, Washington, D.C., June 1997.

[31] Gregory, Murfin, et al, *Evaluation of Severe Accident Risks: Sequoyah, Unit 1,"* NUREG/CR-4551, Vol. 5, Rev. 1, Parts 1 & 2, Washington, D.C., December 1990.

[32] Kennedy, R.P., Cornell, C.A., et al, "Probabilistic Seismic Safety of an Existing Nuclear Power Plant," Nuclear Engineering and Design 59, March 1980, 315-338.

[33] Fleming K., et al, "Technical Issues in the Treatment of Dependence in Seismic Analysis," OECD-NEA Workshop on Seismic Risk, August 1999.

[34] Cheok, M.C., Parry, G.W. & Sherry, R.R., "Use of Importance Measures in Risk-Informed Regulatory Applications," Reliability Engineering and System Safety 60, 213-226, 1998.

[35] Kassawara, R.P., Hardy, G.S., & Tong, W.M., "Seismic IPEEE Industry Insights," Nuclear Engineering and Design, 192, 1999, 179-187.

[36] Hirschberg, S., Spiekerman, G., & Dones, R., *Severe Accidents in the Energy Sector*, Project GaBE, Paul Scherrer Institut, Villigen, Switzerland, November 1998.

[37] USNRC, *Plans for Developing a Safety Goal*, NUREG-0735, Washington, D.C., October 1980.

[38] USNRC, *Towards a Safety Goal: Discussion of Preliminary Policy Considerations*, NUREG-0764, Washington, D.C., March 1981.

[39] USNRC, *Workshop on a Proposed Safety Goal*, NUREG/CP-0020, Washington, D.C., September 1981.

[40] USNRC, *Safety Goals for the Operation on Nuclear Power Plants*, 10CFR Part 50, Washington, D.C., August 1986.

[41] USNRC-ACRS, *An Approach to Quantitative Safety Goals for Nuclear Power Plants*, NUREG-0739, Washington, D.C., October 1980.

[42] Cohen, B.L., *The Nuclear Energy Option: An Alternative for the 90s*, Plenum Press, New York, 1990.

[43] Wu, J.S. & Apostolakis, G.E., "Experience with Probabilistic Risk Assessment in the Nuclear Power Industry," Journal of Hazardous Materials, 29 (1992) 313-345.

[44] Higson, D.J., "Nuclear Reactor Safety Goals and Assessment Principles," Nuclear Safety, Vol. 26, No.1, Jan-Feb 1995, 1-13.

[45] USNRC, *Criteria for preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, NUREG-0654, Washington, D.C., Feb 2001.

Appendix A: Summary Of Results Of Workshop On Safety Principles Governing Seismic Risks Of Nuclear Power Plants

Held at

Massachusetts Institute of Technology

Cambridge, MA 02139 USA

10 September 2001

Chairman: Prof. Michael W. Golay

Principal Investigator: TEPCO/MIT Project on External Event Nuclear Power Plant Risks

Participants List:

Name	Company	Address	Tel/Fax	e-mail
Michael Golay	MIT	77 Mass Ave, 24-223, Cambridge, MA 02139	617-253-5824 /617-258-8863	golay@mit.edu
George Apostolakis	MIT	78 Mass Ave, 24-221, Cambridge, MA 02139	617-252-1570	apostola@mit.edu
Robert Budnitz	FRA Inc.	2039 Shattuck Ave, #402, Berkeley, CA 94704	510-644-2700/ 510-644-1117	budnitz@pacbell.net
Gabriel R. Toro	Risk Engineering	3 Farmers Row, Acton, MA 01720	978-266-2639	toro@riskeng.com
Kenneth C. Rogers		6202 Perthsire CT., Bethseda, MD 20817		kcrogers@auc.com
Robin K. McGuire	Risk Engineering	4155 Darley Ave, #A, Boulder, CO 80305	303-499-3000	mcguire@riskeng.com
Forrest J. Remick		305 E. Hamilton Ave, State College, PA 16801-5413	814-237-1038	FJR1@psu.edu
Thomas S. Kress	ACRS	102B Newridge Rd, Oak Ridge, TN 37830	865-483-7548/ 865-482-7548	tskress@aol.com
Koichi Miyata	Toden Software Inc.	Tokyo Bijyutsu club Bldg., 6-19-15 Shimbashi, Minato-ku, Tokyo 105-004, Japan	(011)81-3-4586-6782 / (011)81-3-4586-7670	tiyata@tsi.co.jp
Fukashi Watanabe	TEPCO	Vchisaiwai-choi-1-3, chiyodu-ku, Tokyo, Japan	(011)81-3-4216-3879	watanabe.fukashi@tepco.co.jp
Mingyang Xu	MIT	NW12-230, MIT, Cambridge	617-253-6613	xumy@mit.edu
Yingli Zhu	MIT	102 green hall, Cambridge	617-225-6113	ylzhu@mit.edu
Gaku Sato	MIT/TEPCO	NW12-211, MIT, Cambridge	617-258-7430	gaque@mit.edu
Tina Ghosh (Rapporteur)	MIT	77 Mass Ave, 24-220C, Cambridge, MA 02139	617-253-5041	tinag@mit.edu
Bukola Afolayan Jejeloye (Rapporteur)	MIT	78 Mass Ave, 24-218C, Cambridge, MA 02139	617-253-2831	rogue@mit.edu

OVERVIEW

The project on External Event Nuclear Power Plant Risks at nuclear power plants has been underway for approximately 18 months and is sponsored by the Tokyo Electric Power Corporation (TEPCO). The project team consists of professors (Michael Golay and George Apostolakis) and students (Bukola Afolayan, Jonathan Kim, Mingyang Xu and Yingli Zhu) from MIT and seismic risk experts from Risk Engineering Associates (Robin McGuire and Gabriel Toro). The project's purpose is to develop safety principles that could be used in the regulation of severe external event-associated risks arising at nuclear power plants. To do this, the project team has been evaluating nuclear power plant and direct seismic risks at an example United States nuclear power plant. It has also been evaluating other forms of nuclear and public risks, which could be used to provide a context in terms of which to view those of the nuclear power plant.

The one-day workshop summarized here was organized by the project. It was used to obtain the views of senior nuclear safety experts concerning the merits and weaknesses of different forms of safety principles developed in the project, and to get their ideas concerning how the overall topic of nuclear seismic risk should be treated. The purpose of the Workshop on Safety Principles Governing Seismic Risks of nuclear power plants was to obtain the views of the invited experts, but not necessarily to seek a consensus among them.

Summary of Project Results

Prof. Michael Golay – Introduction, Project Overview, Alternative Safety Principles, and Summary of the More Important Results

Professor Golay presented the project results developed to-date. Most of this material had been provided to the workshop participants in advance in order to help them focus upon the topics of greatest interest, and to provide a means by which the participants could evaluate the implications of the different safety principles examined in the project.

The main points from the presentation included showing that the conditional mean fatalities due to seismic events at nuclear power plants are considerably smaller than the corresponding direct fatalities from seismic events. These ratios depend, somewhat, upon the areas over which the direct seismic risks are evaluated. A mild consensus was evident that comparison of total direct seismic risks to total nuclear seismic risks is reasonable, but that this comparison should be restricted to earthquakes strong enough to damage the nuclear power plant (stronger than magnitude 5, which can affect regions about of 10km). On a national basis nuclear seismic risks are even smaller in comparison to direct seismic risks (which are small compared to other accidental risks, but which occur with sufficient frequency to permit actuarial treatments of them). There is also a finite radionuclide inventory in the reactor's core; therefore, for earthquakes stronger than a certain magnitude (about magnitude 7), the mean conditional nuclear seismic fatalities reach an asymptote as much of this inventory becomes released. However, this is not the case for direct seismic consequences, which continue to grow steadily as the earthquake magnitude increases (with both the size of the affected region and the density of damage increasing).

In the discussion of the fraction of nuclear risks contributed by earthquakes it is seen in the NRC's Independent Plant External Event Examinations (IPEEE) that seismic core damage risks (CDFs) due to earthquakes typically contribute a large share of the total. At the plant of the project's study, the value is about 25%, and at some other plants it is greater.

Basis For Comparison

TEPCO's representative presented ideas under consideration at TEPCO for possible treatment of nuclear seismic risks. The main points discussed concerned the following questions:

1. What goals should be set for external event risk?
2. Should external safety goals be set separately from entire safety goal?
3. Should all seismically induced CDF of individual units be added up to obtain the total CDF for multi-unit site?
4. Does the required level of the goal depend on the level of uncertainty?
5. Should the mean or median of a distribution be used in seismic PSA for judgment?

The participants addressed possible answers to these questions in the discussions.

Seismic risks are disparate, arising in such forms as prompt and latent fatalities, injuries, and property and income losses. The participants brought up the possibility of converting the units of risks for purposes of obtaining an overall scalar measure of the vector of risks. While the project results showed two ways of weighting the early and latent fatalities, some participants (Tom Kress and Robert Budnitz) suggested using the dollar as the standard unit, i.e., putting a cost on every damage. No consensus emerged upon either the value of making such a transformation or of a straightforward way of formulating one. It was also pointed out how latent fatality results, with uncertainties, could be added to the fatalities from nuclear power plants in a variety of ways. The expected conditional mean latent fatalities are of about the same magnitude as the direct fatalities, when measured either by the numbers of individuals involved or in terms of expected life shortening, but are considerably smaller when measured in terms of social valuation of latent vs. prompt fatalities. Joe Murphy noted (in absentia, by email) that the latent fatality might be overestimated because of the way in which consequence codes model factors like wind direction.

Other forms of seismic consequences discussed were comparative property losses, where the conditional nuclear losses are typically smaller than the direct losses by ratios greater than those obtained for fatalities and injuries. However as noted by participants, the methodology and estimates used for nuclear property losses are deficient. It was because of this reason that estimated losses from tourism and potential loss of values of houses were added to the nuclear losses by the project team.

There was a consensus that seismically induced nuclear risks should not be compared to direct seismic risks. Some reasons given (particularly by Tom Kress) include the difficulty of comparing seismic "benefit" with seismic cost, whereas the acceptable societal risk from nuclear power does depend on a cost-benefit analysis; one can compare the overall benefit of nuclear energy to the overall cost of nuclear energy, and the seismic-induced risk should be wrapped into the calculation of the overall cost of nuclear energy. There is no analogous cost-benefit analysis for other direct seismic risks. While the direct risks are totally naturally based, the nuclear risks are technology based. It was agreed that there should not be separate seismic risk guidelines, but rather the seismic risk should be seen as just another contributor to overall risk from nuclear energy. Therefore, the participants did not accept the overall proposal of basing acceptable risks upon a comparison of nuclear seismic risks to overall risks from seismic events. Rather, there was consensus among the invited participants that it is sufficient for overall nuclear risks to be kept small relative to all accidental and cancer risks, as is currently required by the United States Nuclear Safety Goals. Many of the participants agreed with the reasoning that nuclear risks rise along with benefits produced by nuclear energy while there were no similar benefits to be derived in exchange for suffering direct seismic event risks. However, a contrary view (by Mike Golay) was that populations are exposed to the direct seismic risks in exchange for the benefits that they derive from living in seismically active places; and thus, that nuclear and direct seismically related benefits can be contrasted, should that be useful.

George Apostolakis suggested a comparison of nuclear risks to other sources of energy risks; however this would require use of a standardized accepted method for risk analysis from all sources, which does not exist currently.

Safety Goal Proposals and Acceptability Criteria

The discussion started with looking for means to decide acceptability criteria for regulators that is practical. Among the seismic Safety Goal Proposals presented by Mike Golay, from the project results, were the following:

- Requiring that nuclear power plants risks be limited according to a criterion that

[nuclear power plant seismic risks (a)/other seismic risks (a)] < L,

where all risks are evaluated on a consistent basis; "a" indicates the same confidence level being used in the evaluation of the risks being compared and L is a value selected by the regulatory authority. The participants agreed that the use of a confidence level for risk evaluation is justified, particularly for risks that are considered to be highly uncertain.

- Requiring that nuclear power plants undertake preparations of high benefit to cost ratios to mitigate seismic nuclear risks. In an examination of possible preparations for mitigation of risk (e.g., prior distribution of KI pills), most were found to be much more expensive than the highest implicit valuations of human life made in the United States (being about $100 million spent to avert the mean fatalities of astronauts and national leaders). One class of such actions deserving further investigation is preparing the nuclear power plant staff to play a greater role in helping those affected by nuclear releases. This could be done by allowing them to take on a broader emergency preparedness role, like broadcasting information and advice directly to the locally affected population, instead of relying on government authorities who would likely be over-worked and concerned about the other effects of the overall disaster.

- Koichi Miyata and Watanabe discussed setting a S3 earthquake magnitude point (the upper limit of seismic safety margin), which will be greater than that of the SSE of a nuclear power plant. The idea is to use the S3 earthquake as a means of deterministically showing that the plant can withstand earthquakes of greater magnitude than the SSE. Once S3 value is known, then deterministic engineering rules can be used for design and regulatory evaluation. It is expected that use of the S3 can help with compliance. However, there is the problem of how to determine the S3 magnitude.

The participants agreed that this is a difficult question to answer, and there were few suggestions on how to choose the S3 magnitude. Robin McGuire suggested that one way to set S3 magnitude is to find an acceptable CDF value, and then to work backward to determine in a best estimate sense, or at a stated confidence level the frequency of an earthquake of magnitude likely to cause core damage. But George Apostolakis and others agreed that this is not a simple task because of the difficulty of accommodating disagreements among experts who might be used for formulation of the needed seismic hazard input information. One would have to use the hazard curve and thereby use probabilistic analysis. No consensus emerged concerning the proper formulation or use of the S3 concept.

George Apostolakis suggested adopting a version similar to the British and Dutch approach of using three-regions approach in the risk space (see Figure 5.9). Risks below a given level are broadly acceptable; risks above a given level are broadly unacceptable; acceptability of everything in the middle band between these values is determined through cost-benefit analysis (i.e., one would balance proposed increment of risk reduction against its cost), and to require greater conservatism and redundancy as one approaches the unacceptable boundary. In the Japanese context, the Japanese regulator would have to define the quantitative limits of these risk regions, and perhaps provide guidance concerning appropriate cost-benefit trade-offs. Forrest Remick pointed out that the United States regulatory approach implicitly is consistent within the British approach. All participants endorsed this idea.

Questions were raised regarding what metric should be used for determination of the boundaries of the different risk regions. Some people suggested use of CDF since this is used in the United States as a supplemental Safety Goal. It was also noted that the Japanese nuclear power plants also have not performed a Level 3 PRA yet. However, George Apostolakis said using individual risks from a Level 3 PRA would be more appropriate for measuring the risks involved. He also argued that this would also make it easier for the public to understand and accept the results.

In response to the question of how this proposal could be used to set goals for the risks from seismic events, it was mentioned that the risks from a nuclear power plant can be categorized in terms of the initiating events. Then it can be seen, in the 3-region risk acceptance approach, which initiating event contributes most to the total risk. From this, one can attempt a risk reduction, should it be justified in cost-benefit terms, unless the nuclear power plant is in the unacceptable region.

For multi-unit plants, it was observed that the overall risks of the site and not those of the individual units should be used in this risk acceptance approach. If CDF were to be used as an acceptance criterion, could the CDFs from each unit just be added up to get an overall CDF for the site? Can the mean risks from each individual unit be added to provide the overall site mean risk? There was consensus that the risks due to independent failures are additive. It may be difficult to meet risk acceptance standards in this way as adding the mean risks may put a site into the "unacceptable" region, or at least outside of the "broadly acceptable" region.

Conclusion

The original proposal, in answer to the question of whether there should be a separate event specific (seismic) goal, was not accepted by the participants. Many favored using uniform and common total risk acceptance criteria for the entire nuclear power industry, but not to use event specific criteria.

With regard to the use of mean results to express the results from a risk analysis and as the basis for judging acceptable performance, it was noted that the mean result is used in the U.S.. The participants agreed that when evaluating the risks from a nuclear power plant, the distribution including mean and median values and their confidence levels should be used. Should such an approach be used in Japan, the national regulators would decide where to set confidence level for the Japanese nuclear power plants.

There were other suggestions which did not advance far like comparing the seismically induces effect of alternate sources of energy, of allocating risks to various event sequences based on whether they have high or low uncertainties, and of allocating resources to consequence prevention (e.g., building more hospitals) rather than accident mitigation (e.g., backfitting nuclear power plants).