NUREG/CR-5042
UCID-21223

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

C. Y. Kimura, R. J. Budnitz*

Prepared for
U.S. Nuclear Regulatory Commission

* Future Resources Associates, Inc.
2000 Center St., Suite 418
Berkeley, CA 94704

Lawrence
Livermore
National
Laboratory

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Information Support Services, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG/CR-5042
UCID-21223

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

Manuscript Completed: October 1987
Date Published: December 1987

Prepared by
C. Y. Kimura, R. J. Budnitz*

Lawrence Livermore National Laboratory
7000 East Avenue
Livermore, CA 94550

Prepared for
Division of Reactor and Plant Systems
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN No. A0448 Task 8
NRC FIN No. A0815 Task 1

*** Future Resources Associates, Inc.**
2000 Center St., Suite 418
Berkeley, CA 94704

Abstract

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

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

Four external hazards were investigated in this report. These external hazards are internal fires, high winds/tornadoes, external floods, and transportation accidents. Analysis was based on two figures-of-merit, one based on core damage frequency and the other based on the frequency of large radioactive releases.

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

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES (continued)

LIST OF TABLES (continued)

CHAPTER 1
INTRODUCTION

1.1 BACKGROUND

The Nuclear Regulatory Commission (NRC) has recently issued its Policy Statement on Severe Accidents [1.1]. This Policy Statement sets the goals and schedule for addressing issues relevant to severe accidents in the licensing of future plants and for the systematic examination of existing plants.

The NRC Severe Accident Policy Statement does not differentiate between events initiating within the power plant and events caused by external initiators, such as earthquakes, floods, and high winds. The evaluation of internally initiated events is more developed than the evaluation of externally initiated events. Therefore the NRC staff is implementing the Severe Accident Policy Statement for internally initiated events. The implementation plan is in a recent NRC internal paper, SECY 86-76 [Ref 1.3].

The evaluation of severe accidents initiated by external events will proceed in two phases [1.4]. The first phase will be an assessment of the margin provided by past and ongoing programs relative to external events beyond the design basis. In addition, this phase includes identification of areas where examination for external vulnerabilities is needed. The second phase will consist of a program for plant specific evaluation, if needed. Information developed in phase one will be used as guidance for phase two.

As part of the research program supporting the implementation of the NRC Policy Statement on Severe Accidents [1.1], the Lawrence Livermore National Laboratory (LLNL) has performed a study of the risk of core damage to nuclear power plants in the United States due to externally initiated events. The broad objective has been to gain an understanding, for existing U.S. light water reactor (LWR) power plants, of whether or not each external initiator is among the major potential accident initiators that may pose a threat of severe reactor core damage or of a large radioactive release to the environment from the reactor.

1.2 WORK REQUIREMENTS

The following work requirements were issued by the NRC staff as specific objectives in implementing this study:

1. Establish the extent to which plants have been reviewed for external event vulnerabilities in the past.

2. Estimate the degree of protection provided by those reviews with regard to external events beyond the design bases.

3. Specify external events that should be included in the vulnerability

search.

4. Identify areas in which the existing requirements and review have not ensured sufficient protection for the external events specified for requirements 1 through 3.

5. Propose an approach consistent with the NRC's Severe Accident Policy by which utilities may evaluate their own individual plants for severe accidents initiated by external events.

Requirements 1 and 3 are covered in Chapter 2 of this report. The external events identified in Chapter 2 of this report as being of sufficient importance to warrant further investigation are covered in individual chapters following Chapter 2. Requirements 2, 4 and 5 are included for each important external event following Chapter 2.

1.3 EVALUATION CRITERIA

Nuclear power plant accidents initiated by external events, or external initiators, are defined for the purpose of this study to be events outside of the power plant itself, which lead to accident scenarios having a finite probability of resulting in damage to the reactor core or release of radioactive material from the reactor core. Although fires within the plant fall outside of this definition, they are included in this report, but all other events internal to the plant are excluded. Internal events excluded from consideration are all Loss-of-Coolant Accidents (LOCAs), all anticipated operational transients, main turbine-generator missiles accidents, internal flooding, deliberate sabotage, etc.

In order to examine the risk to U.S. nuclear power plants from any specific externally initiated event, it has been necessary to employ specific evaluation criteria to discriminate between the significant and the less significant levels of risk. These evaluation criteria have used the guidance provided by the NRC in their Safety Goal Policy Statement [1.2] and Policy Statement on Severe Accidents [1.1]. Two different figures-of-merit have been used as evaluation criteria.

These two figures-of-merit are being used solely for the purpose of screening the relatively more important external initiators from the relatively unimportant ones. There is no implication here that individual plants that are currently licensed to operate or authorized for construction must meet these figures-of-merit.

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

> "providing reasonable assurance, giving consideration to the uncertainties involved, that a core-damage accident will not occur at a U.S. nuclear power plant." [1.2]

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

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

> "Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation." [1.2]

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

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

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

the large uncertainties in PRA results.

1.4 REPORT ORGANIZATION

This report has been organized into seven chapters. Chapter 2 identifies the externally initiated events that are considered in more detail in subsequent chapters of this report. Chapter 3 discusses internal Fires or Fires within the nuclear power plant controlled site boundary. Chapter 4 considers all natural phenomena including hurricanes and tornadoes, that could result in high wind velocities at the nuclear power plant site. Chapter 5 discusses floods external to the nuclear power plant itself caused by heavy precipitation, high water, dam failures or the combination of these. Chapter 6 discusses transportation accidents, i.e. aircraft crashes, ship/barge, railroad, truck accidents, which can affect the power plant by actual physical damage to the plant or by releasing hazardous materials near the plant site. Chapter 6 also considers gas/oil pipeline accidents near the power plant site. Chapter 7 presents a summary of important findings and conclusions.

1.5 REFERENCES

[1.1] "Policy Statement on Severe Reactor Accidents", U.S. Nuclear Regulatory Commission, Federal Register,' Vol. 50, pg. 32138. August 8, 1985.

[1.2] "Policy Statement on Safety Goals", U.S. Nuclear Regulatory Commission, Federal Register, Vol. 51, pg. 30028. August 21, 1986.

[1.3] U.S. NRC Policy Paper SECY 86-76, "Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information", memorandum from V. Stello. February 28, 1986.

[1.4] U.S. NRC Policy Paper SECY 86-162, "Treatment of External Events in the Implementation of the Severe Accident Policy Statement", memorandum from V. Stello. May 22, 1986.

CHAPTER 2
IDENTIFICATION OF EXTERNALLY INITIATED EVENTS

2.1 INTRODUCTION

For the purposes of this study, "externally initiated events" can be broadly grouped into the following categories:

1. Internal Fires,
2. High Winds/Tornadoes,
3. External Floods,
4. Transportation Accidents, and
5. Others

Internal Fires are considered to include all fires within the power plant controlled site boundary. High winds/tornadoes are considered to include all natural phenomena including hurricanes, that result in high winds at the power plant site. External floods are considered to include all floods resulting from high water, heavy precipatation, dam failures, and combinations of high rains and dam failures. Transportation accidents are considered to include aircraft crashes on the power plant site, ship/barge collisions with power plant structures and ship/barge accidents near the power plant site which release hazardous materials, truck accidents near the power plant site which release hazardous materials, railroad accidents near the power plant site which release hazardous materials, and gas/oil/chemical pipeline accidents near or on the power plant site which release hazardous materials. The Others category includes all externally initiated events that are not covered in the first five categories. A partial list of such events, not all of which are natural events, includes:

a) Extreme Heat,
b) Extreme Cold,
c) Ice,
d) Hail,
e) Snowstorms,
f) Dust storms, sandstorms,
g) Lightning Strikes,
h) External Fires (i.e. forest fires, grass fires),
i) Extraterrestrial Activity (i.e. Meteorite Strikes, Satellites),
j) Volcanic Activity,
k) Damage or Destruction due to Military Action,
l) Avalanche, landslide,
m) Release of Hazardous Materials from On-site Storage,
n) Accidents from Nearby Industrial, or Military Facilities, and
o) Industrial Sabotage

Earthquakes are also considered an external hazard to a nuclear power plant but will not covered in this report. Earthquakes will be covered in a separate report to be published on a later schedule.

2.2 NRC REGULATORY REQUIREMENTS

When the U.S. Atomic Energy Commission (AEC now the NRC) regulatory staff first began to review nuclear power plant designs, its scope of review was less defined than it is presently. The requirements for acceptability evolved as new facilities were reviewed. In 1971, the General Design Criteria for Nuclear Power Plants or GDCs were formally adopted as the minimum requirements for the principal design standards and have been used as guidance in reviewing new plant applications since then. Safety guides issued in 1970 became part of the Regulatory Guide Series in 1972. These guides describe methods acceptable to the AEC (now NRC) regulatory staff for implementing specific portions of the regulations, including certain GDC, and formalize staff techniques for performing a facility review. In 1972, the AEC released the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, now known as Regulatory Guide 1.70. It provided a standard format for these reports and identified the principal information needed by the staff for its review. The Standard Review Plan (SRP, NUREG-75/087) was published in December 1975 and updated in July 1981 (NUREG-0800) to provide further guidance for improving the quality and uniformity of staff reviews, to enhance communication and understanding of the review process by interested members of the public and nuclear power industry, and to stabilize the licensing process.

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

2.2.1 Title 10 of the Code of Federal Regulations (10 CFR)

Title 10 of the Code of Federal Regulations (10 CFR) [2.1] specifies in general terms, the conditions and factors that must be considered in constructing, licensing, and operating a nuclear power plant and the regulatory process that must be followed in performing this function. The specific parts of 10 CFR that consider external hazards as defined for this report are listed below according to the external event categories given above:

1) Internal Fires
10 CFR Part 50.48 Fire Protection,
10 CFR Part 50 Appendix A General Design Criteria for Nuclear Power Plants, Criterion 3-Fire Protection,
10 CFR Part 50 Appendix R Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979;

2) High Winds/Tornadoes
10 CFR Part 50 Appendix A General Design Criteria for Nuclear Power Plants, Criterion 2-Design Bases for Protection against Natural Phenomena,
10 CFR Part 50 Appendix A General Design Criteria for Nuclear Power Plants, Criterion 4-Environmental and Missile Design Bases,
10 CFR Part 100.10 Factors to Be Considered When Evaluating Sites;

3) External Floods
10 CFR Part 50 Appendix A General Design Criteria for Nuclear Power Plants, Criterion 2-Design Bases for Protection against Natural Phenomena,
10 CFR Part 100 Reactor Site Criteria,
10 CFR Part 100 Appendix A Seismic and Geologic Siting Criteria for Nuclear Power Plants;

4) Transportation Accidents
10 CFR Part 50.34 Contents of Applications; Technical Information,
10 CFR Part 100 Reactor Site Criteria,
10 CFR Part 100.10 Factors to Be Considered When Evaluating Sites;

5) Others
10 CFR Part 50.55a Codes and Standards,
10 CFR Part 50 Appendix A General Design Criteria for Nuclear Power Plants, Criterion 2-Design Bases for Protection against Natural Phenomena,
10 CFR Part 50 Appendix A General Design Criteria for Nuclear Power Palts, Criterion 44-Cooling Water,
10 CFR Part 100 Reactor Site Criteria,
10 CFR Part 100 Appendix A Seismic and Geologic Siting Criteria for Nuclear Power Plants;

2.2.2 NRC Regulatory Guides

The specific guidance used by the NRC staff to review nuclear power plants for protection against the external hazards as defined in this report is given by documents known as NRC Regulatory Guides. The specific NRC Regulatory Guides are listed below according to the external event categories defined in subsection 2.1:

1) Internal Fires

Regulatory Guide 1.120 Fire Protection Guidelines for Nuclear Power Plants;

2) High Winds/Tornadoes

Regulatory Guide 1.13 Fuel Storage Facility Design Basis,

Regulatory Guide 1.76 Design Basis Tornado for Nuclear Power Plants,

Regulatory Guide 1.91 Evaluations of Explosions Postulated to Occur on

Transportation Routes Near Nuclear Power Plant Sites,

Regulatory Guide 1.117 Tornado Design Classification, Structures, Systems & Components to be Protected Against Tornadoes;

3) External Floods

Regulatory Guide 1.27 Ultimate Heat Sink,

Regulatory Guide 1.59 Design Basis Floods for Nuclear Power Plants,

Regulatory Guide 1.102 Flood Protection for Nuclear Power Plants;

4) Transportation Accidents

Regulatory Guide 1.78 Assumptions for Evaluating Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,

Regulatory Guide 1.91 Evaluations of Explosions Postulated to Occur on Transporation Routes Near Nuclear Power Plants,

Regulatory Guide 1.95 Protection on Nuclear Power Plant Plant Control Room Operators Against an Accidental Chlorine Release;

5) Others

Regulatory Guide 1.17 Protection of Nuclear Power Plants Against Industrial Sabotage.

2.2.3 NRC Standard Review Plan

The NRC Standard Review Plan (SRP) [2.2] serves as the guide for the NRC staff in their review of the Preliminary Safety Analysis Reports (PSARs) and Final Safety Analysis Reports (FSARs) submitted by the applicant. The SRP calls for the following externally initiated events to be considered when reviewing safety analysis reports (SARs) for nuclear powers plants (listed according to the external event categories given above):

1) Internal Fires
SRP No. 9.5.1 Fire Protection Program;

2) High Winds/Tornadoes
SRP No. 3.3.1 Wind Loadings,
SRP No. 3.3.2 Tornado Loadings,
SRP No. 3.5.1.4 Missiles Generated by Natural Phenomena,
SRP No. 3.5.1.5 Site Proximity Missiles (Except Aircraft),
SRP No. 3.5.2 Structures, Systems, and Components to Be Protected from Externally Generated Missiles,
SRP No. 3.5.3 Barrier Design Procedures;

3) External Floods
SRP No. 2.4.2 Floods,
SRP No. 2.4.3 Probable Maximum Flood (PMF) on Streams and Floods,
SRP No. 2.4.4 Potential Dam Failures,
SRP No. 2.4.10 Flood Protection Requirements,
SRP No. 3.4.1 Flood Protection;

4) Transportation-Related Accidents
SRP No. 2.2.1-2.2.2 Identification of Potential Hazards in Site Vicinity,
SRP No. 2.2.3 Evaluation of Potential Accidents,
SRP No. 3.5.1.6 Aircraft Hazards;

5) Others
SRP No. 2.4.5 Probable Maximum Surge and Seiche Flooding,
SRP No. 2.4.6 Probable Maximum Tsunami Flooding,
SRP No. 2.4.7 Ice Effects,
SRP No. 2.4.8 Cooling Water Canals and Reservoirs,
SRP No. 2.4.9 Channel Diversions;

2.3 MATERIAL REVIEWED FOR THIS STUDY

Each externally initiated event category except for the "Others" category is further studied in a separate chapter of this report following broadly the same organization. The technical material that has been reviewed in this project consists of the following:

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

2) Technical papers and reports, including discussion of the PRA methodology for the initiator, technical studies on the associated phenomena, and studies of the traditional engineering approaches to assuring protection against the external initiator;

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

4) PRA literature on the external initiator, including both full-scope PRA studies and partial PRA type analyses;

5) Event data base at nuclear power plants, including not only the larger events of interest, but smaller events that did not cause serious problems but comprise aspects of the overall data base;

6) NRC's current research program, where applicable.

Not all of this material has been studied in detail, although important aspects of each category above have been examined.

2.4 REFERENCES

[2.1] Title 10 Code of Federal Regulations, Energy (10 CFR) Parts 0 to 199, Revised as of January 1, 1987, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

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

CHAPTER 3---INTERNAL FIRES

3.1 OBJECTIVES

The work reported here is a review and evaluation of what is known about the risk of core-damage accidents and of the potential for large radiological releases due to internal fires in U.S. nuclear power plants. This review and evaluation have been done to accomplish the tasks discussed in Chapter 1.

To summarize, the broad objective is to understand, for U.S. light-water power reactors, whether or not internal fires are among the major accident initiators that may pose a threat of severe core damage or of a large radioactive release. Due to limited resources in this project, this broad objective cannot be addressed for all U.S. plants, so the project scope has been limited to examining a few specific plants whose potential for fire-initiated accidents has been studied in greater detail using PRA methods.

The evaluation criteria, as outlined in Chapter 1, involve two different figures-of-merit (FOM): (1) The first FOM is met if individual plants have mean core-damage frequencies in the range of about one part in 100,000 per reactor year. This is not a firm numerical objective, but a range. (2) The second FOM is met if a "large release" is calculated to occur with a mean frequency less than 1 in 1,000,000 per reactor year. Both of these FOMs are discussed in more detail in Chapter 1.

3.2 TECHNICAL MATERIAL REVIEWED

The technical material that has been reviewed in this chapter consists of the following:

1) the NRC's regulatory approach to assuring that reactors are adequately protected against fires (the General Design Criteria, the Standard Review Plan, 10 CFR 50 Appendix R, Branch Technical Position BTP 9.5-1, various regulatory guides);

2) papers and reports concerning internal fires, including discussions of the PRA methodology for fires, technical studies on fire phenomena, and methods for modeling fire initiation, growth, suppression, and damage;

3) documentation about how reactors are designed and built by their owners and reviewed by NRC for fire safety (selected SARs, SERs);

4) the PRA literature on fires, including eight full-scope PRAs that have studied fires;

5) the data base on fires at nuclear power plants, including not only major fires but also smaller fires that did not cause serious problems but that comprise aspects of the fire data base;

6) NRC's current research program on fires.

Not all of this vast body of material has been studied in detail, although important aspects of each category above have been examined. To achieve the objectives, the project team has concentrated on the PRA literature on fires. This includes not only the primary literature (the PRAs themselves), but also the literature on PRA methodology, and a small amount of secondary literature in which the PRA results and insights have been analyzed and summarized.

3.3 EVALUATION OF MATERIAL REVIEWED

3.3.1 Current Regulatory Requirements

There have been regulatory requirements concerning fire protection since the very earliest days of reactor safety regulation. Unlike the situation for many other safety issues, there have not been a large number of important changes to these regulations. Over the years, there has been one very major regulatory change, which occurred in 1980 with the adoption of 10 CFR 50, Appendix R ("Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979"). Although this title directly addresses the older operating plants, the technical requirements of Appendix R have been implemented in all newer plants as well.

For the purposes of this analysis, it has been assumed that the Appendix R requirements apply equally well to all nuclear power reactors. This assumption is largely but not entirely valid, in that some plants received exemptions to some of the requirements due to unusual circumstances. However, these exceptions are judged to be of only minor significance, so in effect it is reasonable to state that Appendix R has been effectively implemented at all plants being studied.

With this assumption, the analysis will proceed as if there are no differences among the regulatory requirements that various plants must meet in the area of fire protection.

3.3.2 Studying the "Degree of Protection Achieved"

To study the degree of protection that has been achieved in the fire-protection area, the approach has been to use the PRA literature. This is because the PRA literature on fires represents the best "realistic" set of analyses of the performance of reactors in fires.

In the initial discussion below, the following two key assumptions have been made:

- the methodology used in the fire PRAs is assumed to be valid ---- that is, it provides an accurate determination of the core-damage frequency from fires, within the quoted uncertainty ranges.

- all of the fire PRAs are assumed to have been performed using a common methodology, so that their results can be compared on a common basis.

Of course, these two assumptions are not completely valid, for three reasons:

- the earlier PRAs were not as comprehensive as the more recent studies;

- there are several assumptions in the methodology whose validity is suspect; and

- some of the reactors made changes (backfits) during or after the PRA studies.

Section 3.3.2.3 below discusses these methodological differences among the various PRAs, and what the differences imply regarding the insights that are derived from these PRAs.

3.3.2.1 Core Damage Frequencies

Using the two key assumptions above, it is instructive to present, in tabular form, the "bottom line" results of the fire PRAs that comprise the data base. Table 3.1 shows the mean core-damage frequencies from the various PRAs studied. This tabular compilation is adapted from earlier compilations by Berry [Ref. 3.12], by Wheelis [Ref. 3.20], and by Kazarians, Siu, and Apostolakis [Ref. 3.15]. Please note that although the table presents only the mean values, the analyses themselves contain much more information, in the form of distributions that reflect the analysts' full state of knowledge. Typically, these distributions range over more than one order of magnitude in the core-damage-frequency axis.

Table 3.1

Core Damage Frequencies from Internal Fire PRAs

[Refs. 3.1, 3.3-3.6, 3.8, 3.9, 3.11]

Station	Mean, Core Damage Frequency/Rx-yr for Fires	Mean, Core Damage Frequency/Rx-yr for All Initiators	Fires, Fraction
Zion 1-2	1.8 E-6	5.7 E-5	3 %
Indian Point 2	1.4 E-4	4.7 E-4	30 %
Indian Point 3	9.6 E-5	2.3 E-4	40 %
Big Rock Point	2.3 E-4 *	9.8 E-4	23 %
Limerick	2.3 E-5	4.4 E-5	55 %
Seabrook	2.5 E-5	2.3 E-4	12 %
Oconee-3	1.0 E-5	2.5 E-4	4 %
Millstone-3	4.8 E-6	7 E-5	7 %

* The Big Rock Point analysis produced a "point estimate", not a mean value.

Besides these eight PRAs, the literature on fires includes several other PRA analyses that could have been included here but have not been. The reasons for not using them here are as follows: (1) Two studies are PRA analyses of plants that have not yet been built. These are a study by General Electric Company of its standard-plant GESSAR-II design, and a study by the U.K. Central Electricity Generating Board of their British adaptation of a standardized Westinghouse design, planned for construction at Sizewell-B. Because neither of these plants now exists, the level of detail in the studies is inadequate for our purposes here. (2) Five reactors were subjected to an abbreviated fire PRA under the auspices of NRC's program for resolution of generic safety issue A-45 (decay-heat removal). As part of the A-45 project, Sandia National Laboratories studied fire-related vulnerabilities to the decay-heat-removal function. The five plants studied are St. Lucie-1, Quad Cities, Turkey Point, ANO-1, and Point Beach. However, these analyses, which used some simplifying assumptions, are not full-scope PRAs. In fact, each of the five reports states that the analysis "...is not intended to be a comprehensive fire PRA." Therefore, the insights derivable from them may not be generally compatible with insights derived from full-scope PRAs even though the broad approach used for the fire analyses is a state-of-the-art approach.

The table 3.1 above for the eight plants studied, if taken at face value, reveals two very important insights:

- The mean core-damage frequencies for fire are in a few cases near 10^{-4}/year, and in several cases in the range near or above 10^{-5}/year.

- In many cases, the calculated mean core-damage frequency for fires is a sizable fraction of the total core-damage frequency.

In comparing the calculated values of mean core-damage frequency to figure-of-merit # 1, we observe that for many of the plants the figure of merit is comparable to the calculated core-damage frequency. Of course, the validity of this conclusion depends on the assumption that these fire PRAs are valid, and that their methodology allows a valid comparison on a common basis. This aspect will be discussed separately below (see Section 3.3.2.3).

3.3.2.2 Frequencies of a "Large Release"

For some of the studies, there is information about what fraction of the calculated core-damage frequency is associated with a large release. This information is derived from the "plant damage states" and "release categories" associated with the fire sequences.* As examples, the fire PRA analyses for Zion, Limerick, Indian Point-2 and -3, and Oconee-3 are discussed.

For Indian Point-2 and Indian Point-3, Westinghouse 4-loop PWRs with large dry containments, the fire-initiated sequences are dominated by release categories involving late containment overpressure failure, and release categories involving an intact containment comprise the remaining sequences. According to the PRA results, no sequences involving a potential "early containment failure" or "large release" are initiated by fires [3.3].

For Zion-1 and Zion-2, also Westinghouse 4-loop PWRs with large dry containments, the conclusion is similar: all but 1% of the fire-initiated core-damage frequency is associated with release categories involving an intact containment, and the remaining small contribution involves a late overpressure failure of containment [3.11].

---

* The "plant damage state" and "release category" for a given PRA accident sequence are used to categorize these sequences according to the characteristics of the accident at the time when the core melting occurs (PDS), and at the time when the containment is breached or bypassed, allowing radioactive material to escape to the environment (RC). According to the PRA analyses, most core-damage accidents are not associated with a "large release" of radioactivity. In laymens' terms, the PDS/RC categorization enables the analyst to segregate those that are from those that are not.

For Oconee-3, a Babcock and Wilcox 2-loop PWR with a large, dry containment, essentially all of the core-damage frequency from fire-initiated sequences involves late containment failures or intact-containment accident sequences. A very small contribution (on the order of 10^{-10} per year) is associated with an early large release category, but for this category, the total from all initiators is about two orders of magnitude larger than the fire contribution [3.8].

For Limerick, a General Electric BWR/4 with a Mark II containment, the fire-initiated sequences are mainly found to be in a release category involving overpressure failure with an intact and functioning suppression pool. The subcooled pool is analyzed to be effective in scrubbing radionuclides before release. The assumed ejection of 15% of the pool water to the environment at containment failure dominates the calculated releases for this type of scenario. A small contribution to early-fatality risk is found, but the fire-initiated sequences contribute only on the order of 1% of the total early-fatality risk [3.4-3.5].

To summarize the above discussion, for each of these six reactors the fire PRA finds that almost all of the calculated core-damage frequency is associated with late overpressure containment failures or core-damage sequences into an intact containment. The early-containment-failure, "large-release" scenarios are either absent (that is, found not to be present) or found to be very minor contributors (less than 10^{-9} per year).

3.3.2.3 Methodological Issues Affecting the PRA Study Conclusions

As mentioned at the beginning of Section 3.3.2, the validity of comparisons of the various fire PRA studies with the two figures-of-merit, and with each other, depends on whether the methodology used is valid and whether the various PRAs were done on a common basis.

In this subsection, the extent to which these issues affect the overall conclusions is explored.

The methodology for fire PRA is relatively straightforward at a conceptual level, and all of the fire PRAs studied here use the same basic structure, consisting of five steps:

1. The analyst must identify important accident scenarios that might be produced by an internal fire, and then identify the important equipment, structures, and other plant safety features whose compromise would lead to an undesirable accident. This identification task, which also involves identifying the location of the susceptible equipment, is usually accomplished in the context of a larger (full-scope) PRA covering other initiators such as transients and LOCAs.

2. The analyst must determine the frequency with which a fire might occur in each important area identified in the first step.

3. The analyst must work out the probability of damage to vital equipment, given the initiation of a fire. This usually involves analyzing fire growth and spread, how it competes with fire suppression, and how fire "size" affects the likelihood and extent of "damage" or "failure".

4. The analyst must carry out a PRA systems analysis to tie together the fire-caused failures with other (non-fire) failures to identify undesired accident sequences. This usually involves traditional PRA-type event-tree/fault-tree methods.

5. The analyst must quantify the trees to identify combinations of failures leading to the undesired end-point. This is also a traditional PRA-type analysis step.

Although this basic structure has been followed in all of the PRAs studied, there has been a steady evolution in the methodology from the earliest studies (for example, the fire PRA for Big Rock Point) to the most recent ones. There seem to be three broad generations of fire PRAs, characterized as follows:

1) The first generation, of which the pioneering Big Rock Point study is typical, used considerable engineering judgment in the analysis, limited the analysis to a few critical areas, and obtained what would now be considered only approximately accurate numerical results. The Big Rock Point study was an important early step in the methodology, but lacked an extensive data base and did not utilize a detailed analysis of fire suppression phenomena.

2) The second generation, of which the Indian Point and Zion analyses are typical, used a stronger data base of fire occurrence frequencies, and employed specially-developed computer codes to study fire growth, fire suppression, and their interaction. These analyses also provide quantified uncertainties based on a rationale, which rationale is embedded in the way the analyses treat data-type and modeling-type uncertainties, combine them, and express them. However, these analyses employ engineering judgment as the main ingredient in identifying critical fire areas.

3) The third generation, of which the Millstone-3, Limerick, and Seabrook studies are typical, includes the most recent and most advanced fire PRAs. PRAs in this group have an improved approach to selecting critical fire areas. The approach uses "location analysis" to identify the physical location of each component or structure important to safety, thereby making the identification of proximity of critical components more automated. Also, these analyses combine fire-caused and non-fire-caused failures in the cut sets, which improves the scope and validity of the analysis.

Each of the various PRAs considered different areas as being susceptible to fires. These differences mean that comparability among them is only approximate. Table 3.2, reproduced from Wheelis' recent Sandia review [Ref. 3.20], shows the areas considered by several of the PRAs used in the fire data base. The table reveals that as the methodology has evolved from the earlier to the later studies, more and more areas are considered significant enough to include.

In addition to the broader trends in an evolving methodology, each of the fire PRAs has used somewhat different models for calculating specific phenomenological aspects, and a slightly different data base. (This generalization is not quite correct, in that some of the PRAs, performed by the same analysis team at about the same time, have used almost the same approach --- an example is the similarity among the Zion, Indian Point-2, and Indian Point-3 analyses.)

On the basis of these identified differences, any comparison of the numerical results on, for example, mean core-damage frequency cannot be done on a strictly common basis. However, the estimated numerical uncertainties in the results are in the range of about one order-of-magnitude or perhaps more, and it is the judgment of the fire-PRA community that the study-to-study differences due to methodology and data base aspects are probably less than this order-of-magnitude spread. Therefore, it is concluded that, within the uncertainties stated, comparisons among the several fire PRAs cited are valid.

More important than study-to-study differences in the methodology are issues related to inadequacies in the methodology. Inadequacies include an incomplete data base for some issues, the incomplete analysis of certain phenomena, and certain approximations made in the extent analyses whose refinement could lead to different insights. In the following paragraphs we will discuss the main inadequacies to provide an appreciation of the level of maturity of fire PRA methodology.

Frequency of fires: The data base used by PRA analysts for fire frequency is taken from nuclear power plant experience, and does not rely on non-nuclear fire experience. This is acceptable because there have been enough small fires in the nuclear experience to provide an acceptable starting point for the PRA analysis. The data base for fire occurrence is usually disaggregated by combining all fires that have happened in a given type of building (turbine building, auxiliary building, etc.) or occasionally in a given type of room (cable spreading room, control room).

TABLE 3-2

General Classification of Areas Examined and Quantified in Fire PRAs
(from Wheelis, 1984)

	Control Room	Cable Spreading Area	Switchgear Room	Pump Room	Diesel Generator Rooms	Turbine Bldg.
Big Rock Point						
Station Power Room		X				
Inside Cable Penetration Area		X				
Outside Cable Penetration Area		X				
Control Room	X					
HTGR						
Cable Spreading Room		X				
Indian Point						
Cable Spreading Room		X				
Electrical Tunnel (PAB)		X				
Switchgear Room			X			
Diesel Generator Building					X	
Limerick						
Safeguards Acces Area		X				
Auxiliary Equipment Room			X			
13 KV Switchgear Room			X			
Static Inverter Room			X			
Cable Spreading Room		X				
Control Room	X					
CRD Hydraulic Equipment Area		X				
General Equipment Area				X		
Millstone 3						
Cable Spreading Room		X				
Control Room	X					
Instrument Rock Room			X			
Switchgear Rooms			X			
Electrical Tunnels		X				
Motor Control and Rod Control Area			X			
Charging and Component Cooling Pump Area				X		
Circulating and Service Water Building				X		
Diesel Generator Rooms					X	
Seabrook						
RHR Spray Vault				X		
Cable Spreading Room		X				
Control Room	X					
Electrical Tunnels and Penetration Area		X				
Chiller Pump Area				X		
PCC Pump Area				X		
Electrical Chase (vertical)		X				
Turbine Building						X
Service Water Building				X		
Resin Fill Tank Area			X			
Zion						
Auxiliary Electrical Equipment Room			X			
Inner and Outer Cable Spreading Room		X				
Auxiliary Building Pump Room				X		

An area-ratio approach is then used to find an appropriate fire-initiation frequency for the actual building or room being studied. Usually, the analyst's judgment is used to modify the raw data in the data base, taking into account such factors as the local fuel loading and the likelihood of transient fuels. Weighting factors for co-location can also be applied. This approach is probably satisfactory given other uncertainties in the analyses, but could be improved by more and better data.

Room-to-room spreading: One of the most important possible flaws in the current methodology is its treatment of room-to-room fire spreading. Most of the extant analyses have used judgment in ascertaining that a fire in a given room will not spread to adjacent spaces. It is typically assumed that the design time-rating of fire barriers correctly describes their actual performance. Thermal barrier performance is usually analyzed by determining the total fuel loading in a given room, calculating the thermal wall loading based on that fuel, and comparing with the barrier's time rating. This approach is generally acceptable, but could be non-conservative if the analysis does not account for fuel being concentrated, say, in one corner of a room. Also, dampers and doors connecting spaces are assumed to remain normally closed, or to close automatically, if so designed. The extent to which these assumptions and approximations are correct is difficult to evaluate, but methodological improvements in this area could enhance our confidence in the study results.

Spread of hot gases: Another area of possible methodological weakness is the treatment of the spread of hot smoke and gas by ventilation systems. This is now analyzed using engineering judgment, because models to calculate these effects are difficult to develop and use.

Control circuitry: Still another issue is the vulnerability of control circuitry in a control-room fire. Spurious actuation of equipment can cause off-normal behavior that the operators may find difficult to control while they are fighting the fire. This entire area is not well analyzed yet. Also, remote shutdown capability can be compromised if a control-room fire damages key circuitry, and spurious operations may not be indicated properly at the remote shutdown panel.

Suppression: The current approaches to analyzing suppression usually use an exponential model for the distribution of times to suppress, but how accurate these are in the tails of the distribution is not well established.

Damage assessment: Assessing the damage to equipment nearby but not directly within a fire is a difficult task, for which the real-world and experimental-test data base is not strong. More research is definitely needed on this issue.

Equipment damage from suppression: Suppression typically involves spreading liquids (water !) or gases (carbon dioxide, halogen compounds, etc.) over a fire. These materials can damage equipment not otherwise involved, and the analysis of this effect is not currently based on a strong enough data base, so it involves judgment and conservative assumptions.

Smoke control: PRAs do not adequately consider smoke-control issues, including impairment of operator and firefighter actions, generation of spurious signals, and damage to equipment. This is a difficult area to model well.

Manual Fire-Fighting Effectiveness: PRAs cannot model well the actions of firefighters, including various adverse effects such as water sprayed onto equipment, smoke spread by portable blowers, and the compromising of fire barriers by the fire fighters.

Fire-earthquake linkages: There are three issues here. First, a large earthquake can cause a fire, due to electrical failures. This issue has never been studied systematically. Second, earthquakes can damage fire-suppression equipment (fire-water piping, for example) or cause spurious actuation of fire suppression equipment. This has also not been studied thoroughly. Third, earthquakes can hamper access by firefighters to vital areas. (A fourth earthquake-fire issue involves fire barriers whose failure could aggravate an earthquake-initiated accident, most obviously because many fire barriers are built of unreinforced blocks that could fall during an earthquake. The seismic-PRA analysis methods do consider this type of effect, which has sometimes been found to be important.)

The current NRC research program in fire protection is significantly smaller than in earlier years. This is largely due to the overall decrease in the NRC research budget rather than to any changed perception of the relative importance of fires. Most of the fire research program is devoted to experimental and code-development work, to enable more realistic modeling and evaluation of fire-protection features and fire-suppression equipment. The program is supporting tests of fire burning characteristics, smoke and other combustion products, and related phenomena in well-instrumented studies. For electrical components, actual fire tests as well as simulated test-chamber studies are being supported.

In the modeling area, several models of various phenomena have been developed and are being tested and verified, including models for cable-tray fires, pool fires, zonation models for room and corridor fires, and a multi-compartment model for studying aerosols and sprays.

All of this work is important, and it will all eventually lead to an improved ability to model fire phenomena realistically. Each area studied represents an issue where our current understanding or our current analysis capability is not adequate, or both.

It is important to emphasize that the mere existence of this research program should not, on its face, lead to conclusions one way or the other as to the adequacy of current regulations or of current plant fire-protection capabilities. An evaluation of current fire-protection adequacy must be made on a different basis.

Based on the discussion above, it is clear that the methodology for fire PRA is not as mature in some areas as would be desirable. Furthermore, each new fire PRA produces improvements in the methodology, so there is a slow advance over the years. However, there do remain substantial uncertainties in the validity of the numerical results. Of course, the analysts are aware of the various limitations, and they attempt to capture them in the quoted uncertainty ranges in the published studies. Unfortunately, the adequacy of these published uncertainty ranges, typically about one order-of-magnitude or more, is not confidently known.

3.3.2.4 Evaluation and Comparison with Figures-of-Merit

Based on the discussion above of methodological accomplishments and uncertainties, it is possible to make a judgment concerning the comparison of the published PRA results with the two figures-of-merit introduced in Section 3.1 above. This judgment is as follows:

Figure-of-Merit # 1, Core Damage Frequency: The results for core-damage frequency from fire-initiated accidents, displayed in tabular form in section 3.3.2.1, reveal on their face a central conclusion, as follows: many of the reactors have calculated core-damage frequencies comparable to the figure-of-merit guideline, which is in the range of 10^{-5}/reactor year. Despite the uncertainties and methodological issues discussed, the central conclusion is valid.

Figure-of-Merit # 2, Frequency of a Large Release: The conclusions here are a summary of the discussion above in Section 3.3.2.2. Section 3.3.2.2 examines five fire PRA studies, for Indian Point -2 and -3, Zion, Limerick, and Oconee-3. The broad summary is that almost all of the calculated fire-initiated core-damage frequencies, for each reactor studied, are associated with late-overpressure containment failure sequences or sequences involving an intact containment. The fraction of fire-initiated core-damage sequences associated with an early containment failure (a "large release") is in every case very small, and in some cases found in the analysis to be zero. Taking this conclusion at face value, one is led to conclude that large-release frequencies are significantly smaller than the value of figure-of-merit # 2, which is 10^{-6} per reactor year. (Of course, generalizing this conclusion to all reactors may not be appropriate.)

3.4 SUMMARY AND CONCLUSIONS

3.4.1 Conclusion # 1: Fires Must Be Included

Based on the discussion above, it is clear that internal fires must definitely be included among those "external initiators" that should be part of any analysis of nuclear power plant vulnerabilities. Omitting full consideration of internal fires from a vulnerability analysis would not be appropriate. The core-damage frequencies calculated are often in the range of interest for figure-of-merit # 1.

3.4.2 Conclusion # 2: "Large Releases" Are Found to Be Unimportant

Based on the discussion above, the five PRAs examined all found that the likelihood of a "large release" is significantly smaller than the value of 10^{-6} per rector year embodied in figure-of-merit # 2. Whether this conclusion applies generally to all reactors is not known.

3.4.3 Identification of Areas in Which Existing Requirements Have Not Ensured Adequate Protection

This issue is very difficult to grapple with on a generic basis. The various plants seem to have quite different vulnerabilities in detail. There are a few broad and general insights, but they are so obvious as to be of little use ---- for example, nobody needs to do an expensive and elaborate PRA study to learn that cable-spreading rooms and control rooms are both key areas worth paying attention to!

Although the list is obvious, there are a few areas where special attention is called for in the analysis. The most important are probably control rooms, cable-spreading areas, areas containing large numbers of electrical switchgear cabinets, and areas with significant likelihood of transient fuels.

The existence of these areas where careful analysis is needed does not mean that the current regulations are necessarily inadequate. Significant research and analysis is surely required before any changes in requirements would be supportable.

3.4.4 A Proposed Approach for Plant Evaluation

The only methodology currently available for the realistic evaluation of the impact of fires is the PRA-based methodology discussed above. The most advanced methodology is the third-generation PRA methodology described above. However, a full-scope PRA is not necessarily required. It is likely that an abbreviated analysis methodology, if it existed, could accomplish most of what can now be learned only from a full-scope fire PRA. [An example in another topic area is the study of earthquake-initiated accidents, using either the "simplified seismic PRA methodology" as proposed by Lawrence Livermore National Laboratory, or of the "seismic margins review methodology" that has recently been developed by NRC and EPRI.]

Development of such a simplified or screening methodology for analysis of internal fires has not yet been accomplished. Therefore, as of this writing the only developed and acceptance method now available for realistic fire analysis is a full-scope fire PRA study. However, recent developments now make possible a much more efficient screening process than previously, so that the analyst can quickly narrow his analysis down to the few key areas of

concern. Especially if the fire PRA is done in conjunction with analyses of internal initiators (as in the proposed Individual Plant Evaluation or IPE approach), the existence of systems models, a support-system dependency matrix, and success criteria can make the fire PRA only a modest additional burden.

The fire PRA need not involve every analytical feature of the most elaborate full-scope fire PRAs. To satisfy the purpose here, it is acceptable to focus attention on the two figures-of-merit and to use them to screen out certain accident sequences. For example, it may be possible to identify and separate those fire-initiated core damage accident sequences that can be shown to occur into an intact containment from those sequences leading to late overpressure containment failures, and from those very few sequences with the potential for a large early release. Attention could then be focused on this last group for the purposes of comparing with figure-of-merit #2. In parallel, comparison with figure-of-merit #1 will allow the screening out of those sequences whos core damage frequency is found to be very low.

3.5 REFERENCES

[3.1] Big Rock Point, 1981: Consumers Power Company, "Big Rock Point Nuclear Power Plant Probabilistic Risk Assessment" (1981)

[3.2] General Electric, 1982: General Electric Company, "GESSAR II, Nuclear Island General Electric Standard Safety Analysis Report, Section 15 -- BWR/6 Standard Plant Probabilistic Risk Assessment" (1982)

[3.3] Indian Point, 1983: Consolidated Edison Company and Power Authority of State of New York, Indian Point Probabilistic Safety Study (1983). The principal contractor was Pickard Lowe & Garrick Inc.

[3.4] Limerick PRA, 1981: Philadelphia Electric Company, Probabilistic Risk Assessment, Limerick Generating Station (1981). The principal contractor was General Electric Company, with assistance from Science Applications, Inc.

[3.5] Limerick SARA, 1983: NUS Corporation, Severe Accident Risk Assessment, Limerick Generating Station, NUS Report No. A-4161, prepared for Philadelphia Electric Company (1983).

[3.6] Millstone-3, 1983: Northeast Utilities, "Millstone Unit 3 Probabilistic Safety Study" (1983)

[3.7] WASH-1400 NRC, 1975: U.S. Nuclear Regulatory Commission, "Reactor Safety Study", (1975)

[3.8] Oconee, 1984: Electric Power Research Institute/Nuclear Safety Analysis Center and Duke Power Company, "Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3", in 4 volumes (1984)

[3.9] Seabrook PRA, 1983: Pickard Lowe and Garrick, Inc., "Seabrook Station Probabilistic Safety Assessment", prepared for Public Service Company of New Hampshire and Yankee Atomic Electric Company, Report PLG-0300 (1983)

[3.10] Sizewell B, 1982: U.K. Central Electricity Generating Board, "Sizewell B PWR Pre-Construction Safety Report", Chapter 15 (1982)

[3.11] Zion, 1982: Commonwealth Edison Company, Zion Probabilistic Safety Study (1982). The principal contractor was Pickard Lowe & Garrick, Inc.

Other References Cited:

[3.12] SAND86-0834C Berry, 1986: D.L. Berry, "Safety Significance of Fires", Sandia National Laboratory, Albuquerque, NM. (1986)

[3.13] NUREG/CR-4334 Budnitz/Expert Panel, 1985: R.J. Budnitz, P.J. Amico, C.A. Cornell, W.J. Hall, R.P. Kennedy, J.W. Reed, and M. Shinozuka, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants", Lawrence Livermore National Laboratory, Livermore, CA. (1985)

[3.14] EPRI/Margins, 1987: NTS Engineering, RPK Consulting, Pickard Lowe & Garrick, Woodward Clyde, Duke Power Co., "Evaluation of Nuclear Power Plant Seismic Margin", Electric Power Research Institute (draft, March 1987)

[3.15] Kazarians, Siu, and Apostolakis, 1985: M. Kazarians, N.O. Siu, and G. Apostolakis, "Fire Risk Analysis for Nuclear Power Plants: Methodological Developments and Applications", Risk Analysis, Vol. 5, p, 33-51 (1985)

[3.16] NUREG/CR-4331 Shieh et al., 1985: L.C. Shieh, J.J. Johnson, J.E. Wells, J.C. Chen, and P.D. Smith, "Simplified Seismic Probabilistic Risk Assessment: Procedures and Limitations", Lawrence Livermore National Laboratory, Livermore, CA (1985)

[3.17] NUREG/CR-2015 SSMRP, 1981: P.D. Smith, R.G. Dong, D.L. Bernreuter, M.P. Bohn, T.Y. Chuang, G.E. Cummings, J.J. Johnson, R.W. Mensing, and J.E. Wells, Lawrence Livermore National Laboratory, "Seismic Safety Margins Research Program, Phase I Final Report", in 10 volumes, Lawrence Livermore National Laboratory, Livermore, CA. (1981 and 1982)

[3.18] NRC, 1985: U.S. Nuclear Regulatory Commission, "Policy Statement on Severe Reactor Accidents", Federal Register, Volume 50, p. 32138 (August 8, 1985)

[3.19] NUREG-1080 NRC, 1985a: U.S. Nuclear Regulatory Commission, "Long Range Research Plan, FY 1986 - FY 1990", Vol. 2 (1985)

[3.20] Whellis, 1984: W.T. Wheelis, "RMIEP Fire Risk Analysis: Review of Current Fire Risk Analyses", Sandia National Laboratories, Albuquerque, NM. Unpublished letter report to NRC (December 1984)

CHAPTER 4 --HIGH WINDS/TORNADOES

4.1 INTRODUCTION

Tornadoes are loosely characterized as violently rotating columns of air, often associated with severe thunderstorms and usually having the appearance of a funnel dipping down from the base of a cloud above it. The tornado vortex behaves erratically, lifting off the ground at irregular intervals and its path is unpredictable. When the tornado vortex touches the ground, severe damage to physical property, livestock, and people may result. Although actual weather systems which produce severe thunderstorms may take several hours to develop, the actual tornado produced from these systems usually develop and strike within a matter of minutes. This time scale is in contrast to the development of other large severe storm systems, such as hurricanes and typhoons, whose development can be observed and tracked over a period of days, thus allowing preparations for their arrival. Fortunately, it has been estimated that only one out of every 100 associated thunderstorms will produce a tornado [4.1].

High winds from tornadoes, hurricanes or wind storms are a potential threat to nuclear power plants due to the risk of damage to the reactor core and the release of radioactive material from the reactor core. This damage may be due to pressure differentials between the inside and the outside of a tornado, missiles generated by a tornado, or direct damage due to dynamic wind loadings. If a missile induced by high winds or a tornado were to impact a structure, critical components or other equipment inside the structure might be damaged or lost.

As the winds associated with hurricanes and other storms are less intense and lower in magnitude than those associated with tornadoes, this chapter will concentrate its discussion on the current state of knowledge regarding risk due to tornadoes. Generally, high winds from wind storms and hurricanes are considered to be the controlling wind level at a higher frequency but at a lower magnitude. Tornadoes usually represent the controlling or design basis wind levels for nuclear power plant at a lower frequency but at a higher magnitude. This is shown by Figure 4.1.

Tornadoes and violent wind storms may occur practically anywhere in the continental United States but occur more frequently in the Eastern half. Figure 4.2 shows the number of tornadoes occurring in the U.S. as recorded by the National Severe Storms Forecast Center (NSSFC) by 1° box from 1954 to 1983 [Ref. 4.22]. Figure 4.3 shows the number of wind storms with speeds of 50 knots (57.5 miles per hour) or greater occurring in the U.S. by 1° box from 1955 to 1967 [Ref. 4.23].

This chapter presents a review and evaluation of the risk from reactor core damage or large radioactive release due to high winds or tornadoes. The purpose of this chapter is to meet work requirements 2, 4 and 5 given in Chapter One of this report for high winds and tornado hazards.

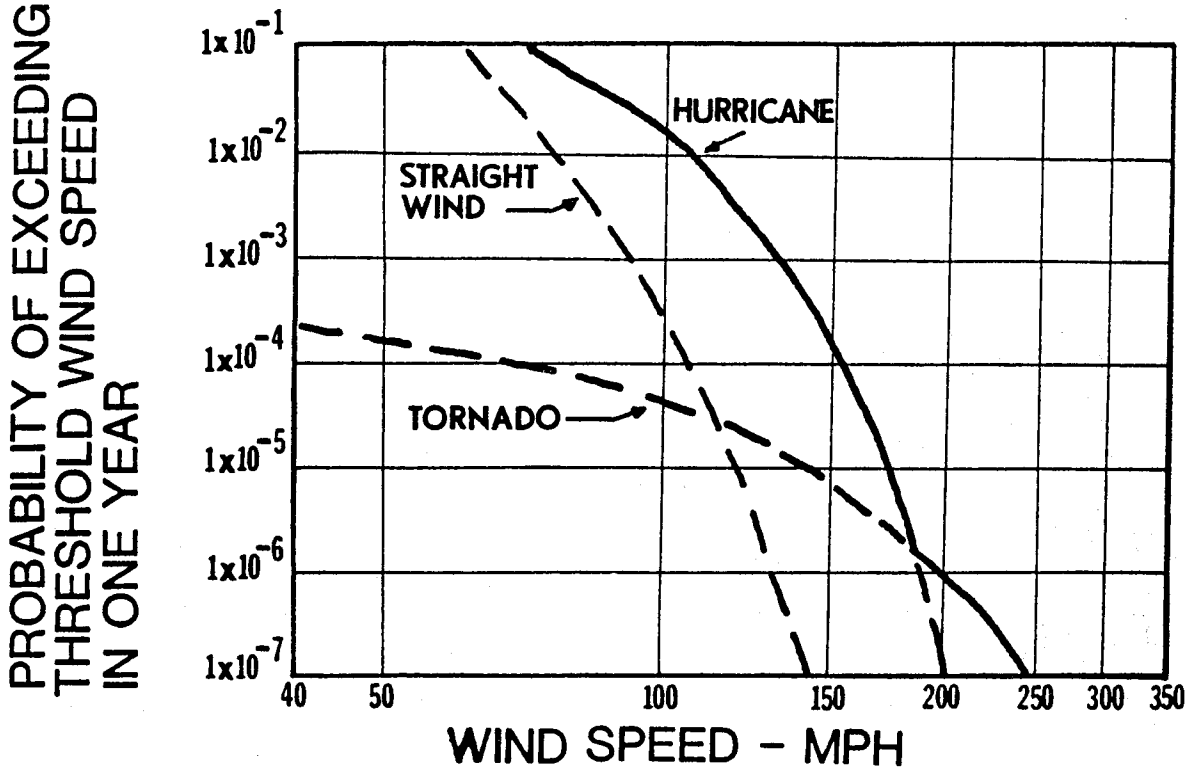


Figure 4.1 Typical Tornado, Hurricane and Straight Wind Hazard Probability Models

Note: Relative positions of straight wind and hurricane probability models could be interchanged depending on site.

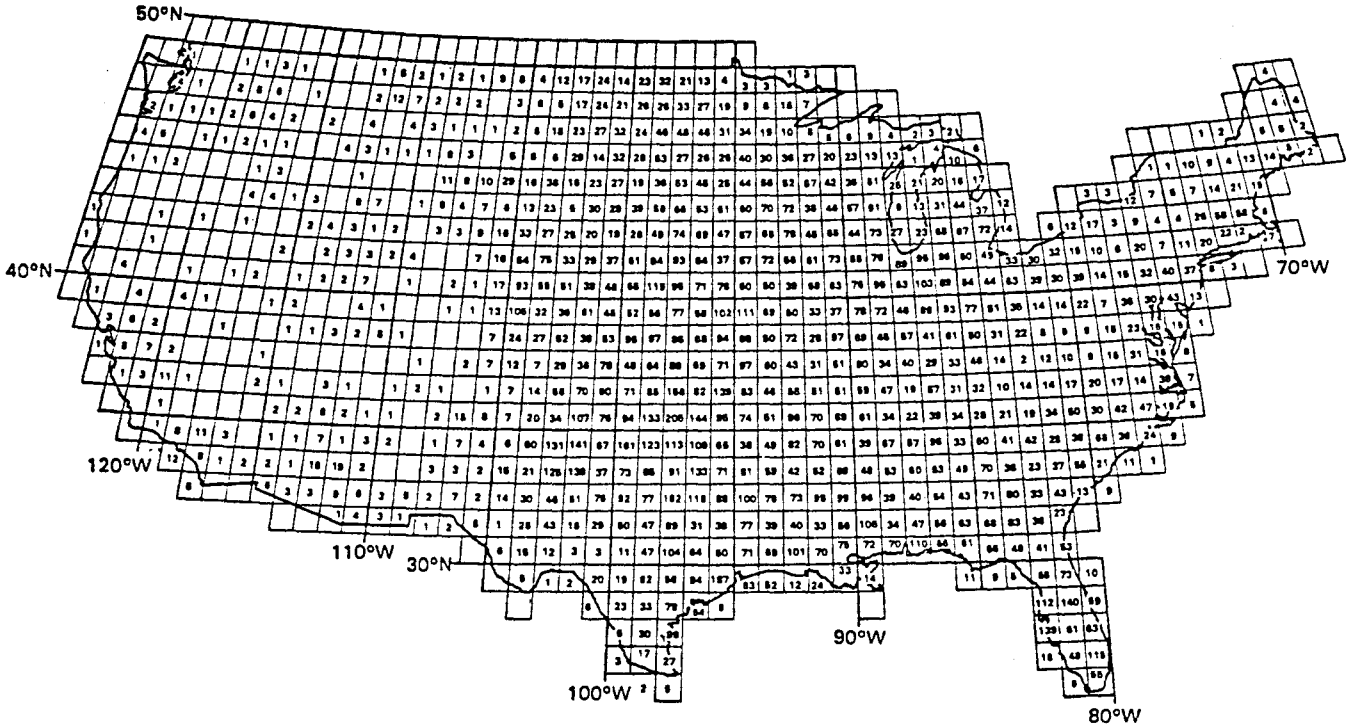
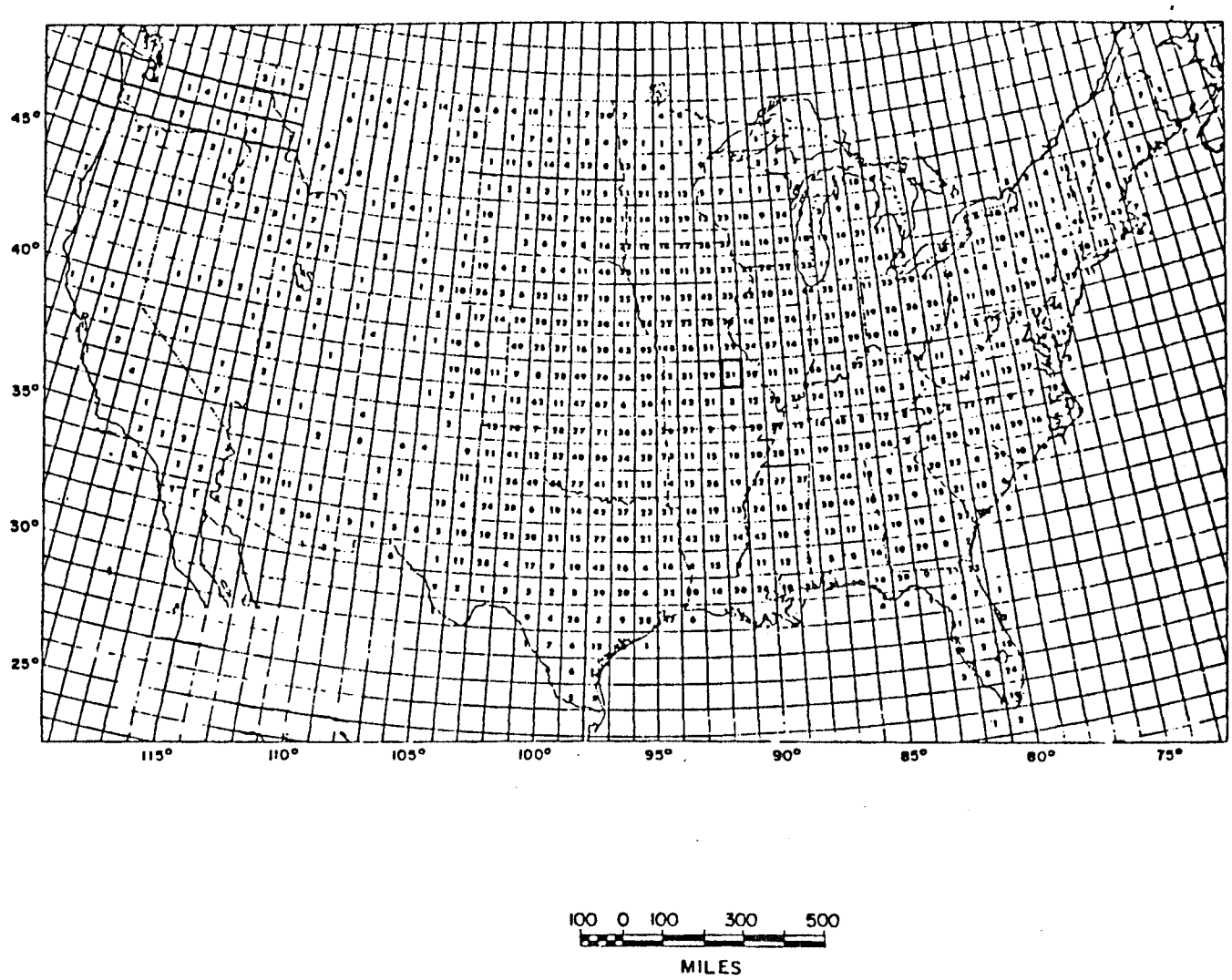


Figure 4.2
Number of Tornadoes in U.S. by 1° Box (1954-1983)
[Ref. 4.22]



FERENCE:
PAUTZ, M.E., 1969: SEVERE LOCAL STORM OCCURRENCES 1955-1967. ESSA TECH. MEMO WBTM FCST 12, OFFICE OF METEOROLOGICAL OPERATIONS, SILVER SPRING, MD.

Figure 4.3
Number of 50 Knot (57.5 Miles/Hour) Windstorms in U.S.
by 1° Box (1955-1967) [Ref. 4.23]

4.2 CURRENT NRC REGULATORY REQUIREMENTS

The regulation of nuclear power plants in regards to protection against high winds and tornado hazards is given by 10 CFR 50 and 10 CFR 100 [4.2] listed in Chapter Two of this report under the High Winds/Tornadoes category. The guidance given to the NRC staff by the SRP [4.3] in reviewing Safety Analysis Reports (SARs) in regards to protection against high winds and tornado hazards is also given in Chapter Two of this report under the High Winds/Tornadoes category.

The specific guidance given by the NRC staff for the design of nuclear power plants for adequate protection against high winds and tornado hazards that are of particular interest are NRC Regulatory Guides 1.76 and 1.117.

Regulatory Guide 1.76 defines three tornado regions for the United States and provides a maximum wind speed, maximum rotational wind speed, Vr, a maximum and minimum translational wind speed, Vt, a radius of maximum rotational speed, a pressure drop across the tornado, and the rate of pressure drop across the tornado [4.4]. Regulatory Guide 1.117 specifies the plant systems, structures, components, areas, etc., that must be protected against tornadoes [4.5].

Even with the use of the Regulatory Guides, additional background material and information may be needed in order to provide the necessary detail to design nuclear power plant protection against high winds/tornado hazards. Two important documents used for this purpose are:

ANSI A58.1 Building Code Requirements for Minimum Design Loads in Buildings and Other Structures, American National Standards Institute Committee A58.1-1972.

ASCE Paper No. 3269 "Wind Forces on Structures", Transactions of the American Society of Civil Engineers, Vol. 126, Part II, 1961.

ASCE Paper 3269 [4.6] and ANSI A58.1 [4.7] serve as the detailed instructions to transform wind velocities into pressure loadings on structures and provides the associated vertical distribution of wind pressures and gust factors.

ASCE Paper 3269 is basically a compilation of research papers on structure design for various loading conditions including, among many others, high wind/tornadoes. ANSI A58.1 presents basically the same requirements and information for high wind/tornado loading conditions as ASCE 3269 except that its information is more formalized into design procedures.

Plants that have undergone licensing review, have either been designed and constructed to meet the requirements of ASCE Paper 3269 or to meet the requirements of ASCE Paper 3269 and ANSI A58.1. It is generally the older plants that have been built according to requirements of ASCE Paper 3269. The

newer plants, in contrast, have generally been built to meet ASCE Paper 3269 and ANSI A58.1. The design requirements between the two papers in so far as high winds/tornado are concerned, are not major and will be considered the same for this report.

4.3 TORNADO RISK BACKGROUND

An evaluation of the risk to nuclear power plants from tornadoes requires a classification of the plant damage states possible and assessment of the site-specific frequency of occurrences of tornadoes capable of producing those plant damage states. For this study, the plant's response to these wind damage states are taken strictly from Probabilistic Risk Assessment (PRA) studies of nuclear power plants. This is discussed further in Section 4.4. PRA studies are not the only source of information for plant response to damage states cause by tornadoes. Licensing documents such as Safety Analysis Reports (SARs) and Safety Evaluation Reports (SERs) also discuss the plant's response to the extreme forces imposed by violent tornadoes. For this study, however, no other source could be found which provided a numerical estimate of the plants response to tornado hazards which could be used to compare with the figure-of-merits discussed in Chapter One.

Of the 11 PRAs reviewed for high wind/tornado hazard, only one plant determined the frequency of large radioactive release following core damage, all others ending their analysis at core damage frequency. Therefore, this chapter will compare the plant core damage frequency with the first figure-of-merit given in chapter one of 10^{-5} per reactor year for core damage frequency.

For further discussion of the risk analysis approach, definition of terms, etc., used in determining the risk to nuclear power plants from tornadoes, see Appendices 4.A.1 to 4.A.6.

4.4 REVIEW OF HIGH WIND/TORNADO PRAs

This subsection presents a brief summary of each of the 11 PRAs studied for the plant's response to the high wind/tornado hazard. Basically, each PRA summary consists of the initiating event frequency and the probability of the core damage frequency due to the high wind/tornado event with important assumptions and conditions made during the probabilistic analysis.

The PRA for Indian Point 2 and 3 [4.8] used a median tornado wind exceedance frequency of 1 E-4 per year. No basis was given for this frequency. The tornado strike model was of the point target class. The tornado strike frequency was assumed to be the same as the tornado wind exceedance frequency due to the use of the point target strike model. Because the damage due to tornado missiles was assumed to be localized, the risk due to tornado winds and straight winds was considered to be the controlling factor. For Indian Point 2, the mean and median core melt frequency due to high winds were determined to be 4.3 E-5/yr. and 2.5 E-5/yr. respectively. The range at the 90% confidence level was 1.2 E-4/yr. to 2.9 E-6/yr. Because

it was felt that there was much conservatism in the initial risk calculation for Indian Point Unit 2, additional analysis was performed taking advantage of site-specific directional wind data, structure wind fragilities, structure tornado missile fragilities, plant building location and shadowing effects. With this additional analysis, the mean and median were determined to be 3.6 E-5/yr., and 1.8 E-5/yr., respectively. For Indian Point 3, the mean and median high wind core melt frequency was determined to be 1.3 E-6/yr. and 4.9 E-8/yr. with the range at the 90% confidence level of 6.5 E-6/yr. to 0. The core melt frequency due to tornado missiles for each unit was less than 1 E-7.

The interesting feature of the Indian Point PRA was the effect of the Unit 1 superheat stack. This structure was unrelated to Unit 2 in terms of plant operation, but its collapse due to high winds and/or tornado missiles could affect the Unit 2 diesel generator building and/or the Unit 2 control building. Critical structures at the Indian Point site sensitive to tornado missiles were found to be the Unit 2 control building, the Unit 2 diesel generator building, the Unit 2 Primary Auxiliary Building (PAB) containing the safeguards motor control centers, the component cooling water heat exchangers and surge tank, the Unit 2 Auxiliary Feedwater (AFW) building, the Unit 3 AFW building, the Condensate Storage Tank (CST), the Refueling Water Storage Tanks (RWSTs), backup city water tanks, the exterior service water pumps, and the offsite power supply including the station auxiliary transformers and gas turbines. An additional note, the Indian Point PRA was the only study that calculated a radioactive release probability for high wind/tornado events. The most probable frequency of a large release for Unit 2 was found to be between 1 E-4 and 1 E-5 due to failure of the control room causing a loss of control, and failure of the gas turbine and diesel generator buildings causing a loss of emergency power. The most probable frequency of a large release for Unit 3 was found to be between 1 E-5 and 1 E-9 for the same accident sequence [4.8].

The Limerick PRA [4.9] used a tornado frequency of 1.13 E-4 tornadoes per square mile per year based on a 30 year record (1950-1980) of 37 tornadoes recorded near the Limerick site. The tornado strike model was of the aerial target class. The tornado strike frequency for tornadoes of F1 intensity or greater was 2.3 E-4/yr. The tornado strike frequency of F5 intensity was 1 E-7/yr. The critical structures that were considered sensitive to tornadoes due to their being designed to lower standards than the design basis tornadoes were: 1. The electrical transformers and substations, 2. The turbine enclosure, 3. The condensate storage tank, and 4. The cooling towers. The core melt frequency due to tornado winds was determined to be 9.0 E-9/yr. The risk due to tornado missiles was considered to be smaller than the risk due to tornado winds due to the localized nature of tornado missile damage. An interesting feature of the Limerick plant design is that offsite power is supplied to the station via underground cables from the substation. The use of these cables should help preserve the protection afforded by the physical separation of two substations from the turbine enclosure. The cables surface in the turbine enclosure as cable buses that are fed into the control enclosure [4.9].

The Millstone 3 PRA [4.10] used a tornado frequency of 1.87 E-4 tornadoes per square mile per year for the one-degree (latitudinal-longitudinal) square within which the Millstone site is located. The tornado strike model was of the point target class. The tornado strike frequency was assumed to be the same as the tornado frequency due to the use of the point target strike model. The core melt frequency due tornado missiles is given as being less than 1 E-7/yr. and the core melt frequency due to high winds was simply stated as being sufficiently low [4.10].

The Oconee 3 PRA [4.11] was basically a site-specific application of the tornado missile risk studies done by the Electric Power Research Institute (EPRI) and by Science Applications, Inc. (SAI) in 1978. The site specific tornado frequency used for the study was 2.5 E-4 tornadoes per square mile per year based on a 30 year record (1950-1980) of 77 tornado segments reported within 50 nautical miles of the Oconee site. A tornado segment was defined as the continuous damage path associated with one touchdown of a tornado. A single tornado could have several segments. The Oconee 3 plant strike model was of the aerial target class. The tornado strike frequency for Oconee was determined to be 3.5 E-3/yr. for the 1 mile site radius and 1.2 E-3/yr. for the 2000 ft. site radius. An extensive compilation of potential tornado missiles at the Oconee site was done along with missile strike frequency calculations on various safety-related plant systems, components and areas. The tornado core damage frequency was determined to be less than 1.0 E-9/yr. with damage to safety equipment at 2.0 E-5/yr. The significant wind accident initiating tornado events were: 1. Loss of load from damage to the switchyards (frequency of 3.0 E-4/yr.), 2. Loss of feedwater (frequency of 2.2 E-7/yr.), and 3. Steam-line break outside of containment (frequency of 1.2 E-6/yr.) [4.11].

The Seabrook 1 and 2 PRA [4.12] used a tornado frequency of 1.26 E-3 tornadoes per square mile per year based on 69 tornadoes recorded within 50 miles of the Seabrook site from 1950 thru 1977. The tornado strike model was of the point target class. The tornado strike frequency was calculated to be 7.77 E-5/yr. The core melt frequency due to high winds was assumed to be less than 3.89 E-8/yr., which was the frequency determined for tornado winds exceeding 360 mph. The core melt frequency due to tornado missiles assuming loss of offsite power and the RWST was 2.06 E-9/yr. The core melt frequency due to tornado missiles assuming loss of offsite power and both emergency diesel generators was 3.4 E-10/yr. For both cases, offsite power was assumed not recovered [4.12].

The Zion 1 and 2 PRA [4.13] used a mean tornado frequency of 1.0 E-3 tornadoes per square mile per year. Its basis was not given. The tornado strike model was of the point target class. The tornado strike frequency was assumed to be the same as the tornado frequency due to the use of the point target strike model. No core melt frequency due to high winds or tornado missiles were explicitly given although it was judged to be acceptably small for tornado missiles. The probability of loss of offsite power due to high winds was assumed to be 1.0 E-3 [4.13].

Five nuclear power plants were the subject of an abbreviated PRA under the auspices of the NRC's program for resolution of generic safety issue A-45 (decay heat removal). These PRAs were not full-scope PRAs but the analysis procedures used to perform the wind analysis did embody the basic philosophy of a full-scope PRA. Sandia National Laboratories studied high winds/tornado vulnerabilities for Arkansas Nuclear One-1, Point Beach 1 and 2, Quad Cities 1 and 2, St. Lucie 1, and Turkey Point 3 power plants [4.14 - 4.18]. Site specific tornado frequencies were found for all five plants based on NSSFC tornado records for a 145 mile radius around each plant from about 1950 to 1983. The tornado strike model for all five plants was of the aerial target class. The core melt frequency due to high winds alone, tornado winds and/or missiles, and both high wind and tornadoes for these five plants are presented in Table 4.1. All of the core melt frequency calculations were done assuming loss of offsite power coupled with and without recovery. The interesting feature of these PRAs is that all these plants except for St. Lucie 1, had structures on site which dominated the risk due to high winds/tornadoes. For example, Point Beach was vulnerable to failure of the diesel generator exhaust stack which could fail both diesel generators. Turkey Point 3 was vulnerable to the failure of the Unit 2 (a conventional fossil-fuel unit on site) concrete stack which could affect the diesel generator building, the diesel generator transfer pumps, the switchgear building, the Unit 3 RWST, the diesel generator fuel oil storage tank, the condensate storage tank, and the circulating water intake pumps.

Table 4.1 summarizes the tornado frequencies, the tornado strike frequencies, the high wind core melt frequencies, the tornado wind/missile core melt frequencies, and the total high wind/tornado core melt frequencies for the twelve PRAs examined for this study. As these previous summaries of PRAs have shown, there are methodological and data base differences between the PRAs. So comparison of the results of one PRA with another PRA should be done with caution.

Table 4.2 lists the vulnerable structures found at each plant whose PRA was examined for this study. Certain plants were found to have structures whose failure could dominate the accident analysis. These structures were not necessarily safety-related or even part of the plant under investigation, but were within the plant boundary. Their failure could affect safety-related equipment, thus serving to intensify the effect of the initiating event. Not all of the structures listed in Table 4.2 would be damaged or destroyed at the same time. This is because the "noteworthy" structure would be able to damage or destroy only a few of the nearby structures. The rest would be in different directions. Another way of stating this observation would be that many of the affected structures/components in Table 4.2 are mutually exclusive.

Figures 4.4 to 4.10 shows the site plans for Limerick [Ref. 4.9], Oconee Unit 3 [Ref. 4.11], Arkansas Nuclear One [Ref. 4.14], Point Beach [Ref. 4.15], Quad Cities [Ref. 4.16], St. Lucie Unit 1 [Ref. 4.17], and Turkey Point [Ref. 4.18]. Site plans are useful in showing the layout of structures and components at a site and for illustrating how some structures may shield other

structures or components from wind damage. Site plans are also useful in showing how certain vulnerable structures such as a tall stack, could threaten many nearby structures but could only damage a few of the nearby structures simultaneously because of the distribution and orientation of neighboring structure location. A good example of this shown by Figure 4.10, the Turkey Point site. The Unit 2 concrete stack threatens many other structures on site if the stack should fail. However, it cannot damage both the Unit 3 RWST and the diesel generator building because they lie in different directions.

Table 4.1

High Wind/Tornado Plant-Specific PRA Frequencies
[Ref. 4.8 - 4.18]

Plant Name	Tornado Frequency (Any Size) (/mi^2-yr.)	Tornado Strike Frequency (Any Size) (/yr.)	High Wind Core Damage Frequency (/yr.)	Tornado Core Damage Frequency (/yr.)	Total High Wind/ Tornado Core Damage Frequency (/yr.)
Indian Point 2	1.00E-04 (Median)	1.00E-04	2.50E-05 (Median) 4.30E-05 (Mean)	<E-7	2.50E-05 (Median) 4.30E-05 (Mean)
Indian Point 2 (addl. analysis)	1.00E-04 (Median)	1.00E-04	1.80E-05 (Median) 3.60E-05 (Mean)	<E-7	1.80E-05 (Median) 3.60E-05 (Mean)
Indian Point 3	1.00E-04 (Median)	1.00E-04	4.90E-08 (Median) 1.30E-06 (Mean)	<E-7	4.90E-08 (Median) 1.30E-06 (Mean)
Limerick 1 and 2	1.13E-04	2.30E-04 (> F1)	9.00E-09	<E-8 (w/o rec.)	9.00E-09 (w/o rec.)
Millstone 3	1.87E-04	1.87E-04	Low	<E-7	<E-7
Oconee 3	2.50E-04	3.50E-03 (1 mi. radius) 1.20E-03 (2000' radius)	Low	<E-9	<E-9
Seabrook 1 and 2	1.26E-03	7.77E-05	<3.89 E-8	2.06E-09 (LOSP & RWST) 3.40E-10 (LOSP & EDGs)	2.06E-09
Zion 1 and 2	1.00E-03	1.00E-03	N.A.	<E-8	N.A.

Table 4.1 (continued)

Plant Name	Tornado Frequency (Any Size) (/mi^2-yr.)	Tornado Strike Frequency (Any Size) (/yr.)	High Wind Core Damage Frequency (/yr.)	Tornado Core Damage Frequency (/yr.)	Total High Wind/ Tornado Core Damage Frequency (/yr.)
Generic Safety Issue A-45 PRAs					
Arkansas Nuclear One-1	5.18E-04 (Regional)	1.53E-03	5.69E-06 (w/o rec.)	2.53E-04 (w/o rec.)	2.59E-04 (w/o rec.)
	4.37E-04 (Local)		1.16E-07 (w/ rec.)	5.19E-06 (w/ rec.)	5.31E-06 (w/ rec.)
Point Beach 1 and 2	6.98E-04 (Regional)	5.38E-04	1.00E-05 (w/o rec.)	5.00E-05 (w/o rec.)	6.00E-05 (w/o rec.)
	4.11E-04 (Local)		6.60E-07 (w/ rec.)	3.30E-06 (w/ rec.)	3.96E-06 (w/ rec.)
Quad Cities 1 and 2	5.18E-04 (Regional)	1.04E-03	<<E-8 (w/o rec.)	5.08E-07 (w/o rec.)	5.08E-07 (w/o rec.)
	5.44E-04 (Local)		<<E-8 (w/ rec.)	1.35E-07 (w/ rec.)	1.35E-07 (w/ rec.)
St. Lucie 1	6.98E-04 (Regional)	1.70E-04	<<E-8 (w/o rec.)	1.61E-08 (w/o rec.)	1.61E-08 (w/o rec.)
	1.20E-03 (Local)		<<E-8 (w/ rec.)	<E-09 (w/ rec.)	<E-09 (w/ rec.)
Turkey Point 3	3.37E-04 (Regional)	1.70E-04	3.30E-05 (w/o rec.)	2.54E-06 (w/o rec.)	3.55E-05 (w/o rec.)
	5.83E-03 (Local)		2.25E-05 (w/ rec.)	1.73E-06 (w/ rec.)	2.42E-05 (w/ rec.)

N.A. = information not available at time of table preparation.
w/o rec. = without recovery of offsite power.
w/rec. = with recovery of offsite power.

Table 4.2

Vulnerable & Noteworthy Structures from High Wind/Tornado PRAs
[Ref. 4.8 - 4.18]

Plant Name	Vulnerable Structures TM = Tornado Missiles W = High Winds	Noteworthy Structures	Affected Structures/ Components
Indian Point 2	Unit 2 Control Bldg. (TM) Unit 2 DG Bldg. (TM) Unit 2 Prim. Aux. Bldg (TM) Units 2 & 3 AFW Bldg. (TM) Cond. Stor. Tank (TM & W) RWST (TM & W) City Water Tank (TM & W) Serv. Water Pumps (TM)	Unit 1 Superheater Stack	Unit 2 DG Bldg. Unit 2 Control Bldg.
Limerick 1 and 2	Turbine Enclosure (W) CST (W) Cooling Towers (W) Electrical Transformers & Substations (W)	None	Not Applicable
Oconee 3	Unit 3 Aux. Bldg. (TM) Control Room Unit 3 portion of Main Turbine Hall (TM) Aux. Shutdown Panel Emer. FW Pumps Main FW Pumps Upper Surge Tank BWST (TM) Standby Shutdown Facility (TM)	None	Not Applicable
Seabrook 1 and 2	Unit 1 Prim. Aux. Bldg. Unit 2 Prim. Aux. Bldg. Unit 1 DG Bldg. Unit 2 DG Bldg. Unit 1 Control Bldg. Unit 2 Control Bldg.	None	Not Applicable

Table 4.2 (continued)

Plant Name	Vulnerable Structures TM = Tornado Missiles W = High Winds	Noteworthy Structures	Affected Structures/ Components
Generic Safety Issue A-45 PRAs			
Arkansas Nuclear One-1	CST (TM) BWST (TM) DG Fuel Oil Storage Tank (TM) New CST (TM) DG Exhaust Stack (TM & W)	DG Exhaust Stack	Fails both DGs
Point Beach 1 and 2	Service Water Pumphouse (TM) Control Room (TM) DG Exhaust Stack (TM & W) CSTs (TM & W) RWSTs (TM & W)	DG Exhaust Stack	Fails both DGs
Quad Cities 1 and 2	Swing DG Bldg (TM & W) Unit 1 DG Room (TM & W) Unit 2 DG Room (TM & W) 4 kV, 480 V Switchgear Area (TM, Stack Fail.) Unit 2 Battery Room (TM, Stack Fail) CSTs (TM & W)	310' Concrete Stack	4 kV, 480 V Switchgear Area Unit 2 Bat. Rm.
St. Lucie 1	CCW Pipes (TM) DG Fuel Oil Storage Tanks (TM) RWST (TM) Intake Cooling Water Pipes (TM)	None	Not Applicable
Turkey Point 3	DG Bldg. (TM, W, Stack Fail.) DG Fuel Oil Transfer Pumps (TM, Stack Fail.) Switchgear Bldg. (TM, Stack Fail.) Control Room (TM) Units 3 RWST (TM, Stack Fail.) Unit 4 RWST (TM) DG Fuel Oil Storage Tank (TM, Stack Fail.) CST (TM, Stack Fail.) Intake Pumps (TM, Stack Fail.) Startup AFW Pumps (TM)	Unit 2 400' Concrete Stack	DG Bldg. DG Fuel Oil Transfer Pumps Switchgear Bldg. Unit 3 RWST DG Fuel Oil Storage Tank CST Intake Pumps

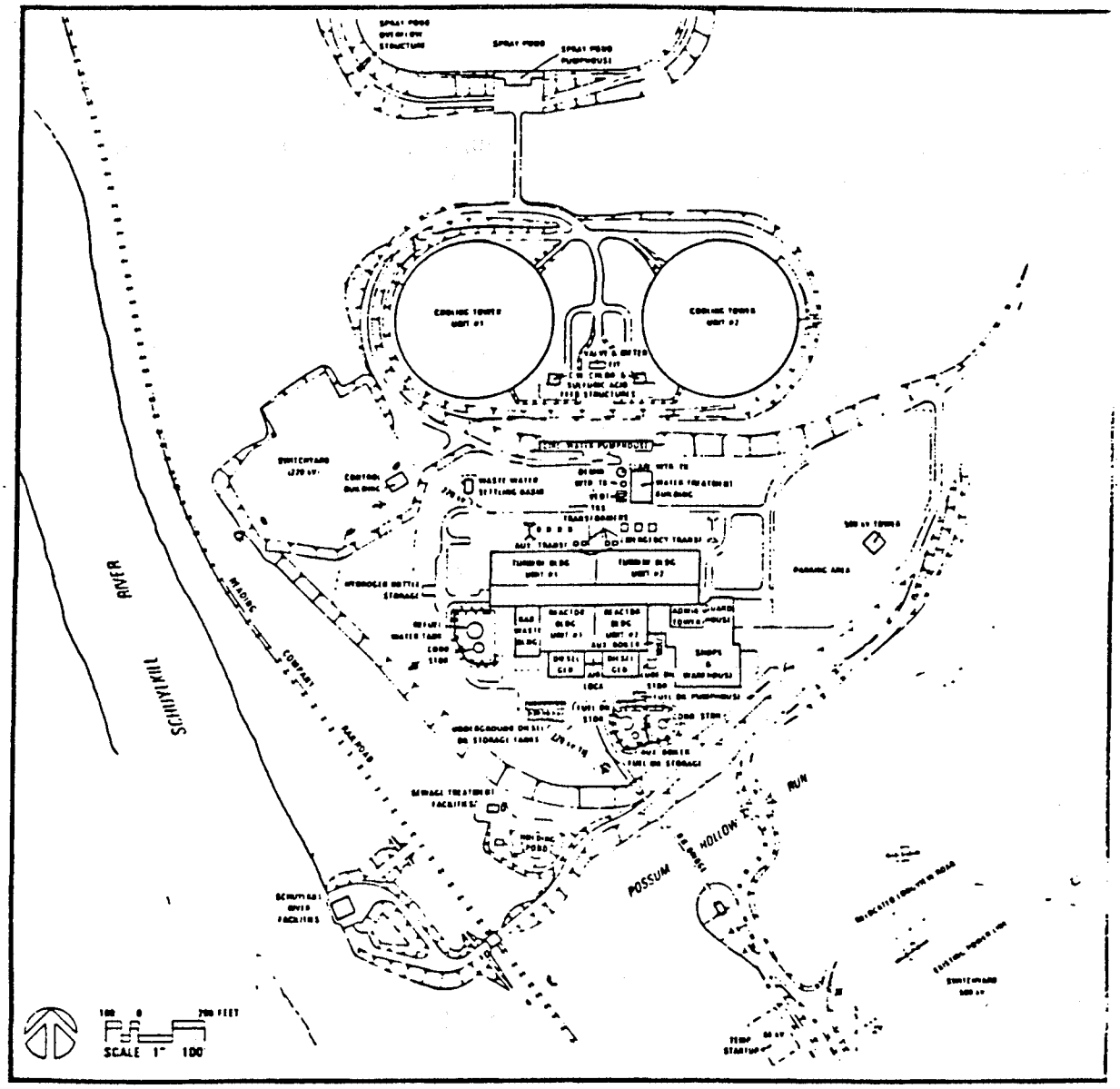
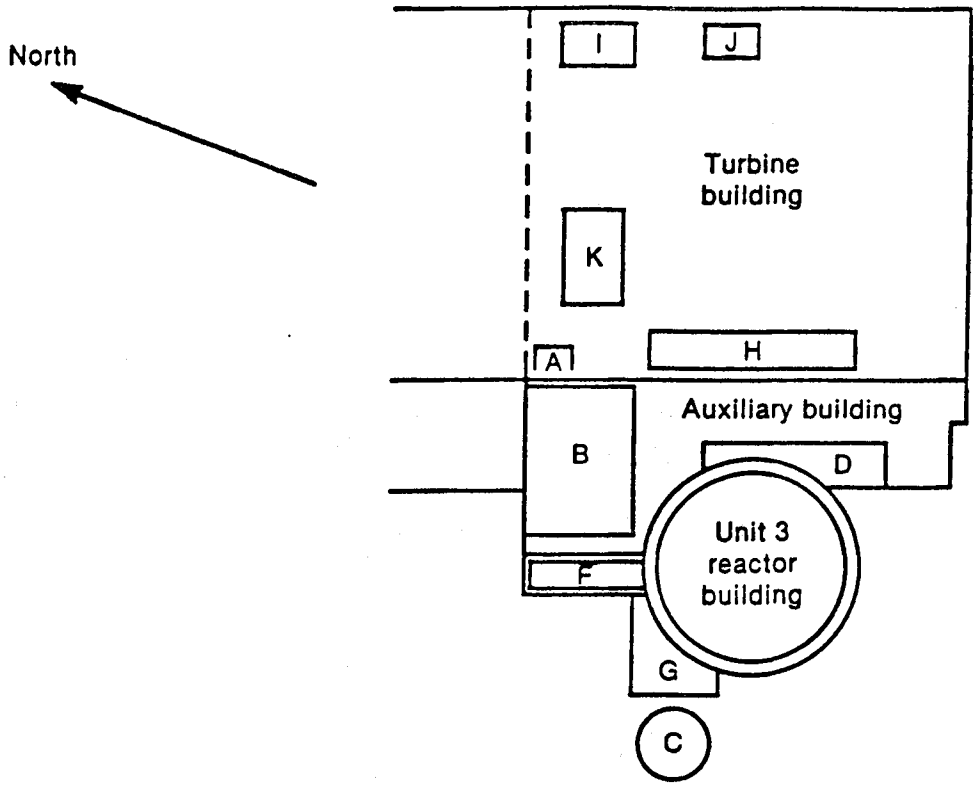
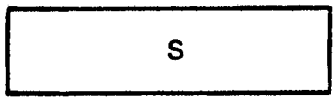


Figure 4.4

Limerick Generating Station SIte [Ref. 4.9]





A Auxiliary shutdown panel
B Control room
C Borated-water storage tank (BWST)
D East penetration area
G West penetration area
S Standby shutdown facility
F Spent-fuel storage (water intake not shown)
H Upper surge tank
I Main-feedwater pumps
J Emergency-feedwater pumps
K 4kv switchgear

Figure 4.5

Oconee Unit 3 Site and Critical Areas for Tornado Missiles [Ref. 4.11]

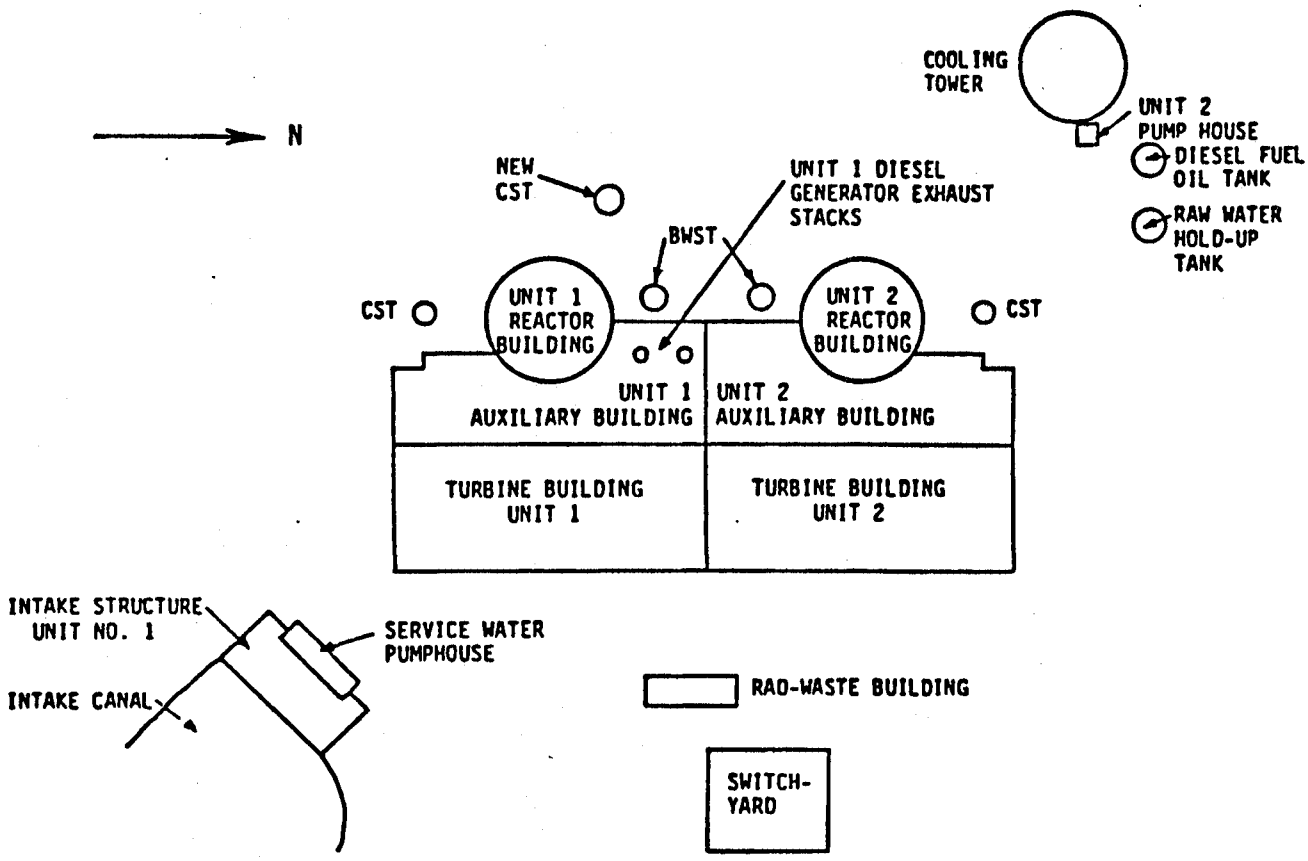


Figure 4.6

Arkansas Nuclear One Site [Ref. 4.14]

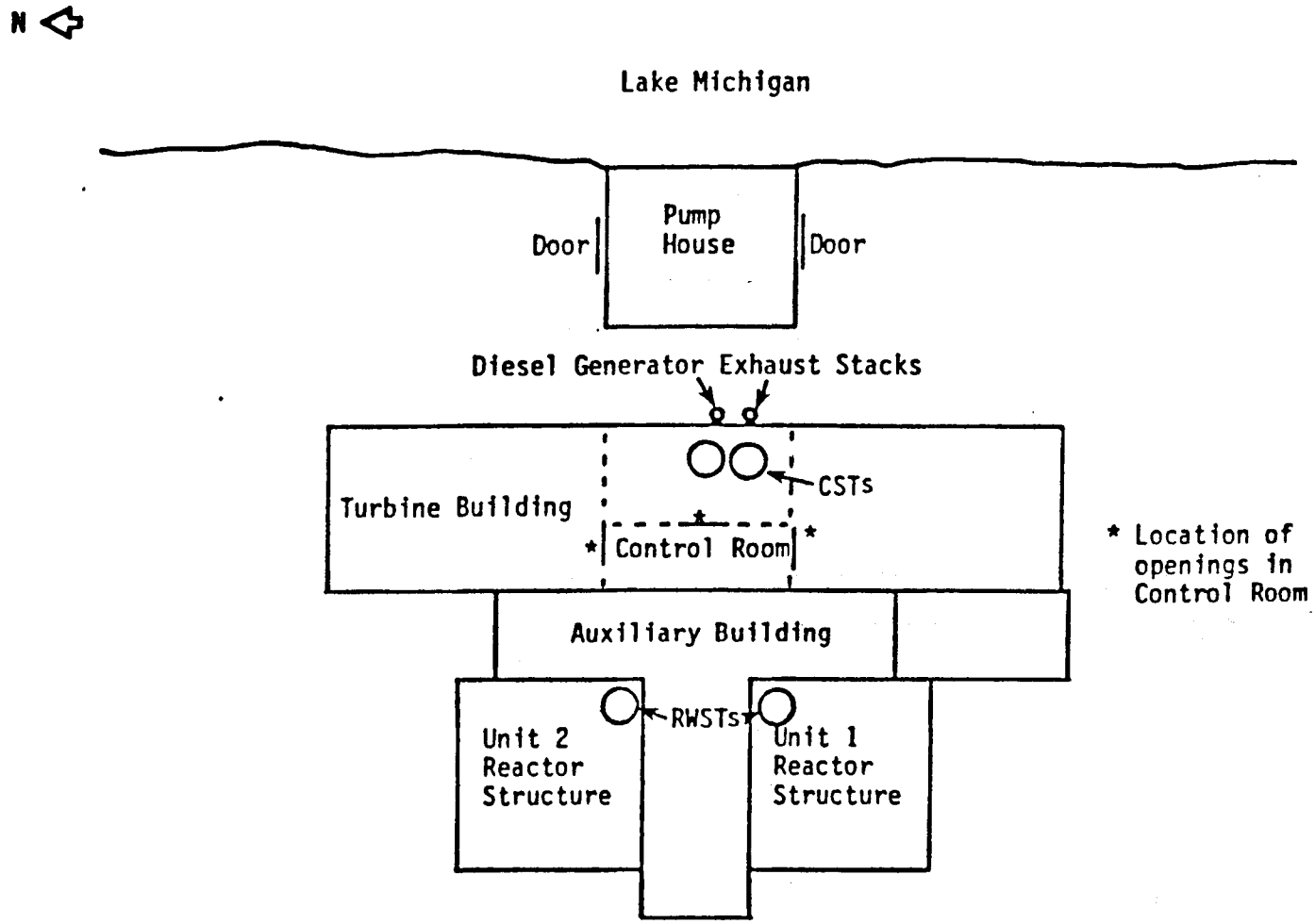


Figure 4.7

Point Beach Site [Ref. 4.15]

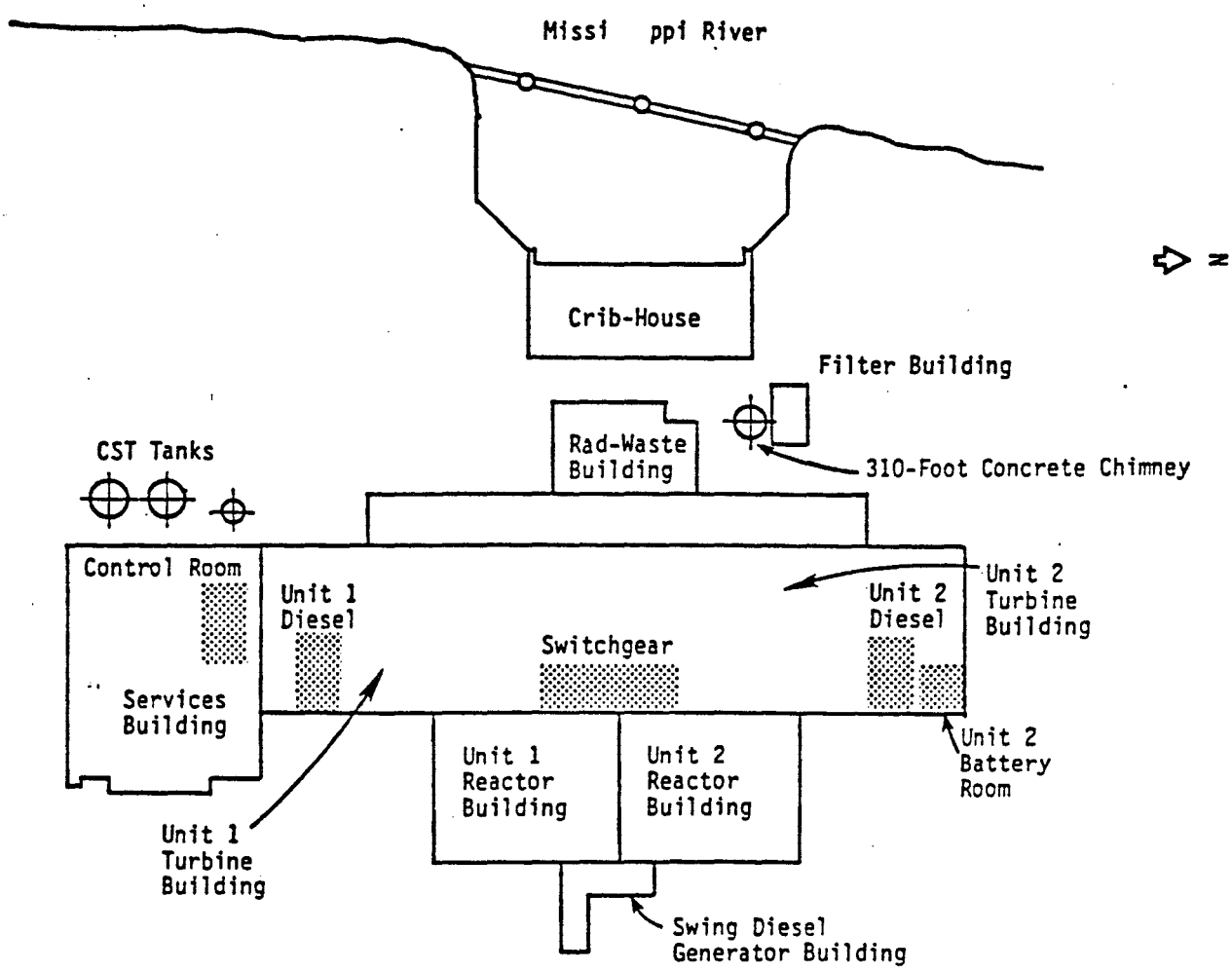


Figure 4.8

Quad Cities Site [Ref. 4.16]

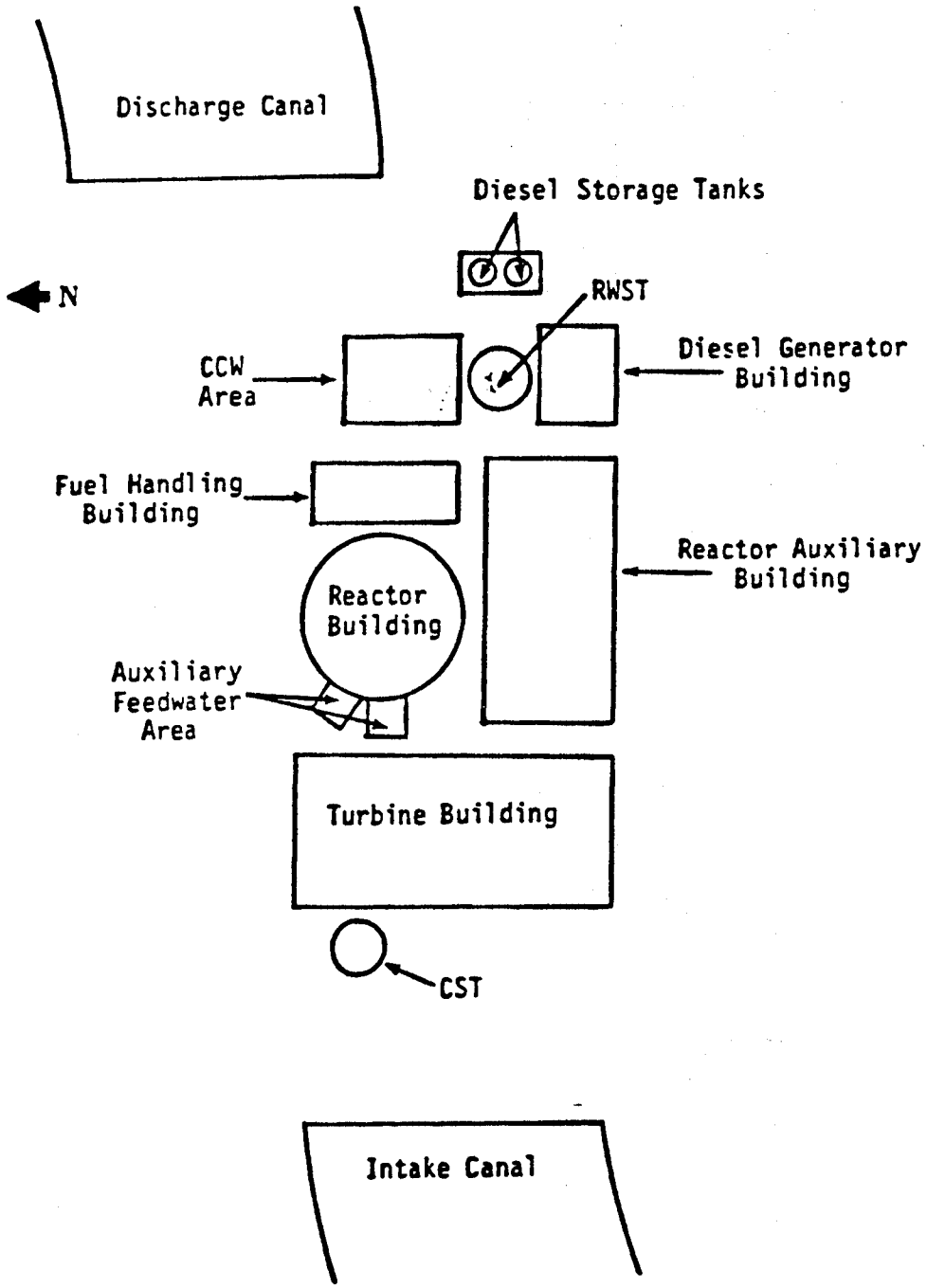


Figure 4.9

St. Lucie Unit 1 Site [Ref. 4.17]

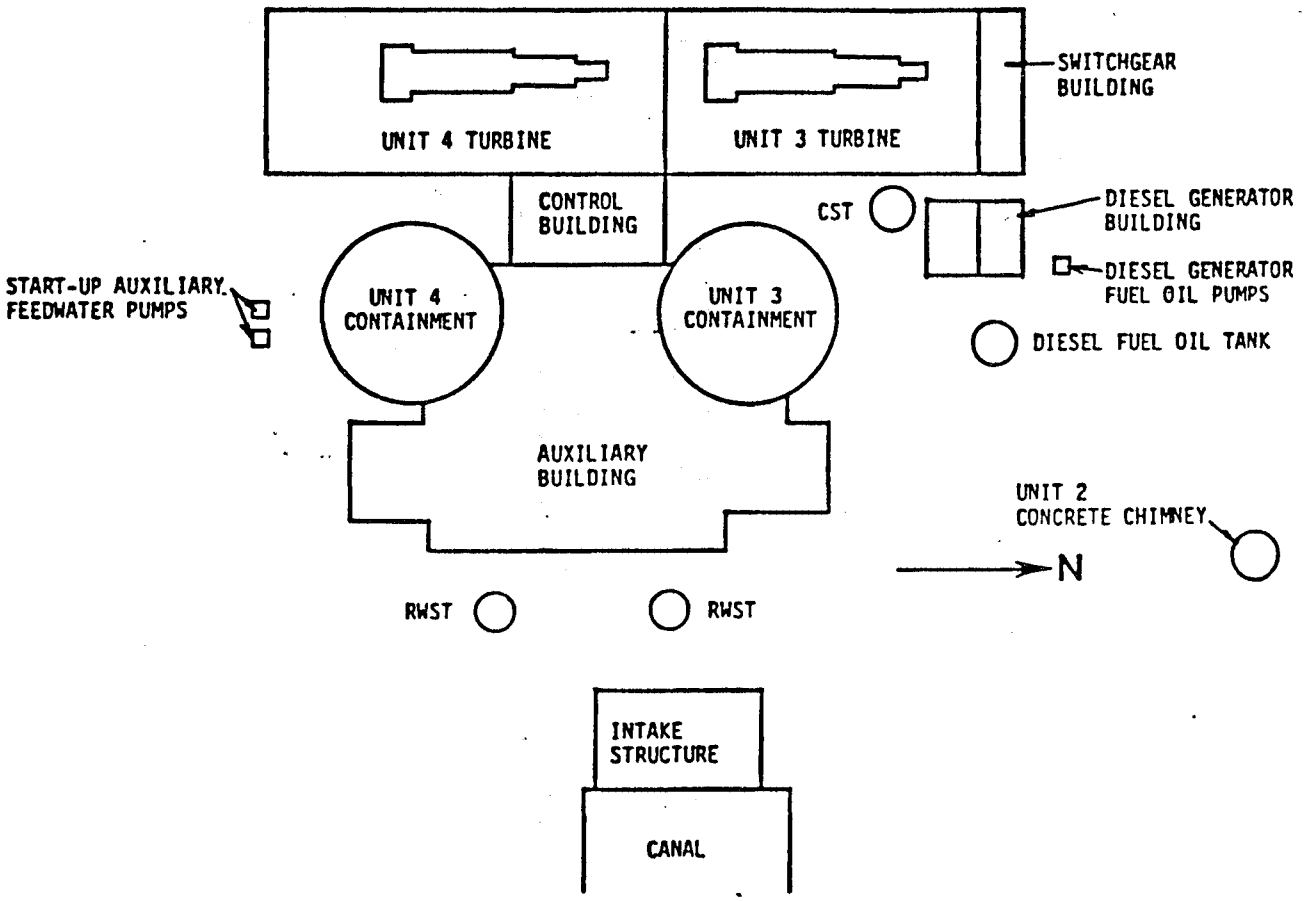


Figure 4.10

Turkey Point Site [Ref. 4.18]

4.5 GENERAL OBSERVATIONS OF HIGH WIND/TORNADO PRAS

Before proceeding with the discussion of the general observations of the high wind/tornado PRAs of the previous section, the following assumptions should be noted. The first assumption is that all of the high wind/tornado PRAs were done using a common methodology and that their results can be compared on a common basis. As can be inferred from the discussion of the individual PRAs of the previous section, this is not strictly true. Some PRAs were merely scoping studies. Others went into much more depth and detail. Only the five A-45 PRAs were done on a strictly comparable basis. The second assumption is that the methodology used in the high wind/tornado PRAs represents the best core damage frequency estimate currently available. With this in mind, the following observations are presented.

The site specific tornado frequency, while important in determining the risk of core damage due to high winds/tornadoes is not the predominant factor. A plant with a high tornado frequency may still have a low risk if the plant design, tornado strike frequency, and other factors reduce the likelihood of core damage from high winds/tornadoes. Caution must be exercised to allow for reasonable uncertainties for each factor used in the core damage and risk calculation levels.

Offsite power sources were either assumed lost or had a relatively high probability of failure. Even if the plant was designed to preserve its connection up to the station transformer yard(s), offsite connections are usually not designed up to the same standards and therefore are considered lost at an early point of the analysis.

Outdoor tanks, such as the Refueling Water Storage Tank (RWST), Condensate Storage Tank (CST), diesel fuel oil storage tanks, etc., that are not protected against tornado missile by barriers tend to be very important in the plant's response to high wind/tornado damage, particularly if the tank can act as a common mode fault which may affect more than one train of safety-related equipment. An example of these common mode fault sources is the RWST whose failure could fail all of the safety injection systems except for the primary accumulators. This failure combined with loss of offsite power could lead to a high probability of core damage. Even with tornado missile barrier protection, outdoor tanks could be vulnerable to tornado damage if the tank is partially filled. This vulnerability is due to the pressure differential of the tornado causing the tank to collapse, possibly failing the tank and making it unavailable as a source of liquid supply.

The controlling high wind or tornado wind speed might be determined by structures that are not built to the same safety standards as the plant itself and whose failure could compromise plant structures or plant safety-related equipment. An example of this is the concrete stack for Turkey Point Unit 2 (an oil-fired plant) whose failure could affect Unit 3 (the nuclear unit).

4.5.1 Recommendations for Plant-Specific High Wind/Tornado Risk Analysis

A full-scope wind PRA is one acceptable approach to analyzing the effect of high winds/tornadoes on a given nuclear power plant. However, it is not necessarily the only approach. It may be feasible to screen out wind/tornado hazards based on either a sufficiently low high wind/tornado frequency or the absence of structures vulnerable to more modest wind/tornado velocities.

A suggested screening approach which could be used to determine if the wind/tornado hazard at a particular site could be screened out or if additional analysis such as a full-scope wind PRA would be necessary, is given below:

1. Assuming that structures constructed to NRC licensing standards will resist high wind/tornado forces, missiles and pressures to at least the design basis, the first task is to identify site-specific structure(s) that may not be built to the same safety standards as the plant itself and whose failure could compromise plant structures or plant safety-related equipment. This identification could start by reviewing the licensing documents for the plant. However, the licensing documents review may not be sufficient to identify these structures. A plant walkdown may be desirable to provide this assurance. One must assure that, for example, a concrete stack located adjacent to the plant and built to non-safety standards does not exist at the site. Particular attention should be paid to tall structures on or near the plant site.

2. Determine the local site-specific tornado frequency, taking into account local geography, local meteorological conditions, typical yearly variations, etc. If the local tornado frequency can be shown with high confidence to be on the same order of magnitude as the first figure-of-merit for core damage, then it would be reasonable to end the analysis at this point.

3. If the local tornado frequency cannot be shown with high confidence to be on the same order of magnitude as the first figure-of-merit for core damage, then it is necessary to determine the tornado strike frequency on the power plant site. An aerial target strike model is the preferred method for determining the tornado strike frequency but if it can be shown that there are no vulnerable structures on or near the plant site with dimensions on the order of the tornado damage width dimensions, then a point target model may be sufficient. If the local tornado strike frequency can be shown with high confidence to be on the same order of magnitude as the first figure-of-merit for core damage, then it would be reasonable to end the analysis at this point.

4. If it cannot be shown that the local tornado strike frequency is comparable to the first figure-of-merit for core damage, then it is necessary to perform some plant specific calculations on the plant's response to damage caused from high winds and/or tornadoes. These calculations could take the form of component-specific fragility curves from high winds or tornado

missiles, calculation of the protection provided by tornado missile barriers, tornado missile strike probabilities, etc. In any systems analyses of wind-induced failures, it is probably necessary to assume the loss of offsite power or assign a high probability to the loss of offsite power. It may not be necessary to perform full-scale event tree and fault tree analysis of the plant's response to the damage caused from high winds and/or tornadoes if the previous calculations can show with high confidence that the core melt frequency is comparable to the first figure-of-merit. Also, if plant systems models have been developed in conjunction with other risk studies, they can probably be adapted to allow analysis of potential plant vulnerabilities due to high wind/tornado events.

4.5.2 FUTURE AREAS OF RESEARCH NEEDED

Additional research is needed in the area of determining local tornado frequency. The efficiency of tornado record-keeping is improving since tornado record-keeping began 30 or so years ago. As additional tornadoes are recorded over time, the uncertainty of local tornado frequency should diminish. However, tornado occurrence is still influenced by the local population density. If eyewitnesses are not present when a tornado touches the ground, it may not be recorded unless it causes substantial damage which lasts long enough for the tornado causing the damage to be recorded. It maybe possible to record tornado occurrences based on radar signatures associated with violent thunderstorms although literature in this area has not been researched.

The ability to analyze tornado strike frequency is improving with time. The effect of each tornado recorded over time is providing additional data for predicting the tornado damage area, path width and path length. New models are being developed which take into account this new information and which are reducing the uncertainty of the estimates of tornado strike frequency.

An area which needs improvement but not an area in which additional research may be required is the definition of tornadoes and their damage intensity rating. The definition of what constitutes a tornado lacks precision. There are dustdevils, waterspouts, microbursts, etc., that should probably be excluded from being classified as tornadoes but their classification should be done on a less subjective basis than is being done presently. Additionally, the damage done by tornadoes on the Fujita Intensity Scale should be defined more precisely so that tornado intensity can be rated on a consistent basis. Both of these areas will require agreement within the high wind/tornado community.

4.6 SUMMARY AND CONCLUSION

High winds and tornadoes must be included among the external initiators considered in the risk analysis of nuclear power plants. Based on the high wind/tornado core damage frequencies presented in Table 4.1, it must be concluded that the risk of core damage due to high wind/tornadoes is comparable to the first figure-of-merit presented in Chapter 1 of this report.

It is possible to separate the probabilistic analysis of the initiating event (high wind/tornado) from the probabilistic analysis of the plant's response to the initiating. That is, the frequency of the high wind/tornado is independent (does not depend) on the plant design. This makes it feasible to estimate the local tornado strike frequency initially and, if low enough, terminate the probabilistic analysis at that point.

4.7 References

[4.1] NUREG/CR-0556 Aerodynamic Considerations on the Injection of Tornado-Generated Missiles from Ground Level into Tornado Flow Fields, R.T. Hall, Naval Surface Weapons Center, Silver Spring, MD. June 1979.

[4.2] Title 10 Code of Federal Regulations, Energy (10 CFR) Parts 0 to 199, Revised as of January 1, 1987, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

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

[4.4] Regulatory Guide 1.76 Design Basis Tornado for Nuclear Power Plants, U.S. Atomic Energy Commission, Directorate of Regulatory Standards, Washington, D.C. April 1974.

[4.5] Regulatory Guide 1.117 Tornado Design Classification, U.S. NRC Office of Standards Development, Revision 1. April 1978.

[4.6] ASCE Paper No. 3269 "Wind Forces on Structures", Transactions of the American Society of Civil Engineers, Vol. 126, Part II. 1961.

[4.7] ANSI A58.1 Building Code Requirements for Minimum Design Loads in Buildings and Other Structures, American National Standards Institute Committee 1972.

[4.8] Indian Point Probabilistic Safety Study, Consolidated Edison Company and Power Authority of State of New York, Pickard, Lowe & Garrick, Inc. 1983.

[4.9] Severe Accident Risk Assessment Limerick Generating Station, Philadelphia Electric Company, NUS Report No. A-4161. 1983.

[4.10] Millstone Unit 3 Probabilistic Safety Study, Section 1, Part 1, Volume 2. August 1983.

[4.11] Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3, the Nuclear Safety Analysis Center, Electric Power Research Institute, Palo Alto, CA and Duke Power Company, Charlotte, N.C., NSAC 60. June 1984.

[4.12] Seabrook Station Probabilistic Safety Assessment, Public Service Company of New Hampshire and Yankee Atomic Electric Company, Pickard, Lowe & Garrick, Inc. Report PLG-0300. 1983.

[4.13] Zion Probabilistic Safety Study, Commonwealth Edison Company, Pickard, Lowe & Garrick, Inc. 1982.

[4.14] NUREG/CR-4713, SAND86-1832 Shutdown Decay Heat Removal Analysis of a Babcock and Wilcox Pressurized Water Reactor, W.R. Cramond, D.M. Ericson, Jr., and G.A. Sanders, Sandia National Laboratories, Albuquerque, NM. March 1987,

[4.15] NUREG/CR-4458, SAND86-2496 Shutdown Decay Heat Removal Analysis of a Westinghouse 2-Loop Pressurized Water Reactor, W.R. Cramond, D.M. Ericson, Jr., and G.A. Sanders, Sandia National Laboratories, Albuquerque, NM. March 1987.

[4.16] NUREG/CR-4448, SAND85-2373 Shutdown Decay Heat Removal Analysis of a General Electric BWR3/Mark I, S.W. Hatch, D.M. Ericson, Jr., and G.A. Sanders, Sandia National Laboratories, Albuquerque, NM. March 1987.

[4.17] NUREG/CR-4710 Shutdown Decay Heat Removal Analysis of a Combustion Engineering 2-Loop Pressurized Water Reactor, G.A. Sanders, D.M. Ericson, Jr., and W.R. Cramond, Sandia National Laboratories, Albuquerque, NM. 1987.

[4.18] NUREG/CR-4762, SAND86-2377 Shutdown Decay Heat Removal Analysis of a Westinghouse 3-Loop pressurized Water Reactor, G.A. Sanders, D.M. Ericson, Jr., and W.R. Cramond, Sandia National Laboratories, Albuquerque, NM. March 1987.

[4.19] UCRL-53526 Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazard Models for Department of Energy Sites, D.W. Coates, and R.C. Murray, Lawrence Livermore National Laboratory, Livermore, CA, Rev. 1. August 1985.

[4.20] NUREG/CR-3058 A Methodology for Tornado Hazard Probability Assessment, J.R. McDonald, Institute for Disaster Research, Texas Tech University, Lubbock, TX. October 1983.

[4.21] NUREG/CR-3670, PNL-5006 Violent Tornado Climatography, 1880-1982, T.P. Grazulis, Environmental Films Inc., St. Johnsbury, VT, May 1984.

[4.22] NUREG/CR-4461, PNL-5697 Tornado Climatology of the Contiguous United States, J.V. Ramsdell, G.L. Andrews, Pacific Northwest Laboratory, Richland, WA. March 1986.

[4.23] Callaway Plant Units No. 1 & 2 Addendum Standardized Nuclear Unit Power Plant System (SNUPPS) Final Safety Analysis Report, Union Electric Company (No. Amendment No.). (No Date).

4.A APPENDICES

4.A.1 TORNADO EVENT MODEL

The evaluation of the hazard to nuclear power plants from tornadoes requires a classification of the plant damage states possible and an estimate of the site-specific frequency of occurrences of tornadoes capable of producing those plant damage states.

The phenomenology of tornadoes is not yet well understood although much has been written describing tornadoes and the damage that they cause. The main difficulty in understanding and modeling the phenomenology of tornadoes is measuring the physical characteristics of tornadoes, i.e. wind velocity, pressure drop, etc. There are two reasons why it is so difficult to measure the physical characteristics of tornadoes. The first reason is the physical threat to whoever or whatever is measuring the tornadoes. Automatic instruments and vehicles carrying instruments are destroyed or badly damaged by most tornadoes. The second reason is the unpredictable nature of the tornado itself. Unless entire regions are instrumented, the appearance of a tornado may be missed. Radar tracking of severe thunderstorm areas is improving tornado prediction and measurement but at the present time, tornado characteristics are based on the damage they have caused.

4.A.2 TORNADO INTENSITY SCALE

The plant damage states caused by tornadoes can be related to the tornado intensity by the Fujita Scale, which is the standard method of relating tornado intensity as expressed as maximum tornado wind speed to observed damage [4.19 - 4.20]. The most intense tornadoes have the highest wind speed, hence cause the most damage. The Fujita Scale has seven intensity classifications ranging from F0 to F6. Their definitions are listed below:

F0 Light Damage
Corresponds to Beaufort Scale 9 through 11. Some damage to chimneys or TV antennae occurs; branches broken off trees; shallow-rooted trees pushed over; old trees with hollow insides break or fall; sign boards are damaged.

F1 Moderate Damage
Beginning of hurricane wind speed or Beaufort Scale 12 at 73 mph. Roof surfaces peeled off; windows broken; trailer houses are pushed or overturned; trees on soft ground uprooted; some trees snapped; moving autos pushed off road.

F2 Considerable Damage
Roofs torn off frame houses leaving strong upright walls standing; weak structures or outbuildings demolished; trailer houses demolished; railroad boxcars pushed over; large trees snapped or uprooted; light-object missiles generated; cars blown off highway; block structures and walls badly damaged.

F3 Severe Damage
Roofs and some walls torn off well-constructed frame houses; some rural buildings completely demolished or flattened; trains overturned; steel frame hanger-warehouse type structures torn; cars lifted off ground, may roll some distance; most trees in forest uprooted, snapped or leveled; block structures often leveled.

F4 Devastating Damage
Well-constructed frame houses leveled, leaving piles of debris; structures with weak foundations lifted, torn, and blown some distance; trees debarked by small flying debris; sandy soil eroded and gravel flies in high winds; cars thrown or rolled considerable distances, finally disintegrating; large missiles generated.

F5 Incredible Damage
Strong frame houses lifted off foundation and carried considerable distances to disintegrate; steel-reinforced concrete badly damaged; auto-sized missiles fly distances of 100 yards or more; trees debarked completely; incredible phenomena can occur.

F6 Inconceivable Damage
Extent and types of damage may not be conceived. Missiles such as iceboxes, water heaters, storage tanks, and automobiles fly long distances, creating serious secondary damage on structures. Assessment of damage is feasible only through detailed surveys involving engineering and aerodynamical calculations, as well as meteorological models of tornadoes.

From the previous definition of the Fujita Tornado Intensity Scale, it is clear that the assignment of F-Scale index level to a tornado is a subjective task rather than being determined by actual physical measurements. Because of this, it is difficult to predict accurately the frequency of tornado occurrence with specified characteristics, i.e. wind speeds greated than the design basis wind speed.

Another measure used to characterize the tornado intensity is the Pearson Path Length, Pl, and Pearson Path Width, Pw, Scales, which indicate the mean length and width of the tornado damage path for damage done by winds greater than or equal to 75 mph [4.19 - 4.20]. Again, the tornado dimensions tend to increase with tornado intensity.

The Fujita Scale and Pearson Scale are used together to classify a tornado's intensity by assigning it a Fujita-Pearson (FPP) number. Table 4.A.1 lists the tornado intensity parameters from the Fujita-Pearson classification scheme.

Table 4.A.1

Fujita-Pearson (FPP) Tornado Classifications [Ref. 4.19 - 4.20]

Index	Fujita Scale Wind Speed (mph)		Pearson Scale Path Length (mi.)		Pearson Scale Path Width (yds.)	
	Range	Median	Range	Median	Range	Median
0	40- 72	56.0	0.3- 0.9	0.6	6- 17	11.5
1	73-112	92.5	1.0- 3.2	2.05	18- 55	36.5
2	113-157	135.0	3.2- 9.9	6.55	56- 175	115.5
3	158-206	182.0	10.0-31.5	20.75	176- 556	366
4	207-260	233.5	31.6-99	65.3	557-1759	1158
5	261-318	289.5	100 -316	208	1760-4963	3361.5

4.A.3 TORNADO FREQUENCY

The second factor needed to estimate the risk to plant from tornadoes is the site-specific frequency of tornado occurrences capable of producing plant damage states which could lead to damage to the reactor core.

The number of tornadoes that occur within a region varies from year to year. Part of the variation is due to normal changes in the local meteorology. Another large part of the variation is due to lack of tornado records. In areas where population is sparse, in remote areas, or in areas of open rangeland or desert, many tornadoes are likely to go unreported. Unless some attempt is made to account for unrecorded tornadoes in the data records, probability assessment is likely to be nonconservative [4.20]. Table 4.A.2 presents the number of recorded tornadoes according to the National Severe Storms Forecast Center (NSSFC) in Kansas City, Missouri [4.20 - 4.21].

From 1950 to 1970, there is a gradual increase in the number of reported tornadoes per year. While there is some variation during the 1970's, the increasing trend is no longer predominant. The gradual increase in the number of tornadoes is believed to be because of more efficient reporting procedures, rather than a change in weather patterns [4.20]. Because the more violent and intense tornadoes cause more damage over a wider area than the less intense tornadoes, fewer violent tornadoes tend to be missed in tornado event tabulations. Figure 4. reproduced from Reference 4.22, shows this trend. The number of tornadoes in the F3 to F5 category has not changed greatly from 1916 to 1977. Conversely, the number of tornadoes in the F0 to F2 category and the total number of tornadoes has increased dramatically from 1950 onwards.

Table 4.A.2

No. of Tornadoes by Year

Year	F-Scale F0, F1, F3, F3 (Calculated)	F-Scale F4, F5 [Ref. 4.21]	Total [Ref. 4.10]
1950	216	7	223
1951	264	5	269
1952	255	17	272
1953	467	22	489
1954	602	7	609
1955	620	10	630
1956	556	13	569
1957	904	26	930
1958	603	5	608
1959	623	7	630
1960	638	7	645
1961	767	6	773
1962	667	6	673
1963	488	6	494
1964	747	12	759
1965	968	31	999
1966	598	7	605
1967	951	17	968
1968	703	12	715
1969	644	7	651
1970	692	9	701
1971	953	11	964
1972	770	4	774
1973	1,185	14	1,199
1974	1,086	37	1,123
1975	950	11	961
1976	920	16	936
1977	911	10	921
1978	867	5	872
1979	906	7	913
1980	N.A.	5	N.A.
Total	21,521	359	21,875
Average	717.37	11.58	729.17
Std. Dev.	234.18	7.85	237.39

N.A. = information not available at time of table preparation.

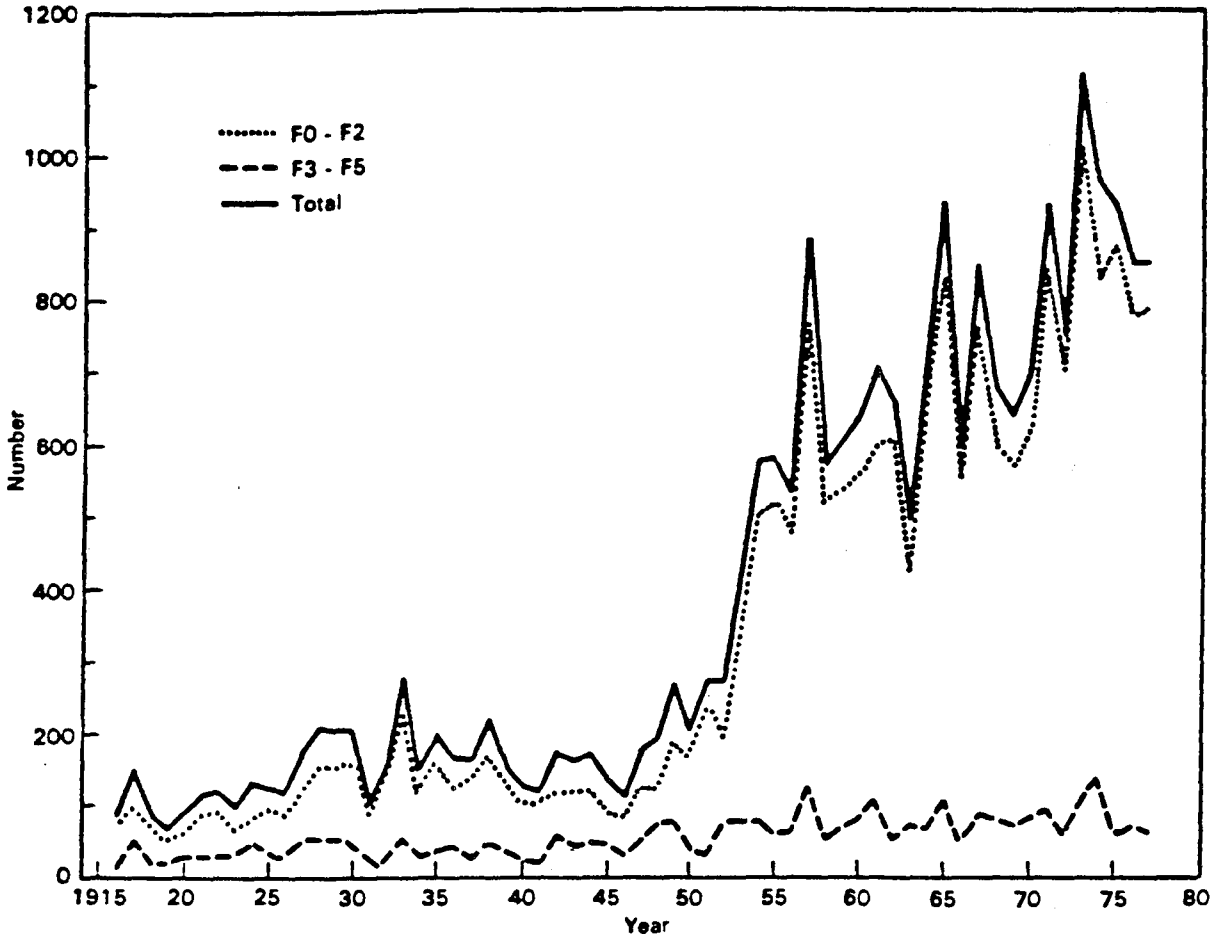


Figure 4.A.1

Number of Tornadoes Reported Each Year, 1916 - 1977. Based on the Data Presented by Fujita (1978). [Reference 4.22]

4.A.4 TORNADO INTENSITY FREQUENCY

Tornadoes are born at all levels of the F-Scale intensity scale. Therefore, one of the factors needed to estimate the probability of the tornado design wind speed occurring at a specific site, is the F-scale intensity frequency. Table 4.A.3 presents the tornado intensity frequency from the NSSFC for 1954 through 1983 [4.22]. A problem with this information is that there are a number of tornadoes in which sufficient information could not be found from recorded data (row denoted "missing") to determine a F-Scale intensity. One possible solution is to assign the same intensity frequency as the recorded tornadoes to the missing tornadoes. This is probably a conservative assumption in that the more intense and violent tornadoes tend to cause more damage and should have more information recorded about them. This is also shown in Table 4.A.3.

Table 4.A.3

Tornado Intensity Distribution

F-Scale Index	Number 1954-1983 [Ref. 4.22]	Percentage 1954-1983 (Calculated) (Missing excluded)
0	5,170	25.35%
1	8,368	41.03%
2	5,292	25.95%
3	1,281	6.28%
4	258	1.27%
5	24	0.12%
Subtotal	20,393	100.00%
Missing	1,969	
Total	22,362	

4.A.5 TORNADO INTENSITY AREA DAMAGE FREQUENCY

A tornado rated by the Fujita F-scale is based on the highest level of damage which occurs within the total tornado damage area. In reality, only a portion of the area affected by a tornado rated, say F-4, actually sustains damage at the F-4 level. The rest of the area affected by this F-4 tornado will sustain damage at a F-3 level or less. Various models have been developed to estimate the area damage frequency by tornado intensity. One model developed by McDonald, based on the Super Outbreak Tornadoes of April 3-4, 1974, is presented in Table 4.A.4 [Ref. 4.20].

Two other models developed by Reinhold and used in the NRC's program for the resolution of generic safety issue A-45 (decay heat removal) [4.14 - 4.18] are presented in Tables 4.A.5 and 4.A.6. The Reinhold 1 matrix is based on a variation of intensity along the tornado length using the percentage of damaged length for each tornado analyzed along with an empirically derived damage width relationship. The Reinhold 2 matrix is based on a variation of intensity along the tornado length based on dividing the tornadoes into 2 to 3 mile segments and assigning the highest intensity observed in each segment as the intensity of the whole segment. The damage width relationship for the second matrix is based on a theoretical wind model with a tornado translation speed of 59 mph.

Table 4.A.4

McDonald Damage Area Distribution [Ref. 4.20]

Maximum F-Scale Index	Actual F-Scale Damage Index					
	0	1	2	3	4	5
0	100.00%	-	-	-	-	-
1	76.05%	23.95%	-	-	-	-
2	57.44%	27.15%	15.41%	-	-	-
3	51.36%	24.53%	14.88%	9.23%	-	-
4	52.27%	22.45%	14.03%	8.43%	2.83%	-
5	52.21%	20.75%	12.21%	8.48%	4.80%	1.55%

Table 4.A.5

Reinhold Matrix 1, Damage Area Distribution [Refs. 4.14 - 4.18]

Maximum F-Scale Index	Actual F-Scale Damage Index						
	0	1	2	3	4	5	6
0	100.0%	-	-	-	-	-	-
1	74.3%	25.7%	-	-	-	-	-
2	65.8%	24.8%	9.4%	-	-	-	-
3	61.5%	26.7%	9.1%	2.7%	-	-	-
4	63.7%	23.4%	9.3%	2.8%	0.8%	-	-
5	63.2%	23.6%	8.8%	3.3%	0.9%	0.2%	-
6	62.5%	23.8%	8.9%	3.3%	1.1%	0.3%	0.1%

Table 4.A.6

Rheinhold Matrix 2, Damage Area Distribution [Ref. 4.14 - 4.18]

Maximum F-Scale Index	Actual F-Scale Damage Index						
	0	1	2	3	4	5	6
0	100.0%	-	-	-	-	-	-
1	75.1%	24.9%	-	-	-	-	-
2	48.4%	41.4%	10.2%	-	-	-	-
3	31.6%	42.3%	20.5%	5.6%	-	-	-
4	31.5%	37.1%	19.9%	8.9%	2.6%	-	-
5	29.3%	33.0%	20.1%	11.2%	5.0%	1.4%	-
6	28.9%	32.6%	17.6%	10.7%	6.3%	3.0%	0.9%

4.A.6 TORNADO STRIKE MODELS

Once the tornado frequency in the local region around the specific site of interest has been determined, then the probability of the tornado actually hitting the site has to be determined. There are basically two class of tornado strike models that are used to determine the tornado strike probability. The first class considers the site to be a point target. This class of models is independent of structure size and orientation and is convenient for determining regional tornado hazard probabilities. The second class of models, called the aerial target concept, considers the size and orientation of the structure or facility in calculating the strike probability. It has been shown that neglecting structure size becomes nonconservative as the size of the structure increases [4.11]. This is because tornadoes can affect areas outside the actual area of destruction caused by the tornado due to missiles being ejected from the damage area. The nonconservatism of the point target models must be considered when selecting a tornado strike model.

CHAPTER 5 --- EXTERNAL FLOODS

5.1 OBJECTIVES

The work reported here is a review and evaluation of what is known about the risk of core-damage accidents and of the potential for large radiological releases due to external floods at U.S. nuclear power plants. This review and evaluation have been done to accomplish the objectives discussed in Chapter 1.

To summarize, the broad objective is to understand, for U.S. nuclear power reactors, whether or not external floods can be among the major accident initiators that may pose a threat of severe core damage or of a large radioactive release. Due to limited resources in this project, this broad objective cannot be addressed for all U.S. plants, so the project scope has been limited to examining a few specific plants whose potential for flood-initiated accidents has been studied in greater detail. From these few studies, some generic insights have been identified. Some of the analysis has used probabilistic methods and some has used more deterministic approaches.

The scope of this chapter does not include flood-driven or storm-driven barges or ships that could collide with safety-related equipment at a reactor site. This type of event is considered a transportation accident, and is covered in Chapter 6 of this report.

The evaluation criteria, as outlined in Chapter 1, involve two different figures-of-merit (FOM): (1). The first FOM is met if individual plants have mean core-damage frequencies in the range of about one part in 100,000 per reactor year. This is not a firm numerical objective, but a range. (2) The second FOM is met if a "large release" is calculated to occur with a mean frequency less than 1 in 1,000,000 per reactor year. Each figure-of-merit is discussed in more detail in Chapter 1.

Chapter 1 also discusses the way these two figures-of-merit will be used. Briefly, Probabilistic Risk Assessment (PRA) methods and results are the main approach to analyzing and calculating plant performance for comparison with the two figures-of-merit.

The scope here will include all phenomena leading to external flooding, in which the source of the water that threatens plant structures and equipment is outside the plant. Internal flooding, in which the source of flood water is within the plant, is explicitly excluded from consideration here.

5.2 TECHNICAL MATERIAL THAT HAS BEEN REVIEWED

The technical material that has been reviewed in this project consists of the following:

1) the NRC's regulatory approach to assuring that reactors area adequately protected against external floods:

- the General Design Criteria, in particular GDC 2;
- 10 CFR 100, in particular Appendix A;
- the Standard Review Plan, in particular Section 2.4; and
- various regulatory guides, particularly including Regulatory Guides 1.27, 1.59 and 1.102;

2) papers and reports concerning external flooding, including discussions of the methodology for determining the design basis flood; methodologies for determining its return period in a probabilistic sense; and studies of how flooding can potentially damage a plant;

3) documentation about how reactors are designed and built by their owners and reviewed by NRC to avoid damage from flooding (selected Safety Analysis Reports (SARs), and Safety Evaluation Reports (SERs));

4) the PRA literature on flooding, including ten PRAs that have studied external floods.

Important aspects of each category above have been examined. To achieve the objectives, the project team has concentrated on the probabilistic literature on flooding. This includes not only the few "PRAs "on actual plants, but also literature on how to determine the probabilities of various types of floods.

5.3 SUMMARY AND EVALUATION OF THE MATERIAL THAT WAS REVIEWED

5.3.1 Current Regulatory Requirements

The basic regulations that are cited in the NRC's regulatory guidance on flooding are General Design Criterion 2 (10 CFR 50 Appendix A), 10 CFR 100, Paragraph 100.10(c) and 10 CFR 100. Appendix A, Section IV (c). These broad regulations are supplemented by more specific guidance in Regulatory Guides 1.27, 1.59 and 1.102. In addition, regulations and guidance on a number of specific issues may be relevant, such as the status of the ultimate heat sink, the service-water system, and the electrical-distribution system.

The approach used by NRC in its regulatory review of external flooding is as follows, in simplified language: the applicant is required first to determine a parameter called the Design Basis Flooding Level (DBFL), and then to assure that critical safety-related components and structures nessary for safe shutdown and maintenance thereof are protected in the unlikely event that the DBFL is exceeded so that site flooding does occur. The methods for determining the DBFL and for designing adequate flood protection for vital

equipment are set down in detail in the regulatory guides.

The DBFL is defined in Regulatory Guide 1.102 as "the maximum water elevation attained by the controlling flood, including coincident wind-generated effects". The detailed methods used to define the DBFL will not be discussed here. Suffice it to say that the process is site-specific and requires consideration of the full range of possible flooding phenomena. Indeed, the DBFL may differ for different parts of a site.

Regulatory Guides 1.59 and 1.102 provide that, if sufficient warning time is available and emergency procedures exist, credit can be taken for early shutdown, in which case it is possible that not all safety-related equipment need be protected against flooding.

The regulatory guidance in Regulatory Guide 1.102 differentiates between two different types of effects of flooding: "structural" effects and "inundation" effects. The "controlling flood" may be different for the two types of effects, since static and dynamic forces may harm structures even if there is no inundation, and vice versa.

The types of flood protection provided will depend, of course, on the site and the local flooding potential from various phenomena. Regulatory Guide 1.102 contains broad guidance on the various barrier types and their functions.

5.3.2 Studying the "Degree of Protection Achieved"

Our approach to understanding the degree of protection that has been achieved by the current plants is to separate the problem into three parts, involving the frequency of the initiating event, the probability of a core-damage accident, and the probability of a large-release accident. We define the following expressions:

F_{CD} = frequency per year of an accident involving core-damage

F_{LR} = frequency per year of an accident involving a large release of radioactivity;

F_F = frequency per year of a flood large enough to cause more than minimal damage to the nuclear power plant;

P_{CD} = probability, given a flood large enough to cause more than minimal damage, that a core-damage accident will occur; P_{CD} is a contingent probability with value between 0 and 1;

P_{LR} = probability, given a core-damage accident from flooding, that the accident will evolve into a "large release" scenario; P_{LR} is a contingent probability with value between 0 and 1.

We observe that, because there has never been a large flood at an operating nuclear power plant, none of these quantities can be determined solely from known empirical information (data, experiments). All must be calculated using models and data that must be extrapolated to the extreme conditions of interest here.

We also observe that F_{CD} and F_{LR} are the two calculated plant parameters that should be compared with the two figures-of-merit discussed above: FOM # 1 involving core-damage accidents and FOM # 2 involving large-release accidents.

We can write, in simplified form, the following expressions for FCD and FLR, where the asterisk (*) is a symbol for numerical multiplication:

$$F_{CD} = F_F * P_{CD}$$

$$F_{LR} = F_F * P_{CD} * P_{LR}$$

We will next discuss what is known about each of these important factors:

5.3.2.1 Frequency of Flooding, F_F:

The factor F_F is obviously the most important one in providing assurance against a reactor accident from external flooding. This is because if the initiating event itself (the large flood) is sufficiently infrequent, then the two figures-of-merit can be satisfied without inquiring into the contingent probabilities of the plant responding safely.

We need to differentiate here among several different types of flood phenomena at different types of sites. The different types of events that could lead to flooding of a nuclear reactor site are generally classified as follows (Regulatory Guide 1.59 and Standard Review Plan Section 2.4.2 contain a more detailed and precise listing):

- all sites: flooding due to severe local precipitation and runoff effects (on the site itself);
- river sites: flooding due to too much water in the river (from precipitation runoff, ice jams, etc.)
- river sites: flooding due to a dam failure (which itself could be due to too much water in the river, an ice jam, etc.);
- ocean, estuarine sites: flooding due to combinations of high tides, wave effects, high wind-driven water levels, surges, seiches, etc.;

- o ocean sites: flooding due to a tsunami;

- o lake sites: flooding due to combinations of high lake water level, wave effects, high wind-driven water levels, surges, seiche, ice jams, etc.;

- o all sites: flooding due to earthquake-induced effects, such as landslides, dam failures, tsunami-type and seiche-type effects.

While phenomena in many of the above categories are mutually exclusive, some of them can occur together so that appropriate combinations of them must be considered in any flooding analysis. In particular, the flooding due to the rare occurrence of an extreme event for one phenomenon may not be as great as the flooding due to less extreme simultaneous occurrences of two or more phenomena. The probabilities of these different types of events can be very different also.

The value of F_F has units of frequency ("events per year"), but since large floods are so rare one often encounters discussions of the "100-year flood", "1000-year flood", and so on. The correct way to think about this terminology is as follows: although the "100-year flood" is popularly thought to be the river flood that will recur every 100 years on the average, the correct logic is that it is the flood level with a 1/100th chance (1% chance) of occurring in any given year. Thus it should be assigned a frequency value of $F_F = 0.01$ per year.

We will next discuss what is known about F_F for the several categories above. As a preview, we offer the following summary observation: the determination of F_F by analysis is fully feasible for return periods in the same range as the historical record (that is, in the range of about 100 years and occasionally longer, corresponding to $F_F = 0.01$/year or perhaps slightly smaller). Extrapolations much beyond that range are difficult and uncertain since they require models and data that must be combined together statistically. The usefulness of these extrapolations varies from one type of flood phenomenon to another.

1) Flooding from Severe Local Precipitation and Runoff: Severe local precipitation can be a cause of flooding by itself, or can accompany other effects to produce a larger flood than any one effect would produce by itself. Here we will consider precipitation by itself. In discussing the other flooding phenomena below in their own sections, it is important to remember that possible large precipitation must be considered in conjunction with these other phenomena.

For severe local precipitation, the method used in NRC regulatory reviews (see Regulatory Guide 1.59) relies on the development of the Probable Maximum Precipitation (PMP) for the specific site. For reactor sites, the PMP has generally been developed by following the guidelines setdown by the National Weather Service [Ref. 5.17, NOAA/NWS, 1956]. The philosophical approach to the PMP is similar to that for the Probable Maximum Flood (PMF, see the next

subsection), and the historical data base is also similar: useful data exist for about 100 years at most sites, but extrapolations to the 1000-year return-period range require a site-specific model, which can be highly uncertain depending on the availability of regional data.

Severe local precipitation can threaten a reactor site (structures, equipment) by itself if it is large enough, or it can pose a threat if it occurs in combination with other severe effects, such as river/lake/ocean flooding, often together with high winds. The analysis of the quantity F_F requires determining how much rainfall is needed to cause certain undesired effects, either structural effects, or inundation effects due to ponding, or see page effects. As a conservative approach, using the PMP as the undesired severe event may be acceptable.

Methods for determining the recurrence of the PMP have been developed by several research groups over the years. The report of the Interagency Advisory Committee [Ref. 5.15 Interagency Committee, 1986] contains a review. One type of approach generates historical (site-specific) data on intense rainfalls over short periods (15- to 60-minute periods, for example), and carries out stochastic (random) analysis to generate severe rainfall likelihoods for much longer periods, such as 8 or 24 hours. Other approaches attempt to study spatial correlations between nearby rain-gauge stations in order to understand time-sequence correlations as an intense rainstorm moves. Since the sizes of extreme rainfalls are limited by the available moisture content in the air mass, some approaches have attempted to introduce these considerations.

The general consensus in this area is as follows: there seems to be no generally accepted methodology for developing reliable values for F_F at recurrences for beyond the historical record for any specific site. The principal difficulty is that storms as large as the PMP can occur only when a number of extremely rare events occur together, and correlations among these rare events are not well understood. In particular, it is not adequate to develop the likelihood of a very large rainfall over, say, four hours from knowledge of 15-minute extreme rainfall data without a sound model of the phenomena. Such a model has not yet been developed, nor its technical basis understood, except for a few specific situations. Thus any attempt to develop F_F values for the most extreme situations will require careful analysis.

2) River Flooding: The design basis river flood generally used for sites on rivers and streams is the Probable Maximum Flood (PMF), defined in numerous publications of the Corps of Engineers such as "EC 1110-2-27,Change 1" [Ref. 5.13, Army Corps of Engineers, 1968], which is cited by reference in Regulatory Guide 1.59. For some of the earlier sites, the Standard Project Flood (SPF) or some other basis was used for the design.

It is important for the analyst to remember that severe local precipitation can add to the flooding level from a severe river flood, and this issue must be accounted for in the flooding analysis.

In the reactor PRA literature, there are a few attempts to calculate values of F_F for specific river sites. However, these few calculations are not the best source of information on F_F, because they are mostly bounding estimates rather than calculations, and because their technical basis is not well explained in the PRAs. It is more instructive to turn to the hydrological literature directly. One recent overview can be found in a 1986 report* by an Interagency Advisory Committee. The Committee formed a Work Group of experts who examined all of the available methodologies that have been developed over the years for determining F_F. The executive summary contains the following key conclusory text:

> "The Work Group's conclusions about defining the probability of the PMF as based on a review of the literature are summarized as follows:
>
> - It is not within the state of the art to calculate the probability of PMF-scale floods within definable confidence or error bounds.
>
> - There is no definable point on the probability scale at which it becomes impossible to define error bounds on flood magnitude and probability estimates. Rather, an analysis displays a gradual transition from commonplace events, whose estimation errors can be defined by statistical random-sampling theory, to unprecedented events, whose errors cannot be defined. Many professionals believe this transition begins at recurrence intervals of about twice the record length and is complete by recurrence intervals in the general area of about 1000 years. The Work Group finds on [sic]** reason to contradict these general perceptions." (page vi)

From the report, one concludes that any claim that a given site-specific PMF is in the range of about the "1000-year flood" would not be accepted broadly in the community of river flooding experts. Using our terminology, values of F_F in the range of 0.01/year might be defensible, but values in the range of 0.001/year are apparently not.

This pessimistic view is not shared by all flooding experts, however. A recent study by the National Academy of Sciences/National Research Council under NRC sponsorship has examined the question from the perspective of which models are most promising for understanding extreme flooding. This committee's report has not been published, but in its meetings and discussions a variety of models have been discussed for extrapolating beyond the historical record, with varying degrees of success (private communication, D.

---

* Work Group on PMF Risk Assessment, under the direction of the Hydrology Subcommittee of the Interagency Advisory Committee on Water Data, "Feasibility of Assigning a Probability to the Probable Maximum Flood", Office of Water Data Coordination (1986). See [Ref. 5.15, Interagency Committee, 1986].

** There is apparently a typographical error in the report here; the correct phrasing is presumably "finds <u>no</u> reason...".

Chery, NRC).

The best summary of the current situation is probably that extrapolations beyond the historical record are difficult except in those few (site-specific) situations where good regional data and a good local site model allow defensable analyses. In any event, extrapolations to values of F_F in the range, say, about 0.001/year are highly uncertain.

3) Dam Failures: To determine F_F for dam failures, it is necessary to analyze the contingent likelihood, given various rare and threatening events, that the dam would "fail", and thereby produce a PMF-sized flood or greater at the site being considered.

It is also important for the analyst to remember that severe local precipitation can add to the flooding level from a dam failure. This issue must be accounted for in the flooding analysis.

There seems to be no single generally accepted methodology for analyzing the frequency F_F of a dam failure that would produce a PMF-sized flood at a downstream reactor site. Such an analysis must be entirely dam-specific (meaning river-specific also), and depends on dam construction, spillway design capability, conditions of the reservoir and embankments, and other such factors. Realistic calculations of the dam failure probability of a specific dam as a function of extreme conditions are difficult to find in the literature; bounding calculations are more common, and would be fully acceptable if based on defensibly conservative models and data. Some bounding calculations provide values of F_F that are quoted as being in the range of 10^{-6}/year or even smaller, especially for modern well-engineered dams [Ref. 5.7 Oconee PRA, 1984]. On the other hand, some dam failures could easily be in the range of about $F_F = 10^{-3}$/year, since the mean value of the data base for F_F for all dams is in the range between 10^{-4} and 10^{-5} year (according to a survey published in the Oconee PRA [Ref. 5.7, Oconee PRA,1984]).

4) Ocean (Coastal and Estuarine) Flooding: We will discuss here the approaches to analyzing F_F for coastal and estuarine sites whose grade elevation is sufficiently close to sea level that there is no easy way to dismiss these effects convincingly.

At such coastal and estuarine sites, the principal causes of extreme flooding are storm surge and wave run-up action, arising from a combination of tropical storms (hurricanes, etc.), extreme tides, and high local rainfall that can inundate a site. At a few sites, it is also important to consider seiche phenomena --- wind-excited or seismic-excited waves occurring at the natural resonant frequency of the water body, so that the resonance results in amplified effects.

As is true for other flooding phenomena, it is important for the analyst to remember that severe local precipitation can add to the flooding level. This issue must be accounted for in the flooding analysis.

At Atlantic sites, the flood design basis is usually a storm surge resulting from the Probable Maximum Hurricane (PMH), determined using methods developed by NOAA and the Army Corps of Engineers similar to those used for the Probable Maximum Flood at river sites. The designer must work out both the "stillwater storm surge" for the specific site topography, and also the wave run-up heights that can come on top of the stillwater surge level. In estuarine areas, the local effects of channeling and river run-off must also be considered.

One methodology that has been used for this type of analysis is to develop historical data for large tides, and separately for hurricane-induced surges and wave run-ups. These are combined using a joint probability analysis, in which the key issue is determining (or assuming) the extent of correlations in these extreme phenomena. One can develop estimates for return periods in the 100-year range without difficulty, since historical data are the basis, but extrapolations beyond a century or so require assumptions or a physically plausible model. Examples of extrapolation models are the "Type I Extreme Value Distribution" and the "Lognormal Distribution", both of which were used in the St. Lucie probabilistic analysis [Ref. 5.10, St. Lucie A-45, 1987; Ref. 5.16 Myers, 1970]. Using these extrapolation models, it was estimated that the frequency of flooding (F_F) to plant grade at St. Lucie, located north of Miami, Florida on the Atlantic Ocean, ranges from about 10^{-5} to 10^{-6} per year. For the Turkey Point site, located south of Miami on the same Florida coast, the same methodology found a range of about 10^{-4} to 10^{-6}/year [Ref. 5.11, Turkey Point A-45, 1987].

Unfortunately, no methodology for making the needed extrapolations has been accepted broadly by the community of flooding experts. All such methodologies are highly uncertain because of the inability to deal with possible correlations among the extreme phenomena, which correlations could invalidate the usual convolution approaches.

Therefore, a conclusion similar to that reached for river sites seems appropriate here, as follows: If it is necessary at a specific ocean/esturaine site to work out the value of F_F for flooding from the PMH or other similar "extreme" flood, because the site is not easily shown to be essentially invulnerable, values of F_F is the range of 0.01/year are probably defensible, but values in the range of 0.001/year or smaller may be very hard to defend. Extrapolations to much smaller values can be accomplished using models and assumptions, but the associated uncertainties in the results for F_F are great. Such extrapolations, although feasible, require careful analyses and documentation.

5) Tsunami Flooding (Ocean Sites): Tsunamis can occur due to either local or distant effects. Distant tsunamis are a relatively frequent event in the Pacific Ocean and less frequent in the Atlantic Ocean and the Gulf of Mexico. There is a reasonable data base for the frequency and size distribution of Pacific Ocean tsunamis, extending for a few hundred years in some areas. The design basis approach for those few sites affected has generally been to work out the amount of potential inundation from a very

large but still credible tsunami wave, and then to provide for enough site-elevation and/or specific engineered features to account for the possible effects.

Determining a realistic value for F_F for large tsunamis is clearly a site specific analysis problem. Considerations include not only the size distribution of tsunamis arriving at the site, but local subsurface topography. The effectiveness of either engineered or natural barriers must be assessed probabilistically, or at least deterministically for several tsunami wave heights. From this analysis, F_F can be calculated.

If F_F is not acceptably small, a response analysis would be required, to determine which critical equipment would fail due to the flood. This analysis would not differ in type from standard PRA-style systems analysis.

For the effect of tsunamis on reactors, there is no probabilistic analysis along these lines in the extant literature. However, the broad outlines of how such an analysis could be accomplished are well known, and this analysis should be well within the capabilities of existing PRA methodology.

6) Lake Sites, High Water Level, Wave Effects, Surges: At lake sites where the plant grade is not sufficiently elevated to be obviously out of danger, site flooding can occur due to one or a combination of several different effects. Very large precipitation (although not necessarily as large as the PMP) is one such effect. Besides precipitation, the most obvious issue is arise in lake level to a height where it can flood over barriers. Wind-driven effects can add to the effective lake level, and wind-generated waves can provide still further height. This is shown schematically in Figure 5.1, taken from the Point Beach PRA study [Ref. 5.18, Point Beach A-45, 1987].

For both small lakes and large ones such as the Great Lakes, water levels rise and fall over the years due to changes in precipitation and other hydrological aspects. Such level changes are a well-known phenomenon, although the historical data base is often limited because good measurements are not available. Figure 5.2, also taken from the Point Beach A-45 study [Ref. 5.8], shows an analysis of Lake Michigan (one of the Great Lakes) near Milwaukee, in which the underlying data base consists of about 100 years' measurements; in this figure, extrapolations have been made out to 500-year recurrence periods.

Analysis of wind-driven effects, including both wind setup and wind-generated high waves, must be accomplished for each specific site, since the depth and shape of the lake bottom near the shore are the key determining factors. Typically, these wind-driven effects are much larger than the variations in lake level itself; for example, the Lake Michigan information in Figure 5.2 shows variations of less than two feet between the 10-year and the 500-year lake levels, while wind-generated effects can be as great as five or even ten feet.

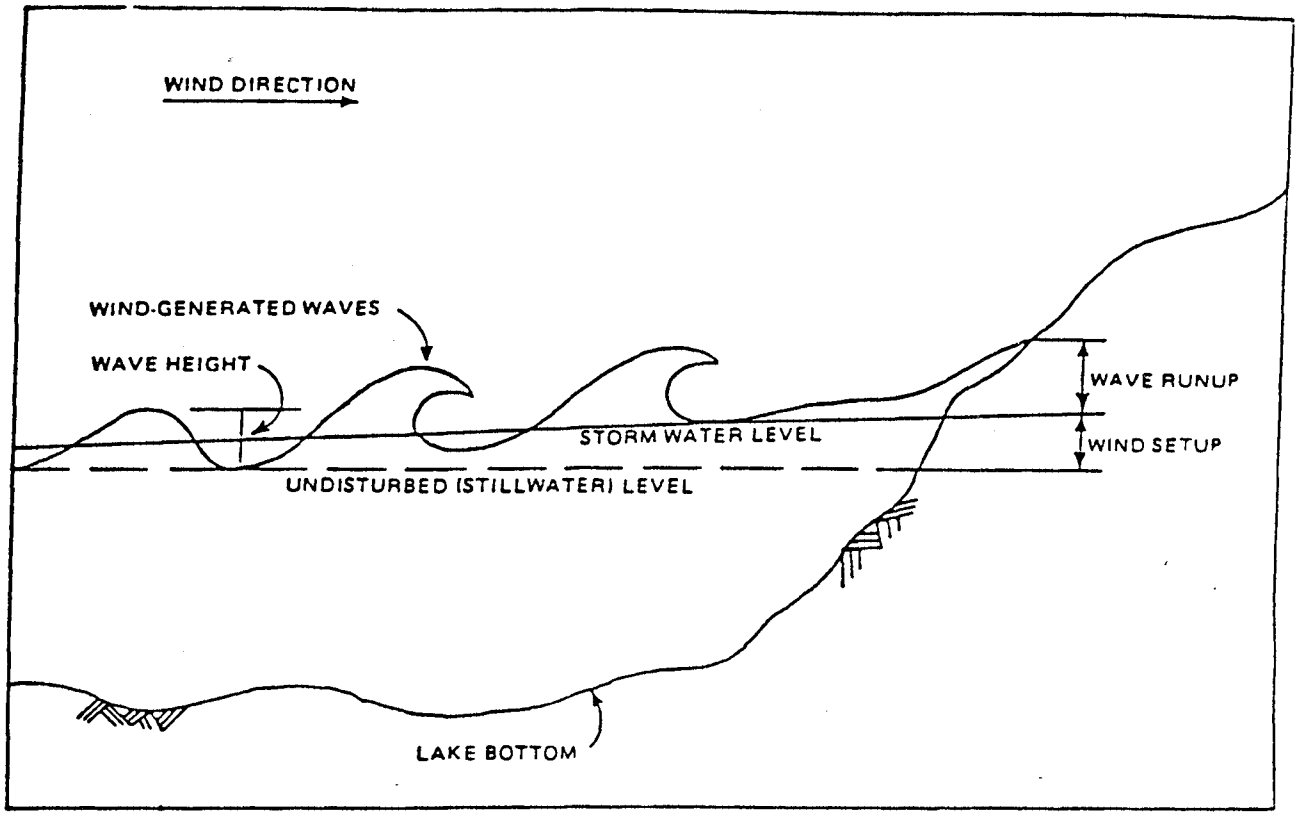
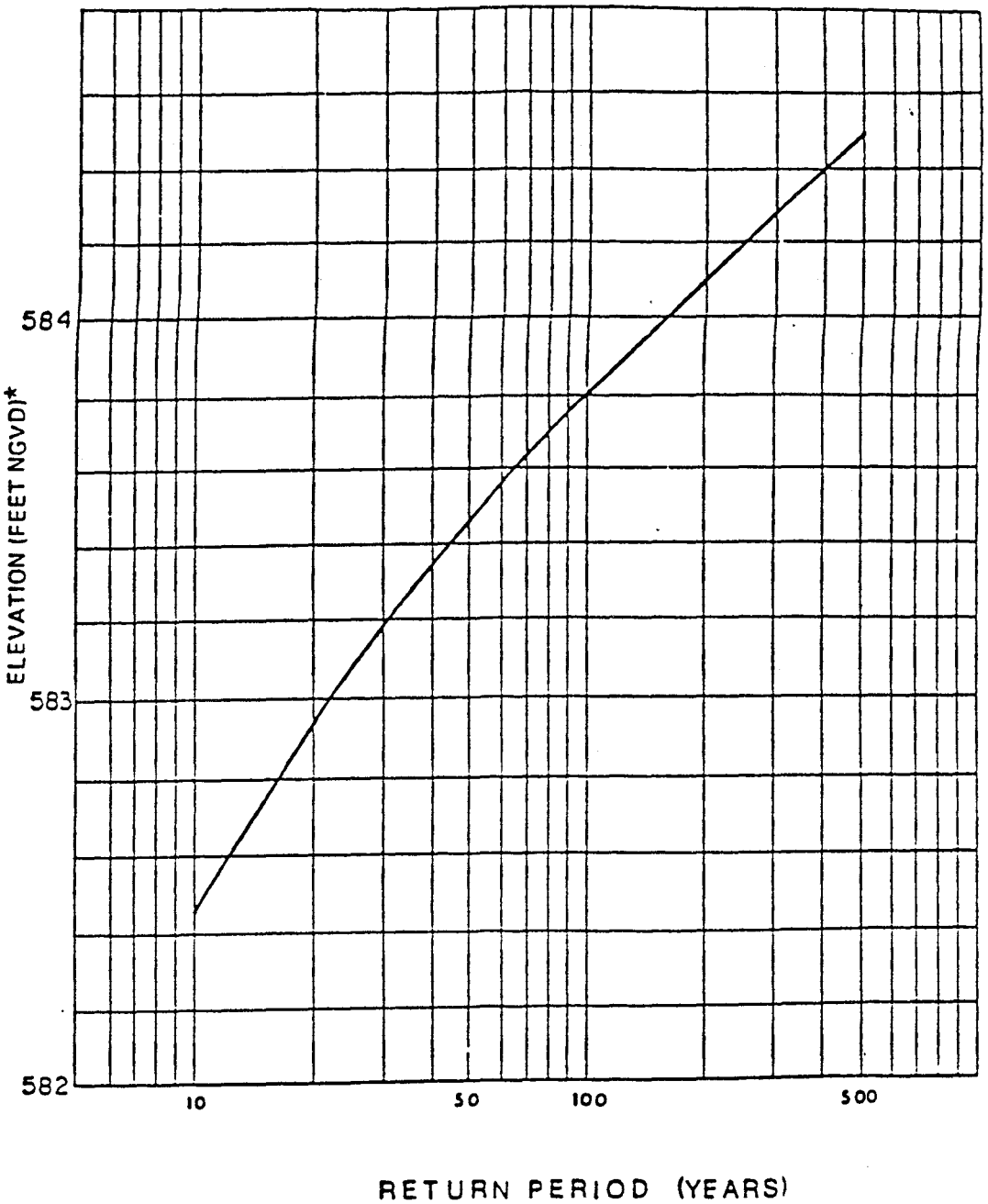


FIGURE 5.1 SCHEMATIC ILLUSTRATION OF WAVE RUNUP

[Ref. 5.8, Point Beach A-45, 1987]



* Note: Elevation is defined in terms of NGVD, subtract 1.2 feet to obtain IGLD.

FIGURE 5.2 FLOOD INSURANCE STUDY HAZARD CURVE FOR LAKE MICHIGAN [Ref. 5.8, Point Beach A-45, 1987]

Analysis of F_F, the frequency/year of a flood high enough to reach important plant equipment, can usually be accomplished based on historical data out to levels like the 100-year recurrence (lake heights down to about F_F = 0.01/year). Both lake-level and precipitation effects must be considered.

Beyond the historical data base, extrapolations are needed. Theoretically based models for wind set-up and wind-generating wave phenomena exist, and have been used for several analyses in the SAR literature. These models can deal with relatively rare phenomena due to combinations of effects, and are routinely used to predict levels in the Great Lakes with recurrences in the few-hundred-year range (F_F = few x 10^{-3}/year). Beyond that, uncertainties are large due to lack of understanding of the correlations among extreme phenomena, such as precipitation-driven high lake levels and their correlation with very high wind effects. Such extrapolations require careful analyses and documentation.

5.3.2.2 Contingent Probability of Core Damage, P_{CD}:

The quantity P_{CD} was defined above (Section 5.3.2) as the probability, given a flood large enough to cause more than minimal damage, that a core-damage accident will occur; P_{CD} is a contingent probability with value between 0 and 1. The exact definition of P_{CD} in a given application will depend, of course, on the character of the flood being studied ---- the phrase "a flood large enough to cause more than minimal damage" requires specfic detail that will be different in different applications.

Whenever the frequency F_F for a serious flood is not demonstrably known to be very small, it is necessary to develop information on P_{CD} in order to enable a comparison with the figure-of-merit # 1 on core-damage frequency (see Chapter 1). This implies some type of systems analysis. Note that because P_{CD} is a probability, and because there is no real-world data base, the systems analysis must be intrinsically probabilistic in character. There are two different approaches to analyzing P_{CD}, one using a bounding approach and the other using a fully-developed PRA methodology.

Bounding Analysis: A bounding analysis for PCD would proceed along the following lines: beginning with the postulated flood, the analysis would first determine whether there would be enough warning time to enable the plant to be secured in a safe-shutdown condition. If so, the analysis need only examine how the flood might compromise the plant's ability to maintain such a shutdown condition. (This analysis is much easier to perform than is a full response analysis).

Assuming inadequate warning time (or some intermediate condition), the analysis must proceed further. The first step is to identify equipment and structures that are "threatened", in the sense that they could be either inundated or structurally affected. The initial list should be as comprehensive as feasible --- of course, for some sites the list may be quite short, depending on the site layout. The most convincing bounding analysis for P_{CD} would then be one in which the total loss of all such threatened

equipment and structures would still leave enough other capability to achieve safe shutdown, and so would not result in a core-damage accident. In this approach, one would probably conservatively assume the loss of all equipment contained within or dependent on any threatened structure. (In probabilistic language, the analysis would need to show that the remaining combinations of failures leading to potential core damage are of sufficiently low contingent probability, taking into account the frequency F_F of the initiating flooding event.)

If the demonstration above cannot be accomplished in a highly convincing manner, the bounding analysis could proceed to develop more detailed models. For example, the analysis could account for recovery of certain equipment or could take credit for performing specified safety functions by alternative means, as appropriate. Another example might be using engineering arguments to demonstrate that structural threats to buildings or foundations would not necessarily compromise the equipment dependent on the structure.

The ultimate objective of the bounding analysis would be a demonstration, taking into account the frequency F_F of the initiating flood, that the combination of $F_F * P_{CD}$ is sufficiently small to satisfy the figure-of-merit #1.

Full Probabilistic Analysis: If a bounding analysis cannot be accomplished, or yields unsatisfactory results, it is necessary to develop a quantitative estimate for P_{CD} through a full PRA-type study. The analysis would begin by developing an event tree for the flooding sequence, that would incorporate the key items of equipment involved and the safety functions they support. This would then need to be combined with a flooding-fragility analysis to determine the contingent probabilities of failure of specific safety functions, systems, and components from inundation or structural failure.

Such a full-scope PRA approach, following normal PRA-type procedures, has never been carried out on any plant. The best examples of the approach are the five A-45 limited-scope flooding PRA studies carried out by Sandia National Laboratories in 1986-87 [Refs. 5.1, 5.8, 5.9, 5.10 and 5.11] which attempted an abbreviated systems analysis of items threatened by external flooding. These analyses can best be characterized as in-between a bounding-type analysis and a full-scope PRA analysis. These analyses developed functional event trees, fragility models in terms of inundation as a function of flood level, and a systems study of combinations of failures leading to core damage. Typically, values of P_{CD} in the range from 0.02 to 1 were found (or used in a "bounding" sense, with conservative assumptions in place of realistic analysis of some issues).

Summary: The best way to summarize this discussion of analyzing P_{CD} is to observe that there do exist approaches to calculating, estimating, or bounding the value of P_{CD}. These methods use analytical tools that are well-developed in the hands of PRA systems analysts for other applications, although they have not been widely applied to external flooding scenarios.

5.3.2.3 Contingent Probability of a Large Release, P_{LR}:

The quantity P_{LR} was defined above (Section 5.3.2) as the probability, given a core-damage accident from flooding, that the accident will evolve into a 'large release' scenario; P_{LR} is a contingent probability with value between 0 and 1. The exact definition of P_{LR} in a given application will depend, of course, on the character of the flood being studied ---- the phrase "given a core-damage accident from flooding" requires specific detail that will be different in different applications.

In some analysis situations, it will not be necessary to analyze P_{LR} in order to assure that figure-of-merit #2 (concerning large-release accidents) is satisfied. This would be true if F_{CD}, the frequency of core-damage accidents, is itself smaller than FOM #2. For those situations where P_{LR} must be analyzed, the starting point is necessarily the preceding analysis of P_{CD} and $F_{CD} = F_F * P_{CD}$. Given such an analysis, the analysis of P_{LR} would require the binning of the various relevant core-damage accident sequences into plant-damage states or other categories that reveal the potential for a large-release scenario. From such information, it should be feasible to develop a quantification (or perhaps an upper bound if appropriate) for the value of P_{LR}, using well-developed PRA-type methods of analysis.

There are no examples in the literature of external flooding analysis that has extended to calculations of P_{LR}, F_{LR}, or something similar.

5.3.3 The Analysis of External Flooding in the Full-Scope PRA Literature

Only a few full-scope PRAs have addressed external flooding, and in most cases the analysis is very brief, often stopping short even of quantifying the value of F_F (the frequency of potentially large flooding) by making qualitative or semi-quantitative arguments.

Here a brief survey of the full-scope PRA literature is presented to provide background about what is known, and why. The findings of each PRA will be summarized more-or-less as published, although some interpretation will be offered where appropriate.

1) Zion [Ref. 5.12]: The flooding concern at Zion is from adjacent Lake Michigan. The PRA was carried out by Pickard Lowe & Garrick under support of the utility, Commonwealth Edison. The PRA writeup (Section 7.4) observes, based on FSAR information, that the grade floor level of all key structures is above the "wave run-up elevation", and that the "sheet pile wall" along the Lake Michigan shore would preclude waves breaking on any key items. Equipment in the auxiliary building below the critical lake level is protected by high access doors. Based on these arguments, and having concluded that there is no need for quantification of either the frequency of a large flood or the plant response, the PRA says that "external flooding is believed to be an insignificant contributor to core melt risk."

2) Indian Point [Ref. 5.2]: The flooding concern at Indian Point is the adjacent Hudson River. The PRA was carried out by Pickard Lowe & Garrick under support of the utilities, Consolidated Edison and New York Power Authority. The PRA states that the "maximum high water mark" at the site is calculated at 13.1 feet (mean sea level) with a frequency estimated at 3×10^{-3}/year. The highest observed river level since the year 1635 would correspond to 7.5 feet at the site. Taking into account a simultaneous river maximum flood, a hurricane at New York Bay, and maximum precipitation causing an upstream dam failure, the river could reach 14.0 feet, with a collective frequency (including the dam failure) estimated to be in the range between about 10^{-8} and 10^{-12}/year. Since all plant grade elevations exceed 14.0 feet for plant facilities, the PRA concludes that "external flooding contribution to core melt frequency is extremely small."

3) Limerick [Ref. 5.3 and 5.4] The flooding concern at Limerick arises from local intense precipitation, since the PMF in Possum Hollow Run and a standard project flood in the Schuylkill river produce high waters 57 feet below plant grade. The PRA was carried out by NUS Corporation under support of the utility, Philadelphia Electric Company. The PRA states that the PMP level of local precipitation does not cause flooding, due to the local topography, and this conclusion is backed up by a detailed hydrological study. Ponding on roofs has also been analyzed and found not to be a problem. Based on these arguments, the PRA states that "the risk to the plant of external flooding is negligible."

4) Oconee [Ref. 5.7]: The flooding concern at Oconee (Unit 3) is due to a postulated failure of the upstream Jocassee Dam. The PRA was carried out under EPRI support by Duke Power Company, the EPRI staff, and a consortium of engineering firms. The PRA does not analyze Jocassee Dam specifically, but uses a generic dam failure data base to make an approximate but conservative estimate of Jocassee Dam failure, based on causes other than earthquakes; the value found is a mean frequency of 2.3×10^{-5}/year. Given the dam failure, the PRA analysis states that analyzing the conditional probability that the Oconee site would be flooded would be complex. A value of unity is therefore conservatively assigned for that contingent probability. The PRA also concludes that the conditional probability of a core-damage accident, given site flooding, is also difficult to analyze; again, a conservative value of unity is assigned. The PRA therefore finds a mean value of 2.3×10^{-5}/year for the core-damage frequency from flooding, due to the possible failure of Jocassee Dam from non-seismic causes.

5) Millstone-3 [Ref. 5.5]: The flooding concern for Millstone-3 is tidal flooding and intense local precipitation. The PRA was supported by the utility, Northeast Utilities. According to the PRA, tidal flooding would be greatest during a Probable Maximum Hurricane (PMH), resulting in maximum still water and maximum wave runup levels of 19.7 and 23.8 feet above sea level, respectively. (For comparison, the maximum level ever recorded in the area is 9.7 feet, with estimated frequency of 3×10^{-3}/year.) Since all safety-related equipment is above 24.0 feet elevation, the PRA concludes that "tidal flooding is an insignificant contributor to plant risk." The Probable Maximum

Precipitation event for the site, which is claimed to be three times greater in magnitude than the 100-year-recurrence intensity, could result in temporary flooding of the site area in the vicinity of safety-related buildings. However, the PRA concludes that flooding of safety-related buildings is prevented by grade floor elevations and access openings at higher elevations. The PRA concludes that "flooding of the Millstone site as a result of intense precipitation was determined to be an insignificant contributor to plant risk."

6) Point Beach [Ref. 5.8]: The flooding concern at Point Beach is from nearby Lake Michigan. The central concern is rising lake level including wave runup. The PRA analysis was accomplished by Sandia National Laboratories as part of the A-45 project that examined plant vulnerabilities of decay-heat-removal systems. This PRA analysis will be discussed in some detail to demonstrate the power of the method.

The design-basis flood at Point Beach is +8.42 feet above median lake level, and the ground floor elevation of the Turbine and Auxiliary Buildings is at +8.00 feet, so emergency procedures call for sandbagging to +9.0 feet when flood warnings occur.

The flooding potential was estimated by using a statistical frequency distribution for lake level from 95 years' gauge data at Milwaukee, and a distribution of wave runup was estimated from lake shore characteristics. Three different alternative statistical models were used for the lake-level distribution, and were given approximately equal weight because there was thought to be no strong preference among them. For two of these three models, a skew coefficient based on the data was used with 2/3 weight, and a zero skew coefficient was used with 1/3 weight. This gave 5 models, each of which was used to generate 5 hazard curves to express the statistical uncertainty in the data. Thus 25 hazard curves were generated in all, with various weights. From these, it was determined that the frequency of the lake level exceeding the +8.0 foot elevation without wave runup is less than about 10^{-10}/year.

The runup estimates were generated using three different models that considered the rip-rap structure closest to the pumphouse; these models were given weights of 20%, 60%, and 20% by subjective judgment. The runup estimates were added to the lake-level estimates to obtain a better family of hazard curves. The finding was that at locations nearest the pumphouse, over topping of the banks will occur with a frequency in the range between about 10^{-3} and a few x 10^{-2} per year. An estimate was made as to how much water would be collected (ponded) between the pumphouse and the turbine building, as a function of how much overtopping would occur and considering natural runoff. About one foot of impounded water would occur with a frequency in the range of about 10^{-4}/year, and two feet would be impounded with a frequency in the range of about 10^{-8}/year.

Component "fragility" was evaluated based on the assumption that a submerged component would fail at a specific "critical flood depth." Assumptions were made about how the sandbags would be deployed and where the

water would go for various flood heights. A functional event tree was developed to study which combinations of lost functions, related to which items of flooded equipment, would lead to a core-damage accident. Support-system failures were also included. The limiting situation is a lake elevation, including runup, of + 16 feet with an estimated mean frequency of about 2 x 10^{-8}/year. By comparison, a dangerous lake-elevation rise without wave runup would occur with a frequency estimated to be much smaller, by several orders of magnitude. The limiting situation causing a core-damage accident is loss of key support systems (diesels, auxiliary feedwater, service water), loss of secondary cooling due to loss of batteries and key switchgear, loss of emergency injection capability due to loss of high-pressure injection, and loss of residual-heat-removal capability.

7) Turkey Point [Ref. 5.11]: The flooding concern at Turkey Point is from hurricane storm surge in nearby Biscayne Bay, an Atlantic Ocean bay south of Miami, Florida. The PRA analysis was accomplished by Sandia National Laboratories as part of the A-45 project that examined plant vulnerabilities of decay-heat removal systems.

The methodology used in this study is similar to that used at Point Beach (see discussion just above). A numerical model was used to predict the time variation in storm surge for a given set of hurricane input parameters. Data on historic hurricane events were then combined with the storm-surge model, and several different runs were made of the model for different parameters, to generate a single hazard curve. The uncertainty in that hazard curve was then estimated using a statistical approach based on several alternative hypotheses for the distributions of various key parameters. A total of 30 hazard curves were generated, with weights. The results were that the mean frequency of still water surge elevations exceeding plant grade (18 feet) is about 2 x 10^{-4}/year, and of exceeding the floodwall elevation of 20 feet is about 6 x 10^{-5}/year. There is very large uncertainty in these estimated values, however: the mean and median values differ by a factor of about 70, and the mean is above the 90th percentile of the distribution. Including wave run-up, the mean frequency of exceeding elevation 19 feet is about 10^{-4}/year.

Failure of components was assumed to occur when submerged to a critical height that was analyzed for each key component. At about 19 feet certain critical equipment is flooded; additional equipment is flooded at about 20 feet; and still other equipment at about 20.5 feet. Three different cases were analyzed: (i) Flooding to +19 feet but below +20 feet (about 6 x10^{-5}/year) produces a core-damage accident due to loss of diesel generators (station blackout) unless offsite power is recovered within 14 hours; assuming offsite-power recovery as 98% probable (2% chance of failure), the core-damage frequency is estimated at about 1 x 10^{-6}/year. (ii) Flooding to +20 feet but below +20.5 feet occurs with frequency of about 1.6 x 10^{-5}/year; core-damage due to auxiliary feedwater failure will occur in two hours because of battery depletion, unless offsite power is recovered (probability of recovery in 2 hours is 80%, or 20% failure probability). This gives a core-damage frequency of about 3 x 10^{-6}/year. (iii) For flooding above +20.5 feet (frequency about 4 x 10^{-5}/year), core-damage will commence within about 30 minutes since all

pumps are submerged, and no recovery actions are considered likely.

Adding up cases (i), (ii), and (iii) the total estimated frequency of core-damage at Turkey Point from hurricane-induced/storm-surge flooding is about 2×10^{-4}/year without offsite-power recovery, and about 1×10^{-5}/year with recovery as assumed. The PRA states that these are conservative estimates, since warning time could result in securing the reactor before the flooding would occur, and since recovery of offsite power might be more likely than assumed.

8) St. Lucie, Unit 1 [Ref. 5.10]: The flooding concern at St. Lucie-1 is from Atlantic Ocean flooding, since the site is on a low lying salt marsh on an island between the open ocean and a salt-water estuary in southern Florida. The controlling problem is hurricane storm surge, as at nearby Turkey Point (see above). The PRA analysis was accomplished by Sandia National Laboratories as part of the A-45 project that examined plant vulnerabilities of decay-heat removal systems.

The hazard methodology used for hurricane storm surge is very similar to that used for Turkey Point (see above). Two different levels of flooding (with wave runup) are of concern, +19 and +22 feet: (i) At +19 feet (frequency estimated at about 9×10^{-7}/year), auxiliary feedwater systems are flooded, causing loss of secondary cooling. Emergency AC and high-pressure injections systems are still available for feed-and-bleed cooling, which can cool the core unless they fail (with a failure probability estimated to be 10% due to non-flooding causes). This produces a core-damage frequency of about 9×10^{-8}/year (flooding frequency x 10%). (ii) At +22 feet, intake cooling water and component cooling water also fail, and there is no recovery available to prevent core-damage. The frequency of core-damage is simply the frequency of a flood at +22 feet, estimated to be about 2×10^{-6}/year. Summing (i) and (ii), the assessed core-damage frequency at St. Lucie-1 from hurricane/storm surge flooding is about 2.1×10^{-6} This is said in the PRA to be a conservative estimate because no credit is taken for early reactor shutdown if enough warning is available.

9) Quad Cities [Ref. 5.9]: The flooding concern at Quad Cities is the Mississippi River. The PRA analysis was accomplished by Sandia National Laboratories as part of the A-45 project that examined plant vulnerabilities of decay-heat-removal systems.

According to the PRA, the likelihood of flooding causing a core-damage accident at Quad Cities is assessed to be very low (below 10^{-7}/year), because procedures call for reactor shutdown if the river were to rise to levels even several feet below where any important equipment could be damaged; cold shutdown followed by removal of the reactor vessel head could be accomplished before flooding could occur, even at conservatively large rates of rise in the river level. Furthermore, the frequency of river levels high enough to be of concern is assessed at about 10^{-6}/year, although with great uncertainty.

10) Arkansas Nuclear One-1 [Ref. 5.1]: The flooding concern at ANO-1 is Dardanelle Reservoir on the Arkansas River. The PRA analysis was accomplished by Sandia National Laboratories as part of the A-45 project that examined plant vulnerabilities of decay-heat-removal systems.

The systems analysis reveals that only one river flooding level is important: at +361 feet all important safety systems would fail by submergence, and core-damage would begin within about 30 minutes if no prior steps have been taken to secure the plant. Recovery actions once the river has reached this level are not considered credible. Since plant shutdown procedures should be initiated once lake levels reach +340 feet (but with no specification for how quickly they must be accomplished), the core-damage estimate based solely on +361-foot flooding is probably highly conservative.

The likelihood of river flooding to the +361-foot level is conservatively estimated to be below 7×10^{-6}/year. The methodology used is to fit a statistical frequency distribution to stream gauge data gathered near the site. Three alternative models were used to estimate the frequency of peak flooding, and subjectively they were assigned equal weighting. For each of these three models, five hazard curves were generated, to capture the statistical uncertainty in the data. Using this approach, the mean frequency of exceeding +361-foot elevation is estimated as about 4×10^{-6}/year. However, coincident failure of an upstream dam, the Ozark Dam, could add 3 feet to the peak of the flood if that dam were overtopped. The PRA states that there is no information to guide an estimate of the probability of dam overtopping. Assuming 100% likelihood of dam overtopping, the frequency for exceeding +361 feet is 7×10^{-6}/year, which is conservatively given as a bound on the frequency of flooding sufficient to cause a core-damage accident at ANO-1.

Table 5.1 summarizes the principal flooding concern, flood frequencies and core damage frequencies of the ten PRAs reviewed for this study. Care should always be exercised when comparing the results of one PRA with another due to methodological and data base differences between the PRAs. Nevertheless, Table 5.1 may be useful in that it presents a limited "snapshot" of the present state of knowledge regarding external flooding hazards to nuclear plants.

Table 5.1

External Flooding PRA Frequencies

Plant	Flooding Concern	Flood Frequencies (/year)	Core Damage Frequencies (/year)
Zion 1 and 2	Lake Michigan	"insignificant"	"insignificant"
Indian Point 2 and 3	Hudson River	3×10^{-3} (13.1 ft. MSL)	"extremely small"
Limerick	Local Precipation	"negligible"	"negligible"
Oconee 3*	Jocassee Dam Failure	2.3×10^{-5}	2.3×10^{-5}
Millstone 3	Tidal Flooding, Atlantic Ocean; Local Precipation	"insignificant"	"insignificant"
Point Beach A-45 PRA	Lake Michigan	$<10^{-10}$ (+ 8.0 ft. w/o runup) 2×10^{-8} (+16 ft. w/ runup)	N.A.
Turkey Point A-45 PRA	Biscayne Bay, Atlantic Ocean	2×10^{-4} (> 18ft. w/o runup)	2×10^{-4} (w/o offsite power recovery)
		6×10^{-5} (> 20 ft. w/o runup) 10^{-4} (> 19 ft. w/ runup)	1×10^{-5} (w/ offsite power recovery)
St. Lucie 1 A-45 PRA	Atlantic Ocean	9×10^{-7} (+ 19 ft.) 2×10^{-6} (+ 22 ft.)	9×10^{-8} (+ 19 Ft.) 2×10^{-6} (+ 22 ft.)

Table 5.1 (continued)

External Flooding PRA Frequencies

Plant	Flooding Concern	Flood Frequencies (/year)	Core Damage Frequencies (/year)
Quad Cities A-45 PRA	Mississippi River	$< 10^{-7}$	N.A.
Arkansas Nuclear One 1 A-45 PRA	Dardenelle Reservoir, Arkansas River	$<7 \times 10^{-6}$ (+ 361 ft.)	N.A.

(*) Assumed 100% contingent probability of core damage given site flooding.

N.A. = information not available at time of table preparation.

5.3.4 Discussion of Methodological Issues

In Section 5.3.2, three different quantities were identified whose determination would be necessary for a given reactor installation to allow a comparison with the two figures-of-merit. These quantities were defined as F_F, P_{CD}, and P_{LR}, which are then combined together to provide the relevant frequencies of core-damage (F_{CD}) and of a large-release accident (F_{LR}).

Flood Frequencies (F_F): In Section 5.3.2.1, the discussion of methods for determining F_F showed that, whenever external flooding cannot be easily ruled out by site/elevation/configuration arguments, the actual calculation of F_F is highly uncertain for floods with return periods much in excess of the historical record, which is one to a few hundred years(corresponding to F_F values not much below about 0.01/year).

There are a number of methodologies available for working out F_F in a formal sense. A good example of the general approach can be found in a recent NRC-sponsored study by Borgman [Ref. 5.14, Borgman, 1983], in which a formal mathematical approach is set down for determining not only F_F but also other related quantities. Unfortunately, the formalism's usefulness is limited by lack of knowledge of the crucial pieces of information, which are the correlations among various extreme events. Lacking these correlations, from either a sound data base or a sound theoretical basis, the analysis of F_F for return periods far beyond the historical data base seems to be quite uncertain and requires careful analysis.

Core-Damage Probabilities, P_{CD}: The discussion in Section 5.3.2.2 of methods for determining P_{CD} and F_{CD} indicates that these methods do exist, and are relatively straight forward extensions of approaches commonly used by PRA analysts for studying other phenomena. However, there have been only a few limited applications of these approaches for external-flooding analysis.

Large-release Probabilities, P_{LR}: The discussion in Section 5.3.2.3 of methods for determining P_{LR} and F_{LR} indicates that the PRA-type method for studying plant-damage-states and related issues exists, but has never been exercised in any external-flooding analysis. If such an analysis were needed for a plant, accomplishing it would be breaking new ground, although such an analysis would be not very different from analyses now routinely performed for other accident initiators, both internal and external.

5.3.5 Evaluation and Comparison with Figures-of-Merit

Based on the discussion above of methods for studying accident frequencies for external flooding probabilistically, it is possible to make a judgment concerning the comparison of the published PRA results with the two figures of-merit introduced in Section 5.2 above. This judgment is as follows:

Figure-of-Merit # 1, Core Damage Frequency: There have been only a very few analyses of core-damage frequency from external flooding. In almost every

study, a bounding-type argument was used to support the assertion that external-flooding-initiated accidents are a very minor contributor to core-damage frequency, in the context of the larger frequencies found for other initiators. In some cases, these bounding arguments have relied on well supported analysis of local site conditions, especially when the reactors are sufficiently elevated to rule out severe flooding. In other cases, calculations of flooding return periods have been relied on, and some of these are only crude "estimates" rather than "analyses" in the more formal sense. It is concluded here that, whenever these return periods are relied on at levels much below F_F of, say, about 10^{-3} per year, the analyses are probably very much too uncertain to be of great usefulness. However, when combined with an analysis of P_{CD} through study of the fragility of safety components, systems, and functions, it is likely that all of the plants studied can meet FOM # 1. Because demonstration of this in actual analyses is available for only a very few plants in the existing literature, this broad conclusion would need to be supported by detailed analysis for any given plant and site.

Figure-of-Merit # 2, Frequency of a Large Release: There have been no analyses of external flooding in the PRA literature that have been carried out to examine the issue of large releases. Meeting FOM # 2 is feasible, of course, if core-damage frequency by itself is low enough. Otherwise, more extensive analysis would be needed.

5.4 SUMMARY AND CONCLUSIONS

5.4.1 Conclusion # 1: External Flooding Must be Included

Based on the discussion above, it is clear that external flooding must definitely be including among those "external initiators" that should be part of any analysis of nuclear power plant vulnerabilities. Omitting their consideration entirely in a vulnerability analysis would not be appropriate. However, for some sites it may be easy to demonstrate that there is no threat from flooding.

5.4.2 Conclusion # 2: Core-Damage Frequencies

Based on the discussion above, there is no conclusive evidence for any plant that its flood-induced core-damage frequency is above figure-of-merit # 1. However, for many sites it may be difficult to show that the frequency of potentially threatening floods is sufficiently small; in those cases some response analysis is needed to determine whether plant systems can reach and maintain a safe-shutdown condition in the rare event that a large flood would occur. Either a response analysis of a bounding character or a realistic systems analysis can be acceptable if appropriately performed.

5.4.3 Conclusion # 3: Frequency of Large-release Accidents

Not enough analysis has been performed to know the extent to which plants meet figure-of-merit # 2 concerning the frequency of potential accidents leading to a large release. Because there exists no plant-specific analysis

even hinting at a solution in this area, the literature is inadequate at present to support a general conclusion.

5.4.4 Identification of Areas in Which Existing Regulatory Requirements Have Not Assured Adequate Protection

External flooding is not well understood on a generic basis. Various plants have quite site-specific flooding concerns. The few broad insights derivable from the extant literature are so obvious as to be of little use: these insights can be expressed, in 6 words, as the trivial advice to "watch out for low-lying areas." In general, it is fair to conclude that the absence of vulnerabilities in the plants that have been studied to date is a broad affirmation of the accomplishments of the current regulatory approach.

5.4.5 A Proposed Approach for Plant Evaluation

Section 5.3 above contains detailed discussions of the methodological approaches that are considered appropriate for analyzing plant behavior and vulnerabilities in the area of external flooding. Broadly summarized, these approaches amount to studying F_F (the frequency of a large enough flood to threaten plant equipment), and then if necessary to study plant response in terms of the contingent likelihood of an accident scenario given the rare large flood.

Bounding analysis is certainly an appropriate approach if it can accomplish the objective of meeting the two figures-of-merit, and taking into account the uncertainties in such an analysis. If bounding analysis is not sufficient, then more comprehensive full-scope PRA approaches would be required.

A general observation is that most reactor sites will be easily analyzed and shown to be protected to an acceptable level against external flooding. Only a few sites may require more than a bounding type of analyses of the probability aspects.

For those few sites where the probability analyses cannot acceptably rule out the issue of external flooding, a plant-response analysis is required. A full-scope PRA analysis is one option, but not the only one. For example, if plant system models have been developed (including a support-systems dependency matrix) as part of an internal-events Individual Plant Evaluation (IPE) study, they can be used to quickly screen out al but the key external flood vulnerabilities.

Thus even a flood response analyses, if needed for a site, should not be a difficult burden, in the context of a broader plant analysis.

5.5 REFERENCES

[5.1] NUREG/CR-4713, SAND86-1832 Arkansas Nuclear One-1 (A-45), 1987: W.R. Cramond, D. M. Ericson, Jr., and G.A. Sanders, "Shutdown Decay Heat Removal Analysis of a Babcock and Wilcox PWR --- Case Study", Sandia National Laboratories, March 1987.

[5.2] Indian Point, 1983: Consolidated Edison Company and Power Authority of State of New York, Indian Point Probabilistic Safety Study (1983). The principal contractor was Pickard Lowe & Garrick Inc.

[5.3] Limerick PRA, 1981: Philadelphia Electric Company, Probabilistic Risk Assessment, Limerick Generating Station (1981). The principal contractor was General Electric Company, with assistance from Science Applications, Inc.

[5.4] Limerick SARA, 1983: NUS Corporation, Severe Accident Risk Assessment, Limerick Generating Station, NUS Report No. A-4161, prepared for Philadelphia Electric Company (1983).

[5.5] Millstone-3, 1983: Northeast Utilities, "Millstone Unit 3 Probabilistic Safety Study" (1983)

[5.6] WASH-1400 NRC, 1975: U.S. Nuclear Regulatory Commission, "Reactor Safety Study", (1975).

[5.7] Oconee, 1984: Electric Power Research Institute/Nuclear Safety Analysis Center and Duke Power Company, "Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3", in 4 volumes (1984)

[5.8] NUREG/CR-4458, SAND86-2496 Point Beach (A-45), 1987: W.R. Cramond, D.M. Ericson, Jr., and G.A. Sanders, "Shutdown Decay Heat Removal Analysis of a Westinghouse 2-Loop PWR--Case Study", Sandia National Laboratories, March 1987.

[5.9] NUREG/CR-4448, SAND85-2373 Quad Cities (A-45), 1987: S.W. Hatch, D.M. Ericson, Jr., and G.A. Sanders, "Shutdown Decay Heat Removal Analysis of a General Electric BWR3/Mark I -- Case Study", Sandia National Laboratories, March 1987.

[5.10] NUREG/CR-4710 St. Lucie (A-45), 1987: G.A. Sanders, D.M. Ericson, Jr., and W.R. Cramond, "Shutdown Decay Heat Removal Analysis of a Combustion Engineering 2-Loop PWR--- Case Study", Sandia National Laboratories, 1987.

[5.11] NUREG/CR-4762, SAND86-2377 Turkey Point (A-45), 1987: G.A. Sanders, D.M. Ericson, Jr., and W.R. Cramond, "Shutdown Decay Heat Removal Analysis of a Westinghouse 3-Loop PWR--Case Study", Sandia National Laboratories, March 1987.

[5.12] Zion, 1982: Commonwealth Edison Company, Zion Probabilistic Safety Study (1982). The principal contractor was Pickard Lowe & Garrick, Inc.

[5.13] Army Corps of Engineers, 1968: U.S. Army Corps of Engineers,"Engineering and Design -- Policies and Procedures Pertaining to Determination of Spillway Capacities and Freeboard Allowances for Dams", Engineering Circular Number EC 1110-2-27, Change 1 (19 Feb. 1968).

[5.14] NUREG/CR-2879 Borgman, 1983: L.E. Borgman, "Feasibility of Quantitative Assessment of Available Margins Inherent in Flood Protection of Nuclear Plants", U.S. Army Engineer Waterways Experiment Station, (1983).

[5.15] Interagency Committee, 1986: Work Group on Probable Maximum Flood Risk Assessment, under the direction of the Hydrology Subcommittee of the Interagency Advisory Committee on Water Data, "Feasibility of Assigning a Probability to the Probable Maximum Flood", Office of Water Data Coordination (1986).

[5.16] Myers, 1970: V.A. Myers, "Joint Probability Method of Tide Frequency Analysis", Environmental Science Services Administration, U.S. Dept. of Commerce, Technical Memorandum WBTM Hydro 11 (1970).

[5.17] NOAA/NWS, 1956: National Weather Service, U.S. National Oceanographic and Atmospheric Administration, "Seasonal Variation in the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1000 Square Miles and Durations of 6, 12, 24, and 48 Hours", Hydrometeorological Report No. 33 (1956).

[5.18] NOAA/NWS, 1975: F.P. Ho, R.W. Schwerdt, and H.V. Goodyear, "Some Climatological Characteristics of Hurricanes and Tropical Storms, Gulf and East Coasts of the United States", National Weather Service, U.S. National Oceanographic and Atmospheric Administration, Technical Report NWS 15 (1975).

CHAPTER 6 -- TRANSPORTATION ACCIDENTS

6.1 INTRODUCTION

This chapter presents a review and evaluation of the risk from reactor core damage due to the hazards posed by transportation accidents. The purpose of this chapter is to meet work requirements 2,4 and 5 as given in Chapter One of this report for transportation accidents.

Transportation accidents are among the factors that are considered in choosing a nuclear power plant site. The movement of hazardous (toxic/flammable/explosive) materials in or near the power plant site whose release, detonation or burning due to an accident involving the vehicle or pipeline, could affect the plant to the degree that damage to the reactor or release of radioactive material from the reactor could result must be considered. Another possible scenario involves the collision of the vehicle itself, with or without hazardous materials, into the power plant, causing sufficient damage to the plant to cause damage to the reactor or release of radioactive material from the reactor.

For the purposes of this report, transportation accidents are defined to be accidents involving aviation traffic of all types, ship and barge traffic, gas/oil/chemical pipelines, railroad traffic and truck traffic. The definition of an accident is defined for each mode of transportation in its relevant section.

The risk to nuclear power plants from transportation accidents and the subsequent risk of reactor core damage are discussed in the following sections. First, the current NRC regulatory requirements pertaining to nuclear power plant protection against the effects of transportation accidents are reviewed. Second, background information is presented for the evaluation of the risk of core damage due to transportation accidents. Next, the risk of core damage is evaluated in separate sections for each of the following modes of transportation:

1. Aviation (Commercial/General/Military),
2. Marine (Ship/Barge),
3. Pipeline (Gas/Oil),
4. Railroad, and
5. Truck.

General observations of the transportation risk analysis are presented in the next section. The final section presents the summary and conclusions for this chapter.

6.2 CURRENT NRC REGULATORY REQUIREMENTS

The regulation of nuclear power plants in regards to protection against the hazards posed by transportation accidents is covered by 10 CFR Parts 50.34, 100, and 100.10 [6.2.1] as listed in Chapter Two of this report under

the Transportation Accidents category. The guidance given to the NRC staff in reviewing Safety Analysis Reports (SARs) in this area is given by SRP Section Nos. 2.2.1-2.2.2, 2.2.3, and 3.5.1.6 [6.2.2] as listed in Chapter Two of this report under the Transportation Accidents category.

6.2.1 NRC Regulatory Guides

The specific guidance used by the NRC staff to review nuclear power plants regarding transportation accidents is given by the following NRC Regulatory Guides [6.2.3 - 6.2.5]:

Regulatory Guide 1.78 Assumptions for Evaluating Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

Regulatory Guide 1.91 Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants

Regulatory Guide 1.95 Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release

Reference 6.2.6, in its review of Regulatory Guide 1.78 [6.2.3], states that it provides detailed guidance for the safety analyses of toxic gas transport hazards. A shipment frequency criterion may exclude a chemical from further consideration. Shipments are defined to be frequent if the number per year equals or exceeds 10 for truck traffic, 30 for rail traffic, and 50 for barge traffic. The quantity of the chemical per shipment is specified for a range of distances from the control room to the accident and for three different types of control room ventilation systems. This information is summarized in Table 6.2.1. For a given distance of closest approach along the route and control room type, only shipments whose size exceeds the values in Table 6.2.1 need be counted. If this count for a chemical exceeds the above frequency criterion, the licensee must provide protection against accidents involving this chemical. If the frequency criterion is not met, then shipment of this chemical need not be considered.

Reference 6.2.7, in its review of Regulatory Guide 1.91 [6.2.4], states that it provides guidance for the safety analyses of explosions postulated to occur on transportations routes near the plant site. The Guide provides that if the peak positive incident overpressure (p_{so}) is below an arbitrary value of one psi, the explosive accident is not considered a threat to the plant and no further consideration is required. A p_{so} of one psi may yield a reflected pressure (the pressure actually applied to the structure) that is about two psi. The Guide also states that if the 1 psi incident overpressure criterion is satisfied, dynamic pressure (drag), blast-induced ground motion, and blast-generated missiles do not require further study. Also, for detonations of vapor clouds formed after an accidental release of hydrocarbon fuels, a TNT equivalent charge with 240% of the hydrocarbon mass is stated to be a reasonable upper bound for representing potential blast energy. Finally, the guide establishes probabilistic and deterministic approaches for analyzing

accidents which do not satisfy the one psi overpressure criterion. In particular, for determining the structural response to the blast loading, a static analysis using twice the appropriate pressure loading (implying a conservative dynamic load factor of 2) or an elastic analysis using dynamic load factors is stated as being acceptable. No provisions for inelastic response are given.

Reference 6.2.6, in its review of Regulatory Guide 1.95 [6.2.5], states that it provides specific safety analysis guidance for onsite or offsite releases of chlorine. Maximum single-container inventory quantities are specified for various standoff distances for six different types of control room. The requirements for control room ventilation system performance, emergency planning, and self containing breathing equipment are also specified.

Note that the current NRC regulatory approach is deterministic in nature, that is, assurance that a specific transportation mode is not of high regulatory concern is provided by determining whether any given transportation mode exceeds certain established threshold values (i.e., number of shipments per year near the site, explosion overpressure limits, distance to transportation route, etc.).

Table 6.2.1

Weights of Hazardous Chemicals that Require Consideration in Control Room Evaluations (for 50 mg/m^3 Toxic Limit and Pasquill Stability Category F) [Ref. 6.2.3]

Distance from Control Room (Miles)	Weight (lbs.) Type A Control Room	Type B Control Room	Type C Control Room
< 0.3	100	100	100
0.3 to 0.5	9,000	2,300	100
0.5 to 0.7	35,000	8,800	400
0.7 to 1.0	120,000	20,000	1,000
1 to 2	270,000	52,000	2,500
2 to 3	1,300,000	280,000	13,000
3 to 4	3,700,000	780,000	33,000
4 to 5	8,800,000	1,400,000	60,000

Type A Control Room - A tight control room having low leakage construction features and the capability of detecting at the fresh air intake those hazardous chemicals stored or transported near the site. Detection of the chemical and automatic isolation of the control room are assumed to have occurred. An air exchange rate of 0.015 per hour is assumed (0.015 of the control room air by volume is replaced with outside air in one hour). The control volume is defined as the volume of the entire zone serviced by the control room ventilation system. The assumption that the air exchange rate is less than 0.06 per hour requires verification by field testing.

Type B Control Room - Same as Type A, but with an air exchange rate of 0.06 per hour. This value is typical of a control room with normal leakage construction features. The assumption that the air exchange rate is less than 0.06 per hour requires verification by field testing.

Type C Control Room - A control room that has not been isolated, has no provision for detecting hazardous chemicals, and has an air exchange rate of 1.2 per hour.

6.2.2 References

[6.2.1] Title 10 Code of Federal Regulations, Energy (10 CFR) Parts 0 to 199, Revised as of January 1, 1987, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

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

[6.2.3] Regulatory Guide 1.78 Assumptions for Evaluating Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, U.S. AEC Directorate of Regulatory Standards. June 1974.

[6.2.4] Regulatory Guide 1.91 Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants, U.S. NRC Office of Standards Development, Revision 1. February 1978.

[6.2.5] Regulatory Guide 1.95 Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release, U.S. NRC Office of Standards Development, Revision 1. January 1977.

[6.2.6] NUREG/CR-2650, SAND82-0774, Allowable Shipment Frequencies for the Transport of Toxic Gases Near Nuclear Power Plants, David E. Bennett, David C. Heath, Sandia National Laboratories, Albuquerque, NM. October 1982.

[6.2.7] NUREG/CR-2462, SAND83-1250, Capacity of Nuclear Power Plant Structures to Resist Blast Loadings, Robert P. Kennedy, Thomas E. Blejwas, David E. Bennett, Sandia National Laboratories, Albuquerque, NM. September 1983.

6.3 TRANSPORTATION ACCIDENTS RISK BACKGROUND

6.3.1 Transportation Accident Area

The NRC Standard Review Plant (SRP) [Ref. 6.3.1] Sections 2.2.1 - 2.2.2 and 2.2.3 states that all activities within an area of radius five miles or 8.05 kilometers must be considered when choosing a location for a nuclear power plant. No basis for an area of this size is given. The five mile radius was probably based on engineering judgement given the best available information at the time it was issued. Whether an area of five mile radius is too large, of sufficient size, or too small to consider all potential transportation accidents is certainly open to debate. Reference 6.3.2 from the "Low-Probability/High-Consequence Risk Analysis" workshop held June 15-17, 1982, in Washington, D.C., by the Society for Risk Analysis, gives the following information:

> "One major source of uncertainty stems from disagreement by experts over how far an evaporated cloud of gas/air mixture could spread and still be flammable. For example, a methane/air mixture will burn if the methane is between 5.3% and 15% of the total, while an ethane/air mixture will burn if the ethane is between 3.0% and 12.5% of the total. To illustrate the practical importance of this disagreement, the following estimates have been for the distance that a gas/air cloud could disperse and still remain flammable."

Source of Estimate	Miles
Federal Power Commission	0.75
American Petroleum Institute	5.2
Cabot Corporation	11.5
U.S. Coast Guard	16.3
Prof. James Fay	17.4
U.S. Bureau of Mines	25.2 to 50.3

> "Obviously, as these estimated distances vary greatly, and as the `correct' answer must depend in any case upon such factors as wind speed, wind direction and turbulence, rate of spill, rate of evaporation, site of spill, and source and place of ignition, substantial uncertainty necessarily exists in the assessment of these potential risks. Substantial uncertainty also exists over whether such gas/air mixtures can detonate or explode, and over what the chances are that a fire ignited on the down-wind edge of a cloud can burn back to the source."

For this report, transportation accidents within five miles of a plant is used for the initial screening analysis, however, this should not be

considered a "hard-and-fast" rule or criteria. There may be situations where an area of five mile radius is not sufficient. For example, suppose that there is a railroad terminal six miles from a plant site where many trains deliver and pickup their loads, some of which are hazardous material shipments. In this case, the railroad terminal should be considered in a hazard analysis to the plant site even though it is outside of the five mile radius zone.

6.3.2 On-Site Versus Off-Site Transportation Accidents

The movement of vehicles (except for aircraft) carrying hazardous materials on-site is usually well controlled and infrequent. Also, the amount of hazardous material carried by each vehicle is usually not large. For these reasons, on-site transportation accidents of vehicles carrying hazardous materials is usually discounted in risk analysis of transportation accidents to nuclear power plants. This brings up the question, what is considered to be an on-site transportation accident and what is considered to be an off-site transportation accident?

The site boundary of a nuclear power plant could be used to differentiate between on-site accidents versus off-site accidents but a precise, readily accepted definition of site boundary could not be found. Section 100.3 of 10 CFR [6.3.3], defines the "Exclusion Area" as "that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area." It is further defined by 10 CFR 100.3 [6.3.3] that an exclusion area "may be traversed by a highway, railroad, or waterway, provided that these are not so close to the facility as to to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety." In most cases, the exclusion area corresponds to the plant site boundary, but occasionally, as in bodies of water, the exclusion area extends over the water and only the shoreline is owned by the plant licensee.

For the purposes of this report, transportation accidents within the exclusion area are considered accidents on site. Transportation accidents outside of the exclusion area but within a five mile radius of the reactor containment are considered near the plant site. Table 6.3.1 lists the minimum exclusion area radius of all currently licensed plants and plants still under construction in the U.S. The average exclusion area radius was found to be 2,828 feet or about 0.54 mile with the smallest exclusion area radius found to be 909 feet (0.17 mi.) for Vermont Yankee and the largest exclusion area radius found to be 6,440 feet (1.22 mi.) for Shearon Harris.

Since 10 CFR 100.3 [Ref. 6.3.3] allows an exclusion area to be crossed by a railroad, highway or waterway provided that certain conditions are met, several power plant sites in the U.S. were approved with railroad, highway or water transportation routes that go through its exclusion area. Some of these transportation routes either were major routes prior to approval or have

developed into major transportation routes since approval. In either case, these plant sites with transportation routes through the exclusion area may have had to perform additional safety analyses or provide additional safeguards in order to gain the approval of the NRC site licensing staff. Table 6.3.1 lists some of nuclear power plant sites in the U.S. which have transportation routes transversing the exclusion area. For some sites, not enough information could be found from available sources to confirm that all modes of transportation (i.e. pipelines) were absent from the exclusion area.

To calculate the probability of core damage due to transportation accidents, an estimate was made of the hazard presented by each mode of transportation. Factors which may affect this estimate are the amount of hazardous material carried by each transportation mode shipment, speed of each transportation mode, mode vehicle weight, transportation mode route distance to plant, etc. An initial estimate of the potential hazard presented by each transportation mode is given in Table 6.3.2. Direct collision is considered to be an actual collision by the vehicle with plant structures within the exclusion area. Accidents near the plant are considered to be transportation accidents outside of the plant exclusion area but within five miles of the reactor containment. Those hazards that were judged to be minor were not investigated further. Those hazards that were judged to be medium may need to be investigated further. Those hazards that were judged to be major were investigated further. Since pipelines cannot move, direct collision does not apply. Note that this is a strictly subjective judgment of the potential hazard represented by each transportation mode. No probabilistic analysis was done at this step. The probabilistic determination for each transportation mode will be done in their relevant sections.

Table 6.3.1

U.S. Nuclear Power Plant Exclusion Areas and Site-Specific Transportation Hazards

Site	Loc. by State	Exclusion (Meters)*	Radius (Feet)*	Site Transversed By:	Reference
Arkansas Nuclear One 1,2	AR	1,046	3,432	PL	[6.3.4]
Beaver Valley 1,2	PA	610	2,001		
Bellefonte 1,2	AL	914	2,999		
Big Rock Point	MI	817	2,680		
Braidwood 1,2	IL	457	1,500	None	[6.3.5]
Browns Ferry 1,2,3	AL	1,219	3,999		
Brunswick 1,2	NC	914	2,999	None	[6.3.6]
Byron 1,2	IL	445	1,460	None	[6.3.7]
Callaway 1	MO	1,200	3,937	HW	[6.3.8]
Calvert Cliffs 1,2	MD	1,150	3,773		
Catawba 1,2	SC	762	2,500	None	[6.3.9]
Clinton 1	IL	975	3,200	None, PL(?)	[6.3.10]
Comanche Peak 1,2	TX	1,544	5,067	PL	[6.3.11]
Donald C. Cook 1,2	MI	610	2,001		
Cooper	NB	746	2,448		
Crystal River	FL	1,340	4,396		
Davis-Besse 1	OH	635	2,083	None	[6.3.12]
Diablo Canyon 1,2	CA	800	2,625	None, PL(?)	[6.3.13]
Dresden 2,3	IL	671	2,201	None	[6.3.14]
Duane Arnold	IA	440	1,444		
Joseph M. Farley 1,2	AL	1,260	4,134		
Enrico Fermi 2	MI	915	3,002	None, PL(?)	[6.3.15]
James A. Fitzpatrick	NY	975	3,199	None, PL(?)	
Nine Mile Point 1,2		1,402	4,600	None, PL(?)	[6.3.16]
Fort Calhoun 1	NB	375	1,230		
Fort St. Vrain	CO	590	1,936		
Robert E. Ginna 1	NY	450	1,476	None	[6.3.17]
Grand Gulf 1,2	MS	695	2,280	HW	[6.3.18]
Haddam Neck	CT	530	1,739		
Edwin I. Hatch 1,2	GA	1,250	4,101	HW	[6.3.19]
Hope Creek 1	NJ	901	2,956	MR (Delaware R)	[6.3.20]
Salem 1,2		1,165	3,822		[6.3.21]
Indian Point 2,3	NY	350	1,148	PL MR (Hudson R)	[6.3.22]
Kewaunee	WI	1,200	3,937		
LaCrosse (Genoa 3)	WI	335	1,099		
LaSalle 1,2	IL	509	1,670	None	[6.3.23]

Table 6.3.1 (continued)

Site	Loc. by State	Exclusion (Meters)*	Radius (Feet)*	Site Transversed By:	Reference
Limerick 1,2	PA	762	2,500	RR, PL(?) MR (Schulykill R?)	[6.3.24]
Maine Yankee	ME	610	2,001		
William B. McGuire 1,2	NC	762	2,500	HW (NC-73)	[6.3.25]
Millstone 1,2,3	CT	736	2,415	RR	[6.3.26]
Monticello	MN	488	1,601	None	[6.3.27]
North Anna 1,2	VA	1,350	4,429		
Oconee 1,2,3	SC	1,609	5,279		
Oyster Creek 1	NJ	402	1,320	None	[6.3.28]
Palisades	MI	671	2,201		
Palo Verde 1,2,3	AZ	900	2,953	None	[6.3.29]
Peach Bottom 2,3	PA	820	2,690	None	[6.3.30]
Perry 1,2	OH	884	2,900	None, PL(?)	[6.3.31]
Pilgrim 1	MA	441	1,447		
Point Beach 1,2	WI	1,207	3,960		
Prairie Island 1,2	MN	715	2,346		
Quad Cities 1,2	IL	380	1,247		
Rancho Seco	CA	640	2,100		
River Bend 1	LA	914	3,000	RR, HW (LA-965)	[6.3.32]
Harold B. Robinson 2	SC	425	1,394		
St. Lucie 1,2	FL	1,561	5,120	HW (FL-A1A) PL (?)	[6.3.33]
San Onofre 1,2,3	CA	600	1,968	PL, RR HW (I-5)	[6.3.34]
Seabrook 1	NH	914	3,000	RR	[6.3.35]
Sequoyah 1,2	TN	556	1,824	MR (Tenn. R)	[6.3.36]
Shearon Harris 1	NC	1,963	6,440	None, PL(?)	[6.3.37]
Shoreham 1	NY	305	1,000	None	[6.3.38]
South Texas 1,2	TX	1,430	4,692	None	[6.3.39]
Virgil C. Summer 1	SC	1,630	5,348	None	[6.3.40]
Surry 1,2	VA	560	1,837		
Susquehanna 1,2	PA	549	1,800	None	[6.3.41]
Three Mile Island 1,2	PA	610	2,001	None	[6.3.42]
Trojan	OR	662	2,172	RR, MR (Columbia R) HW (US-30)	[6.3.43]
Turkey Point 3,4	FL	1,269	4,163		
Vermont Yankee	VT	277	909		
Vogtle 1	GA	1,097	3,600	None	[6.3.44]
WNP-2	WA	1,950	6,398	RR (DOE)	[6.3.45]

Table 6.3.1 (continued)

Site	Loc. by State	Exclusion (Meters)*	Radius (Feet)*	Site Transversed By:	Reference
Waterford 3	LA	914	2,999	PL, RR HW (LA-18) MR (Miss. R)	[6.3.46]
Watts Bar 1,2	TN	1,200	3,937	MR (Tenn. R)	[6.3.47]
Wolf Creek 1	KS	1,200	3,937	None, PL(?)	[6.4.48]
Yankee Rowe	MA	945	3,100		
Zion 1,2	IL	415	1,362	None	[6.3.49]
Average Exclusion Area Radius		829	2,756		
Minimum Exclusion Area Radius		277	909		
Maximum Exclusion Area Radius		1,963	6,440		

Notes:
MR = Marine Transportation Route (Body of Water)
PL = Pipeline
RR = Railroad
HW = Highway (Route No.)

*Unless otherwise specified by the Reference column, all site-specific information from Reference 6.3.50.

Table 6.3.2

Potential Hazards from Transportation Accidents to Nuclear Power Plants

	Accidents Within Plant Exclusion Area			Accidents Outside of Exclusion Area But Within Five Miles of Reactor Containment(s)	
Transportation Mode	Direct Collision with Plant Structures	Explosion, Fire Release	Hazardous Material Release	Explosion, Fire	Hazardous Material
Aviation	Major	Major	Minor	Minor	Minor
Marine	Major	Major	Major	Major	Major
Pipeline	N.A.	Major	Major	Major	Major
Railroad	Minor	Major	Major	Major	Major
Truck	Minor	Medium	Medium	Medium	Medium

N.A. = not applicable.
Minor = not investigated further.
Medium = may need to be investigated further.
Major = should be investigated further.

6.3.3 References

[6.3.1] NUREG-0800 (Formerly issued as NUREG-75/087) Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. July 1981.

[6.3.2] Low-Probability/High-Consequence Risk Analysis, Issues, Methods, and Case Studies, Ray A. Waller and Vincent T. Covello, editors, Plenum Press, New York, NY, 1984, "Catastrophic Loss Risks: An Economic and Legal Analysis and a Model State Statute", Michael B. Meyer, pg. 337-360.

[6.3.3] Title 10 Code of Federal Regulations, Energy (10 CFR) Parts 0 to 199, Revised as of January 1, 1987, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

[6.3.4] NUREG-0308 Safety Evaluation Report Related to the Operation of Arkansas Nuclear One, Unit 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. November 1977.

[6.3.5] NUREG-1002 Safety Evaluation Report Related to the Operation of Braidwood Station, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. November 1983.

[6.3.6] Brunswick Steam Electric Plant Units 1 and 2 Final Safety Analysis Report, Carolina Power & Light Company, Amendment 12. June 1972.

[6.3.7] NUREG-0876 Safety Evaluation Report Related to the Operation of Byron Station, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. February 1982.

[6.3.8] Callaway Plant Units No. 1 & 2 Addendum Standardized Nuclear Unit Power Plant System (SNUPPS) Final Safety Analysis Report, Union Electric Company (No Amendment No.). (No Date).

[6.3.9] NUREG-0954 Safety Evaluation Report Related to the Operation of Catawba Nuclear Station, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. February 1983.

[6.3.10] NUREG-1203 Technical Specifications Clinton Power Station, Unit No. 1, Docket No. 50-461, Appendix "A" to License No. NPF-55, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. September 1986.

[6.3.11] NUREG-797 Safety Evaluation Report Related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. July 1981.

[6.3.12] NUREG-0421 Safety Evaluation Report Related to Construction of Davis-Besse Nuclear Power Station, Units 2 and 3, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. July 1978.

[6.3.13] NUREG-1151 Technical Specifications Diablo Canyon Nuclear Power Plant, Units 1 and 2, Docket Nos. 50-275 and 50-323, Appendix "A" to License Nos. DPR-80 and DPR-82, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. August 1985.

[6.3.14] Dresden Nuclear Power Station Units 2 and 3 Safety Analysis Report, Commonwealth Edison Company, Amendment 22. May 7, 1970.

[6.3.15] NUREG-1141 Technical Specifications Fermi-2, Docket No. 50-341, Appendix "A" to License No. NPF-43, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. July 1985.

[6.3.16] NUREG-1193 Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, Docket No. 50-410, Appendix "A" to License No. NPF-54, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. October 1986.

[6.3.17] Robert Emmett Ginna Nuclear Power Plant Unit No. 1 Final Facility Description and Safety Analysis Report, Rochester Gas and Electric Corporation, (No Amendment No.). March 1969.

[6.3.18] NUREG-0831 Safety Evaluation Report Related to the Operation of Grand Gulf, Nuclear Station, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. September 1981.

[6.3.19] NUREG-0411 Safety Evaluation Report Related to the Operation of Edwin I. Hatch Nuclear Plant Unit No. 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. June 1978.

[6.3.20] NUREG-1186 Technical Specifications Hope Creek Generating Station, Docket No. 50-354, Appendix "A" to License No. NPF-50, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. April 1986.

[6.3.21] Salem Nuclear Generating Station Units 1 and 2 Final Safety Analysis Report, Public Service Electric and Gas Company, Amendment 27. (No Date).

[6.3.22] Indian Point 3 Nuclear Power Plant Final Safety Analysis Report Update, Power Authority of the State of New York, Revision 0. July 1982.

[6.3.23] NUREG-0519 Safety Evaluation Report Related to the Operation of LaSalle County Station Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. March 1981.

[6.3.24] NUREG-1149 Technical Specifications Limerick Generating Station, Unit No. 1, Docket No. 50-352, Appendix "A" to License No. PF-39, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. June 1985.

[6.3.25] NUREG-0422 Safety Evaluation Report Related to the Operation of McGuire Nuclear Station, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. March 1978.

[6.3.26] NUREG-1031 Safety Evaluation Report Related to the Operation of Millstone Nuclear Power Station, Unit No. 3, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. July 1984.

[6.3.27] Monticello Nuclear Generating Plant Final Safety Analysis Report, Northern States Power Company, (No Amendment No.). (No Date).

[6.3.28] Oyster Creek Nuclear Power Plant Unit No. 1 Facility Description and Safety Analysis Report, Jersey Central Power and Light Company, (No Amendment No.). November 1967.

[6.3.29] NUREG-0520 Safety Evaluation Report Related to Construction of Palo Verde Nuclear Generating Station Units 4 and 5, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. February 1979.

[6.3.30] Peach Bottom Atomic Power Station Units No. 2 & 3 Final Safety Analysis Report, Philadelphia Electric Company, (No Amendment No.). (No Date).

[6.3.31] NUREG-1162 Technical Specifications Perry Nuclear Power Plant, Unit No. 1, Docket No. 50-440, Appendix "A" to License No. NPF-45, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. March 1986.

[6.3.32] River Bend Station Final Safety Analysis Report, Gulf States Utilities, Amendment 1. October 1981.

[6.3.33] NUREG-0949 Technical Specifications St. Lucie Plant Unit No. 2, Docket No. 50-389, Appendix "A" to License No. NPF-16, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. April 1983.

[6.3.34] NUREG-0712 Safety Evaluation Report Related to the Operation of San Onofre Nuclear Generating Station, Units 2 and 3, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. February 1981.

[6.3.35] NUREG-0896 Safety Evaluation Report Related to the Operation of Seabrook Station, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. March 1983.

[6.3.36] NUREG-0011 Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. March 1979.

[6.3.37] NUREG-1208 Technical Specifications Shearon Harris Nuclear Power Plant Unit No. 1, Docket No. 50-400, Appendix "A" to License No. NPF-53, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. October 1986.

[6.3.38] NUREG-0420 Safety Evaluation Report Related to the Operation of Shoreham Nuclear Power Station, Unit No. 1, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. April 1981.

[6.3.39] NUREG-0781 Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. April 1986.

[6.3.40] NUREG-0717 Safety Evaluation Report Related to the Operation of Virgil C. Summer Nuclear Station, Unit No. 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. February 1981.

[6.3.41] NUREG-0776 Safety Evaluation Report Related to the Operation of Susquehanna Steam Electric Station, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. April 1981.

[6.3.42] NUREG-0107 Safety Evaluation Report Related to Operation of Three Mile Island Nuclear Station, Unit 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. September 1976.

[6.3.43] Safety Evaluation Report Trojan Nuclear Plant, U.S. Atomic Energy Commission, Directorate of Licensing, Washington, D.C. October 1974.

[6.3.44] NUREG-1137 Safety Evaluation Report Related to the Operation of Vogtle Electric Generating Plant, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. June 1985.

[6.3.45] WPPSS Nuclear Project No. 2 Final Safety Analysis Report, Washington Public Power Supply System, Amendment No. 1. July 1978.

[6.3.46] NUREG-0787 Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. July 1981.

[6.3.47] NUREG-0847 Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. June 1982.

[6.3.48] NUREG-1136 Technical Specifications Wolf Creek Generating Station, Unit No. 1, Docket No. STN 50-482, Appendix "A" to License No. NPF-42, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. June 1985.

[6.3.49] Safety Evaluation Report Zion Nuclear Power Station, Units 1 and 2, U.S. Atomic Energy Commission, Directorate of Licensing, Washington, D.C. October 1972.

[6.3.50] NUREG-0348 Demographic Statistics Pertaining to Nuclear Power Reactor Sites, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. October 1979.

6.4 AVIATION ACCIDENTS

The effect of an aircraft of sufficient weight, traveling at sufficient speed, crashing at a nuclear powerplant site may result in physical damage and disruption to the plant to the extent that damage to the reactor core damage and release of radioactive material from the reactor core may result. Only physical damage to the plant is considered because there is insufficient hazardous material carried by the aircraft, except for onboard fuel, to affect the plant sufficiently to ultimately cause damage to the reactor core. The fuel aboard the aircraft is considered to be covered by physical damage to plant. No sabotage or deliberate "kamikaze" crashes are considered.

6.4.1 Aviation Safety Requirements

The movement of aircraft in the United States is controlled by the Federal Aviation Administration (FAA) through Title 14 of the Code of Federal Regulations (14 CFR) [Ref. 6.4.1]. Section 121 of 14 CFR regulates commercial aviation in the United States for aircraft capable of carrying more than 30 passengers and/or a payload of more than 7,500 pounds. Section 125 of 14 CFR regulates commercial aviation in the United States for aircraft capable of carrying more than 20 but less than 30 passengers and/or a payload of more than 6,000 but less than 7,500 pounds. Section 135 of 14 CFR regulates commercial aviation in the United States for aircraft capable of carrying less than 20 passengers and/or less than a payload of 6,000 pounds. Section 91 of 14 CFR regulates general aviation, that is, all aircraft not involved in commercial operations.

6.4.2 NRC Acceptance Criteria

The U.S. NRC has issued the following in their Standard Review Plan (SRP) [Ref. 6.4.2] as their acceptance criteria for the siting of nuclear power plants near airports and/or airways. The probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is considered to be less than 10^{-7} per year if the distances from the plant meet all of the requirements listed below:

(a) The plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2,

(b) The plant is at least 5 statute miles from the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation,

(c) The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern.

The FAA, in compiling airport use statistics, defines an aircraft operation as the airborne movement of aircraft in controlled or noncontrolled airport terminal areas and about given enroute fixes or at other points where counts can be made. There are two types of operations--local and intinerant. Local operations are performed by aircraft which: (a) Operating in the local traffic pattern or within sight of the airport, (b) Are known to be departing for, or arriving from, flight in local practice areas within a 20 mile radius of the airport, and (c) Execute simulated instrument approaches or low passes at the airport. Itinerant operations are all aircraft operations other than local operations [Ref. 6.4.3].

If the above proximity criteria are not met, then a detailed review of the aircraft hazards must be performed. The SRP [Ref. 6.4.2] provides a procedure by which the probability of an aircraft crash can be calculated for various situations.

For Federal airways or aviation corridors that pass through the vicinity of a site, the probability per year of an aircraft crashing into the plant, P_{FA}, is given by the following equation:

Eq. 6.4.1 $P_{FA} = C \times N \times A/W.$

where C = inflight crash rate per mile for aircraft using airway,
N = number of flights per year along the airway,
A = effective area of plant in square miles, and
W = width of airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in miles.

For civilian and military airports and heliports, the probability of an aircraft crashing into site is given by the following equation:

Eq. 6.4.2 $PA = S_L \, S_M \, C_j \times N_{ij} \times_j A$

where S_L = summation over all flight trajectories, for i = 1 to L, affecting the site,
S_M = summation over all different types of aircraft, for j = 1 to M, using the airport,
C_j = probability per square mile of a crash per aircraft movement, for the jth aircraft,
N_{ij} = number (per year) of movements by the jth aircraft along the ith flight paths, and
A_j = effective plant area (in square miles) for the jth aircraft.

The values for C_j are given by Table 6.4.1 reproduced from Ref. 6.4.2. The data given by Table 6.4.1 for U.S. air carriers and for U.S. Navy (USN)/U.S. Marine Corps (USMC) and U.S. Air Force (USAF) aircraft was first presented by Ref 6.4.4 in 1973. According to Ref. 6.4.4, the bases for this data were aircraft accidents resulting in fatalities that occurred with a few

miles of a runway within a 60-degree reference flight path symmetric about extended centerline of the runway. The U.S. air carrier analysis was based on 80,000,000 movements. The USN/USMC and USAF analyses were based on 55,000,000 and 39,000,000 movements, respectively.

Table 6.4.1

U.S. NRC SRP Probability of Fatal Crash Versus Distance from End of Runway [Ref. 6.4.2]

Distance From End of Runway (miles)	Probability (x 10^8) of a Fatal Crash per Square Mile per Aircraft Movement			
	U.S. Air Carrier	General Aviation	USN/USMC	USAF
0-1	16.7	84	8.3	5.7
1-2	4.0	15	1.1	2.3
2-3	0.96	6.2	0.33	1.1
3-4	0.68	3.8	0.31	0.42
4-5	0.27	1.2	0.20	0.40
5-6	0	NA	NA	NA
6-7	0	NA	NA	NA
7-8	0	NA	NA	NA
8-9	0.14	NA	NA	NA
9-10	0.12	NA	NA	NA

NA indicates that data was not available for this distance.

From Ref. 6.4.2, the effective plant areas are calculated including the following: (a) A shadow area of the plant elevation unpon the horizontal plane based on the assumed crash angle for the different kinds of aircraft and failure modes, (b) A skid area around the plant, taking into account artificial berms or any other man-made and natural barriers, as determined by the characteristics of the aircraft under consideration, and (c) The areas of those safety-related structures, systems and components which are susceptible to impact or fire damage as a result of aircraft crashes.

6.4.3 Hazard to Nuclear Power Plants from Aviation Accidents

All power plant sites are exposed to aviation accident hazards to some extent due to the ability of aircraft to travel practically anywhere. Because of the increased traffic density near airports, plant sites on approaches to airports face higher exposure rate to aviation accident hazards. Generally, it is commercial aviation traffic, as opposed to general aviation, that poses the greatest hazard to nuclear power plants due to their heavier aircraft that travel that higher speeds. However, given the higher traffic density of general aviation traffic, it is not inconceivable that general aviation could pose a greater hazard given the right circumstances. Military aviation traffic could also pose a hazard to a nuclear power plant if the plant is located near a heavy aircraft base such as a bomber or transport base.

The Three Mile Island site is used as an example in an initial screening analysis to demonstrate a method by which plants could determine if further analysis is necessary in order to meet the first figure-of-merit for probability of core damage from aviation accidents.

The methodology presented in Appendix 6.A.1 is used to perform the initial screening analysis.

The Three Mile Island Units 1 and 2 nuclear power plant site in Londonderry Township, Pennsylvania, is located on Three Mile Island in the Susquehanna River about 12 miles southeast of Harrisburg. According to the Three Mile Island Unit 2 Final Safety Analysis Report (FSAR) [Ref. 6.4.5], the Harrisburg International Airport, formerly Olmstead Air Force Base, is located on the north bank of the Susquehanna River about 2 1/2 miles northwest of the site. This airport has one runway, $130^{o}/310^{o}$. The FSAR [Ref. 6.4.5] states that aircraft making their final approaches to 310° could pass near or over the site although this would not be a standard VFR (Visual Flight Rules) approach. Figure 6.1 shows the Three Mile Island Site and the distance and bearing from the site to the Harrisburg Internaltional Airport [Ref. 6.4.6].

According to the Three Mile Island 2 Safety Evaluation Report (SER) [Ref. 6.4.6], the risk was judged acceptably low for either unit provided that less than 2,400 operations per year were by aircraft in excess of 200,000 pounds, the postulated design basis aircraft. At the time of the assessment, there was one scheduled flight per day by an air carrier using a commercial aircraft in excess of 200,000 pounds and occasional use of the airport by military flights of cargo aircraft in excess of that weight.

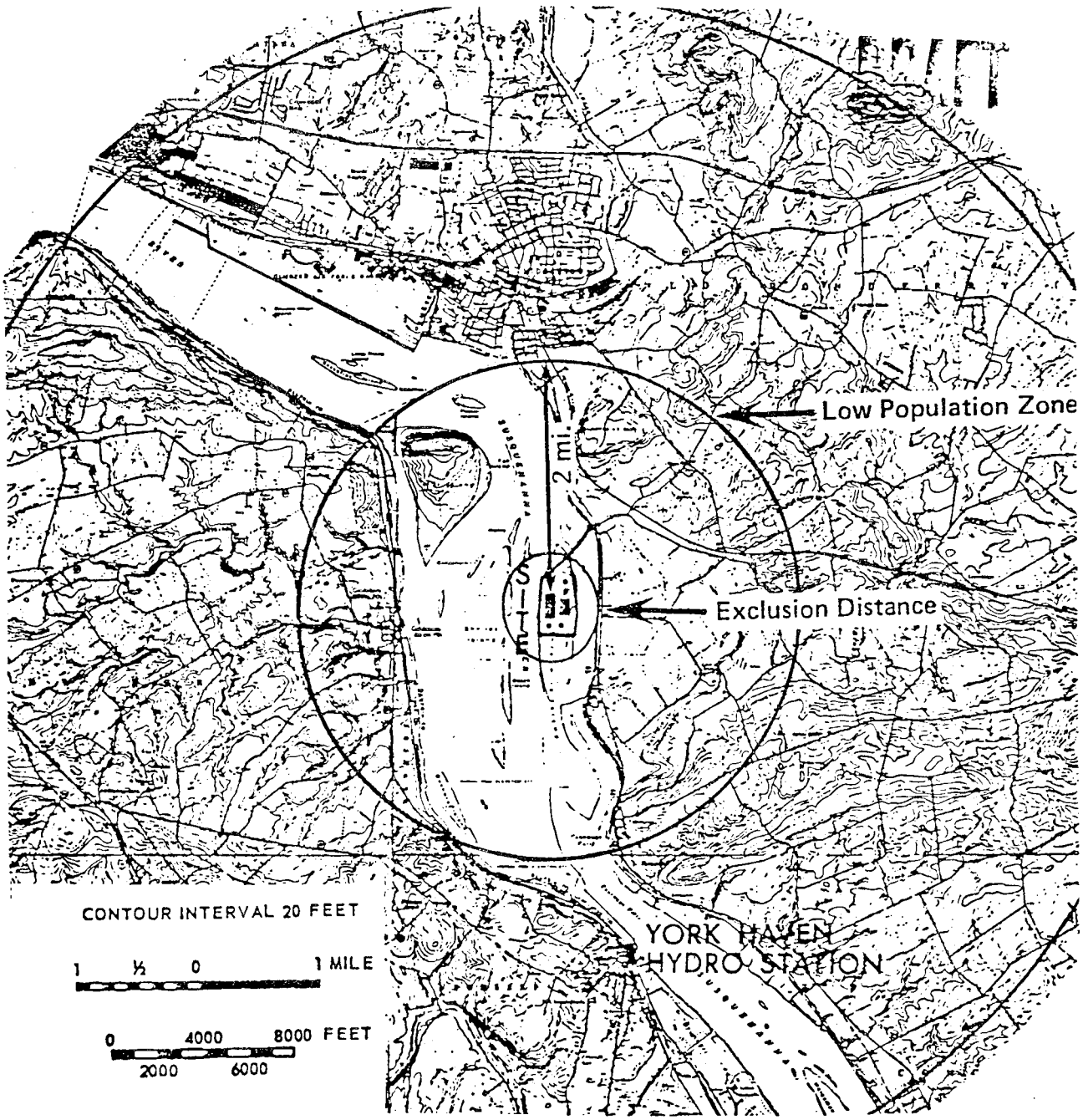


Figure 6.1

Three Mile Island Site [Ref. 6.4.6]

Since the time of that assessment, aircraft types used by U.S. commercial air carriers and traffic density has changed. As Table 6.A.2.6 of Appendix 6.A.2 shows, the earlier models of the Boeing 727, (B727-100), has a maximum takeoff weight of about 169,000 pounds. The later models of the Boeing 727, (B727-200) has a maximum takeoff weight of about 209,500 pounds [Ref. 6.A.2.14 - 6.A.2.15]. Commercial aircraft generally become heavier as new models are introduced due to "stretch-out" of the fuselage for increased passenger capacity, and higher performance engines to compensate for the increased weight and for better fuel economy.

Table 6.A.2.7 of Appendix 6.A.2 shows the type of aircraft used by U.S. commercial air carriers at the Harrisburg International Airport from 1977 to 1985. According to Table 6.A.2.7, an average of 4,687 flights departed from Harrisburg International Airport annually from 1977 to 1985. In 1984, the number of aircraft weighing over 200,000 pounds, departing from the Harrisburg International Airport was 2,075 out of a total of 4,283 for a percentage of 48.4%. In 1985, the number of aircraft weighing over 200,000 pounds, departing from the Harrisburg International Airport was 2,481 out of a total of 5,791 for a percentage of 42.8%.

Also located at the Harrisburg International Airport is the 193rd Special Operations Group of the Pennsylvania Air National Guard [Ref. 6.4.7]. This unit operates the EC-130E, an electronic warfare model of the basic C-130 Hercules, a 4-engined turboprop cargo aircraft. This aircraft has a gross operating weight of 175,000 pounds. The basic design of the C-130 was first developed 36 years ago so its potential replacement by newer, more advanced and heavier aircraft should not be ignored.

Because of the deregulation of the commercial airline industry in 1978, the general trend has been an increase in traffic density. Even small airports, such as Harrisburg, have experienced a general upward trend in commercial airline traffic. Therefore, as 1985 is the most recent year in which data is available for Harrisburg International Airport, it should be most indicative of future operations.

Using 6000 departures per year from the Harrisburg International Airport and assuming that each departure implies a landing and takeoff, there will be 6000 flights that will fly near the Three Mile Island Site each year. If an aircraft lands on runway $130^{o}/310^{o}$ towards the Northwest (310°), it will probably land from the Southeast which implies a landing approach near the Three Mile Island site. If an aircraft lands towards the Southeast (130°), it will probably take off towards the Southeast which implies a takeoff pattern near the Three Miles Island site.

The Harrisburg International Airport is 2.5 miles from the Three Mile Island site. Assuming only commercial airline accidents within 2.5 miles of the plant site can affect the site, then 6000 flights per year times 5 miles (the flight path 2.5 miles northeast and southwest of the Three Mile Island site) gives 30,000 aircraft miles per year within 2.5 miles of the Three Miles Island site. From Table 6.A.2.2 of Appendix 6.A.2, a commercial aircraft

accident rate (aircraft operating under 14 CFR 121, 125 and 127) of 7.7×10^{-9} is obtained. Multiplying the aircraft miles within 2.5 miles of the Three Mile Island site with the commercial aircraft accident rate gives:

$$30{,}000 \text{ ac miles} \times 7.7 \times 10^{-9} \frac{\text{ac accidents}}{\text{ac miles year}} = 2.3 \times 10^{-4} \frac{\text{accidents}}{\text{year}}$$

Assuming half of the commercial aircraft flights departing the Harrisburg International Airport have an operating weight of 200,000 pounds or greater, then the frequency of a commercial aircraft weighing 200,000 pounds or more having an accident within 2.5 miles of the Three Mile Island site is:

$$1/2 \times (2.3 \times 10^{-4} \frac{\text{accidents}}{\text{year}}) = 1.2 \times 10^{-4} \frac{\text{accidents}}{\text{year}}$$

6.4.4 Power Plant Response to Aviation Accidents

Since the NRC regulations regarding aviation hazards to nuclear power plants are only partly probabilistic in nature and do not relate to core damage or large release frequency, to obtain a probabilistic estimate of the frequency of core damage due to aviation accidents, one must turn to probabilistic risk analysis. Unfortunately, the few PRAs that have considered aviation accidents (Indian Point, Millstone 3, Seabrook, Zion [Ref. 6.4.9 to 6.4.12]) have dismissed aviation accidents on the basis of the aviation accident frequency.

The only probabilistic analysis of a power plant's response to an aircraft crash is a 1971 paper by Chelapati, Kennedy and Wall [Ref. 6.4.8] which modeled aircraft engines as projectiles impacting the plant walls. The aircrafts were divided into two categories, small aircraft and large aircraft.

For small aircraft (less than or equal to 12,500 pounds in weight), the aircraft engines were idealized as projectiles ranging in weight from 230 to 800 pounds with the relative distribution of aircraft engine weight determined from aircraft census. Within five miles of an airport, small aircraft engines were modeled with an impact velocity ranging from 67 to 105 miles per hour. Beyond five miles from an airport, small aircraft engines were modeled with an impact velocity ranging from 67 to 280 miles per hour.

For large aircraft (greater than 12,500 pounds in weight), the aircraft engines were idealized as projectiles ranging in weight from 450 to 4200 pounds with the relative distribution of aircraft engine weight determined from aircraft census. Within five miles of an airport, large aircraft engines were modeled with an impact velocity ranging from 95 to 185 miles per hour. Beyond five miles from an airport, large aircraft engines were modeled with an impact velocity ranging from 175 to 610 miles per hour.

From the distribution of aircraft engine weight, impact velocity, and wall thickness, a probability of wall penetration was obtained. Table 6.4.2 presents the probability of wall penetration for various combinations of aircraft weight, wall thickness and plant location. Note that the frequency of core damage or large release was not calculated.

The assumption that large aircraft will impact with a velocity of less than 185 miles per hour within five miles of an airport is probably reasonable. Federal regulations [14 CFR 91.70, Ref. 6.4.1] control the maximum airspeed of all aircraft below 10,000 feet MSL (mean sea level). The requirements are:

"(a) Unless otherwise authorized by the Administrator, no person may operate an aircraft below 10,000 feet MSL at an indicated airspeed of more than 250 knots (288 m.p.h.).

(b) Unless otherwise authorized or required by ATC, no person may operate an aircraft within an airport traffic area at an indicated airspeed of more than-
(1) In the case of a reciprocating engine aircraft, 156 knots (180 m.p.h.); or
(2) In the case of a turbine-powered aircraft, 200 knots (230 m.p.h.)."

The regulations state further that:

"(c) No person may operate an aircraft in the airspace underlaying a terminal control area, or in a VFR corridor designated through a terminal control area, at an indicated airspeed of more than 200 knots (230 m.p.h.)."

Beyond five miles from an airport, the aircraft impact velocity in an aircraft crash is not easily determined since this speed is not as tightly regulated and terminal control areas vary in their control radius.

Table 6.4.2

Probability of Penetration as a Function of
Plant Location and Concrete Thickness [Ref. 6.4.10]

Plant Location	Aircraft Type	Probability of Penetration: Thickness of Reinforced Concrete			
		1 foot	1.5 feet	2 feet	6 feet
<= 5 miles from airport	Small, <= 12,500 lbs.	0.003	0	0	0
	Large, < 12,500 lbs.	0.96	0.52	0.28	0
>= 5 miles from airport	Small, <= 12,500 lbs.	0.28	0.06	0.01	0
	Large, > 12,500 lbs.	1.0	1.0	0.84	0.32

< = defined as less than or equal to.
> = defined as greater than or equal to.

6.4.5 Aviation Accident References

[6.4.1] Title 14 Code of Federal Regulations, Aeronautics and Space (14 CFR) Parts 60 to 139, Revised as of January 1, 1987, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

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

[6.4.3] FAA Statistical Handbook of Aviation, Calender Year 1982, U.S. Department of Transportation, Federal Aviation Administration, Office of Management Systems, Information Analysis Branch, Washington, D.C. December 31, 1982.

[6.4.4] "Reactor Siting in the Vicinity of Airfields", D.G. Eisenhut, Transactions of the American Nuclear Society, 1973 Annual Meeting, Chicago, Illinois, June 10-14, 1973, Volume 16, pg. 210-211.

[6.4.5] Three Mile Island Unit 2 Final Safety Analysis Report (FSAR), Metropolitan Edison Company.

[6.4.6] NUREG-0107 Safety Evaluation Report Related to Operation of Three Mile Island Nuclear Station, Unit 2, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. September 1976.

[6.4.7] "USAF Almanac", Air Force Magazine, Vol. 70, No. 5. May 1987.

[6.4.8] "Probabilistic Assessment of Aircraft Hazard for Nuclear Power Plants", C.V. Chelapati, R.P. Kennedy, and I.B. Wall, Nuclear Engineering and Design, Vol. 19, pg. 333-364. 1972.

[6.4.9] Indian Point Probabilistic Safety Study, Consolidated Edison Company and Power Authority of State of New York, Pickard, Lowe & Garrick, Inc. 1983.

[6.4.10] Millstone Unit 3 Probabilistic Safety Study, Section, Part 1, Volume 2. August 1983.

[6.4.11] Seabrook Station Probabilistic Safety Assessment, Public Service Company of New Hampshire and Yankee Atomic Electric Company, Pickard, Lowe & Garrick, Inc. Report PLG-0300. 1983.

[6.4.12] Zion Probabilistic Safety Study, Commonwealth Edison Company, Pickard, Lowe & Garrick, Inc. 1982.

6.5 MARINE (SHIP/BARGE) ACCIDENTS

Marine accidents (considered as accidents involving both ship and/or barge) pose a hazard to a nuclear power plant due to the possible release of hazardous material towards the plant and/or the possibility of explosion and fire with resulting physical damage to the plant due to blast, debris and fire. There is also the possibility of physical damage to the cooling water intake and outlet structures due to collision by the ship or barge. Sabotage and deliberate crashes are not considered in this subsection.

6.5.1 Marine Safety Regulations

According to Ref. 6.5.1, the Coast Guard Department of the U.S. Department of Transportation regulates all bulk transport by water. Requirements for the design, construction, equipment, maintenance, and inspection of commercial vessels, including those used for bulk hazardous material shipments are contained in 46 CFR Subchapter D for tank vessels, parts 30-40 [Ref. 6.5.3], 46 CFR Subchapter I for cargo and miscellaneous vessels, parts 90-106 [Ref. 6.5.4], 46 CFR Subchapter N for dangerous cargoes, parts 146-149 [Ref. 6.5.5], and 46 CFR Subchapter O for certain bulk dangerous cargoes, parts 150-155 [Ref. 6.5.5]. Additional requirements for certain ships and barges that carry bulk oil shipments are prescribed in 33 CFR part 157 [Ref. 6.5.2]. Coast Guard requirements for dangerous cargo require vessels to notify the appropriate captain of the port in advance of arrivals and departures.

Part 4.05 of 46 CFR [Ref. 6.5.3] defines a notice of marine casualty as whenever any of the following occur:

a) All accidental groundings and any intentional grounding which also meets any of the other reporting criteria or creates a hazard to navigation, the environment, or the safety of the vessel;
b) Loss of main propulsion or primary steering, or any associated component or control system, the loss of which causes a reduction of the manuevering capabilities of the vessel. Loss means that systems, component parts, sub-systems, or control systems do not perform the specified or required function;
c) An occurrence materially and adversely affecting the vessel's seaworthiness or fitness for service or route, including but not limited to fire, flooding, or failure or damage to fixed fire extinguishing systems, lifesaving equipment, auxiliary power generating equipment, or bilge pumping systems;
d) Loss of life;
e) Injury causing a person to remain incapacitated for a period in excess of 72 hours;
f) An occurrence not meeting any of the above criteria but resulting damage to property in excess of $25,000. Damage cost includes the cost of labor and material to restore the property to the service condition which

existed prior to the casualty, but does not include the cost of salvage, cleaning, gas freeing, drydocking or demurrage.

6.5.2 Hazard to Nuclear Power Plants from Marine Accidents

According to Ref. 6.5.1, a billion tons of cargo were shipped by marine transport in 1982. Hazardous commodity shipments constituted 55% of the total or 549 million tons. The flows are concentrated with the 10 top region-to-region flows accounting for 65% of the total tonnage. The pattern of flows follows the distribution of petroleum, since 85% of the hazardous tonnage is crude or processed petroleum. From 1977 to 1982, the tonnage increased less than 4%. However, the commodity mix has changed significantly. Chemical shipments dropped 13% and petroleum products dropped 22%, while the "all other" category doubled.

Only power plant sites with large waterways with ship and/or barge traffic that go through or are near the power plant exclusion are (see Section 6.3.2 for further discussion of exclusion area and off-site transportation accidents) and also carry hazardous material are exposed to the hazards of marine accidents.

The Waterford 3 nuclear power plant site in St. Charles Parish, Louisiana near the town of Taft, will be used as an example of an initial screening analysis to demonstrate a method by which plants could determine if further analysis will be necessary in order to meet the first figure-of-merit for core damage frequency.

According to the Waterford Safety Evaluation Report (SER) [Ref. 6.5.7], the exclusion area boundary extends across the Mississippi River to the opposite (east) bank. The Waterford reactor building is about 1,200 ft. from the Mississippi River, one of the major inland water shipping routes in the United States. The centerline of the main shipping channel is about 1,000 ft. from the shoreline so the main shipping channel is about 2,200 ft. from the Waterford 3 reactor building. Figure 6.2 shows the Waterford site and the distance from the reactor containment to the Mississippi River [Ref. 6.5.6].

From the Waterford Final Safety Analysis Report (FSAR) [Ref. 6.5.6] 117,092 vessels traveled the section of the Mississippi River between Baton Rouge and New Orleans, Louisiana in 1970. In 1975, 161,750 vessels moving 201,600,768 tons of cargo and 10,462 passengers traveled the section of the Mississippi River between Baton Rouge and New Orleans, Louisiana. The Waterford plant is located upstream of New Orleans and downstream of Baton Rouge at river mile 129.6. Table 6.5.1 shows that in 1970, 113,183 trips or about 96.6% of the total marine trips past the Waterfor site involved vessels with a draft of 18 feet or less. Table 6.5.2 shows that in 1975, 156,123 trips or about 96.5% of the total marine trips past the Waterford site involved vessels with a draft of 18 feet or less. From Table 6.5.1, in 1975, a total of 3,909 trips or about 3.4% of the total marine trips past the Waterford site involved vessels with draft of 19 feet or more. From Table 6.5.2, in 1975, a total of 5,627 trips or about 3.5% of the total marine trips

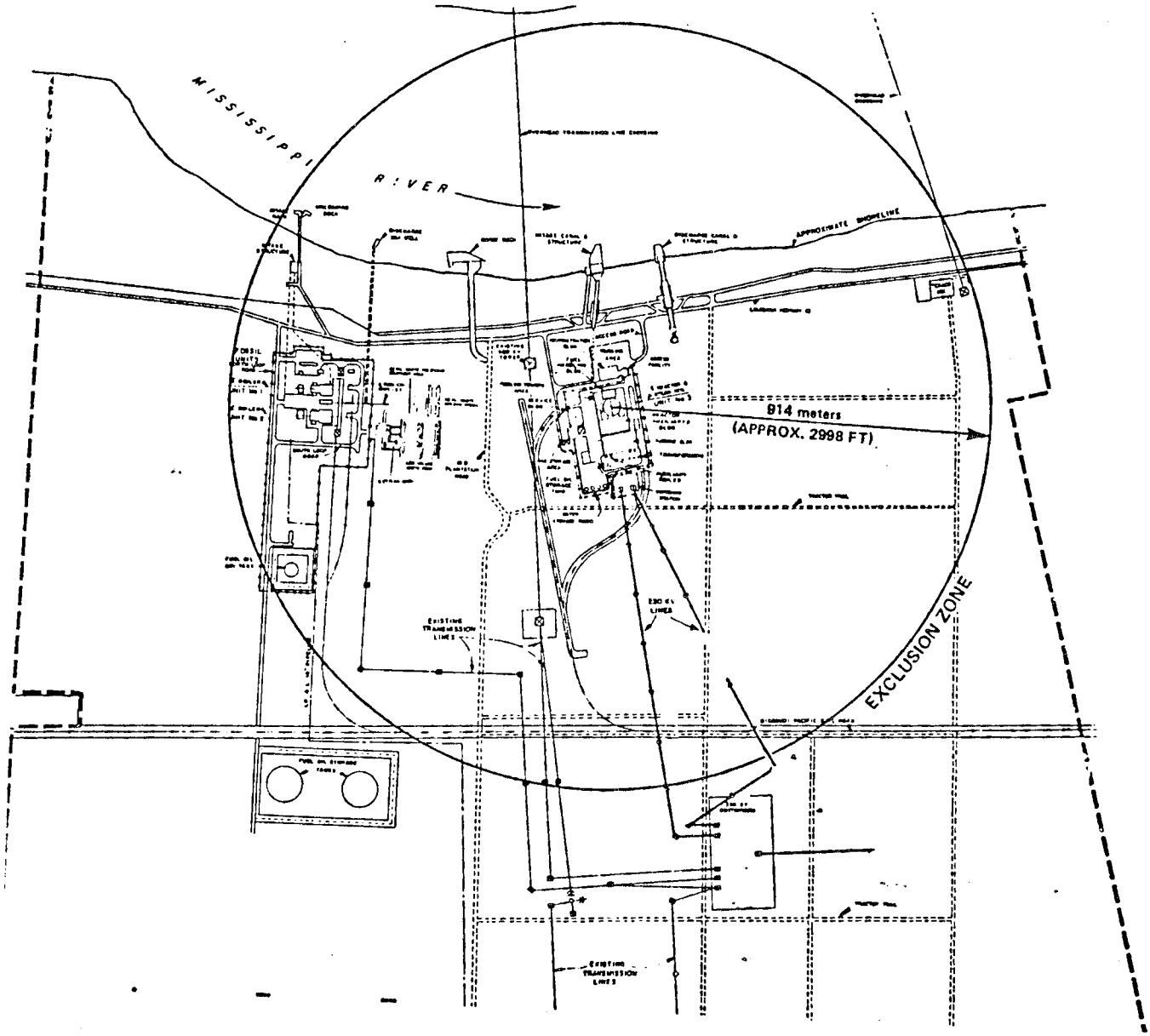


Figure 6.2

Waterford Site [Ref. 6.5.6]

past the Waterford site involved vessels with draft of 19 feet or more. Tables 6.5.1 and 6.5.2 show that at least for the period 1970 to 1975, although the total number of vessels traveling past the Waterford site on the Mississippi River increased, the proportion of vessels with draft greater than 18 feet did not change. Draft is defined as the distance from the water level to the lowest point of the vessel underwater. It is measured when the vessel is stopped, either tied to a dock or at anchor. It should be noted that the draft of a vessel can change depending on the extent to which it is loaded. A vessel that is fully loaded would obviously have a deeper draft than the same vessel unloaded. It should also be noted that no correlation could be found between vessel draft and vessel capacity. Such a correlation would allow an estimate of the number of vessels capable of carrying an amount of cargo equivalent to the design basis accident.

According to the Waterford FSAR [Ref. 6.5.6], during the period between 1972 and 1976, 62 commercial vessel casualties were recorded by the U.S. Coast Guard on the Mississippi River between River Mile 115 to 135. The Waterford plant site is at River Mile 129.6. Fifty three of the 62 accidents involved some type of freighter, dry cargo barge or liquid tank barge. Eight had commodities specified such as liquid caustic soda, ammonia, styrene, adiponitrile, sulfuric acid, etc., which could be a potential hazard to the Waterford plant. The remaining accidents either did not involve any cargo or the cargo was not specified. One accident was recorded as being due to an explosion and/or fire involving liquid bulk cargo and was caused by improper safety precautions in either loading inflammable liquid or fueling or repairs.

Table 6.5.1

Vessel Trips and Draft, Between Baton Rouge and New Orleans, Louisiana, 1970 [Ref. 6.5.6]

Draft (feet)	Self-Propelled Vessels			Non-Self Propelled Vessels			
	Pass. & Dry Cargo	Tanker	Towboat or Tugboat	Dry Cargo	Tanker	Total	Pct.
< 18	588	292	18,319	55,324	38,660	113,183	96.66%
19-20	327	197	0	3	1	528	0.45%
21-22	362	162	0	5	0	529	0.45%
23-24	211	165	0	6	0	382	0.33%
25-26	212	158	0	49	0	419	0.36%
27-28	136	142	0	5	0	283	0.24%
29-30	231	162	0	41	0	434	0.37%
31-32	225	173	0	0	13	411	0.35%
33-34	160	157	0	0	0	317	0.27%
35-36	157	118	0	0	0	275	0.23%
37-38	179	57	0	0	0	236	0.20%
39-40	33	62	0	0	0	95	0.08%
Total	2,821	1,845	18,319	55,433	38,674	117,092	100.00%

Draft defined as distance from water level to lowest point of vessel underwater. Measured when vessel is stopped, either tied to a dock or at anchor.

Table 6.5.2

Vessel Trips and Draft, Between Baton Rouge and New Orleans, Louisiana, 1975 [Ref. 6.5.6]

Draft (feet)	Self-Propelled Vessels			Non-Self Propelled Vessels			
	Pass. & Dry Cargo	Tanker	Towboat or Tugboat	Dry Cargo	Tanker	Total	Pct.
< 18	672	282	24,781	81,986	48,402	156,123	96.52%
19-20	262	182	128	15	21	608	0.38%
21-22	290	268	3	3	12	576	0.36%
23-24	249	391	18	19	11	688	0.43%
25-26	229	384	4	58	24	699	0.43%
27-28	141	189	3	8	3	344	0.21%
29-30	177	205	4	94	1	481	0.30%
31-32	160	159	1	5	0	325	0.20%
33-34	222	225	0	3	1	451	0.28%
35-36	285	261	0	0	16	562	0.35%
37-38	185	257	0	0	11	453	0.28%
39-40	132	307	0	0	1	440	0.27%
Total	3,004	3,110	24,942	82,191	48,503	161,750	100.00%

Draft defined as distance from water level to lowest point of vessel underwater. Measured when vessel is stopped, either tied to a dock or at anchor.

6.5.3 Power Plant Response to Marine Accidents

According to the Waterford SER [Ref. 6.5.7], potential ship or barge impact on the Waterford 3 cooling water intake structure is not a safety hazard. The intake structure is only needed for normal operation. Safety-related plant cooling is provided by wet and dry cooling towers which are located in the nuclear plant island structure away from the Mississippi River. The possibility of a plant transient caused by the loss of normal cooling water was not reviewed.

According to the Waterford SER [Ref. 6.5.7], the analyzed accident scenario for marine accidents was the detonation of a 300,000 barrel barge filled with gasoline. It was estimated that the detonation of the gasoline-air mixture filling the free volume of a 300,000 barrel gasoline tanker would produce at the plant a peak reflected overpressure of 2.7 psi and that this was an acceptable overpressure for the safety-related buildings. No analysis could be found as to the probability of core damage given a peak overpressure of 2.7 psi on safety-related buildings.

6.5.4 Marine Accident References

[6.5.1] OTA-SET-304 Transportation of Hazardous Materials, U.S. Congress, Office of Technology Assessment, U.S. Government Printing Office, Washington, D.C. July 1986.

[6.5.2] Title 33 Code of Federal Regulations, Navigation and Navigable Waters (33 CFR) Parts 1 to 199, Revised as of July 1, 1986, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

[6.5.3] Title 46 Code of Federal Regulations, Shipping (46 CFR) Parts 1 to 40, Revised as of October 1, 1986, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

[6.5.4] Title 46 Code of Federal Regulations, Shipping (46 CFR) Parts 90 to 139, Revised as of October 1, 1986, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

[6.5.5] Title 46 Code of Federal Regulations, Shipping (46 CFR) Parts 140 to 155, Revised as of October 1, 1985, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

[6.5.6] Waterford Steam Electric Station, Unit No. 3 Final Safety Analysis Report, Louisiana Power & Light Company, No Amendment No. (No Date).

[6.5.7] NUREG-0787 Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. July 1981.

6.6 PIPELINE ACCIDENTS

Pipeline accidents could pose a hazard to a nuclear power plant due to the release of hazardous material towards the plant and/or the possibility of explosion and resulting physical damage to a plant due to blast and debris. No sabotage is not considered in this subsection. The hazard to a power plant from a nearby pipeline is particularly acute due to the large amount of material that can be carried by a large diameter high-pressure pipeline.

6.6.1 Pipeline Safety Regulations

According to Ref. 6.6.1, the Gas Pipeline Safety Act of 1968 authorizes the Secretary of the Department of Transportation to establish federal safety standards for the transportation of natural gas by pipeline. The actual authorization for pipeline construction and natural gas pricing is issued by the Federal Power Commission (FPC).

6.6.2 Hazard to Nuclear Power Plants from Pipeline Accidents

Only power plant sites with pipelines going through or near the power plant exclusion area (see Section 6.3.2 for further discussion of exclusion area and off-site transportation accidents) are exposed to pipeline accident hazards.

The Indian Point site in Westchester County, New York near the town of Buchanan, will be used as an example of an initial screening analysis to demonstrate a method by which plants could determine if further analysis will be necessary in order to meet the first figure-of-merit for core damage frequency.

According to the Indian Point PRA [Ref. 6.6.2], there are two natural gas transmission lines passing through the site about 400 feet from the nearest Unit 3 plant structure and 1,000 feet from the Unit 2 plant structures. Both pipelines, one of 26-inch outside diameter (OD) and the other of 30-inch OD, contain relief valves set at 750 psi at some distance from the plant, and normally operate at a maximum of 650 psi. Figure 6.3 shows the Indian Point site and the distance from the reactor containments to the pipeline easement [Ref. 6.6.4]

Ref. 6.6.2 further states that about 500 transmission and gathering pipeline accidents occur annually in the U.S. and that there are about 280,000 miles of transmission pipelines in the U.S. Table 6.A.4.2 of Appendix 6.A.4 confirms the miles of transmission gas pipelines in the U.S. Assuming that the gas pipeline leak rate applies to transmission and gathering pipelines equally and that 450 accidents occured in transmission lines (as assumed in the Indian Point PRA [Ref. 6.6.2] gives a gas pipeline accident rate of:

$$\frac{450 \text{ accidents/year}}{280{,}000 \text{ miles}} = 1.6 \times 10^{-3} \text{ accidents/mi.yrs.}$$

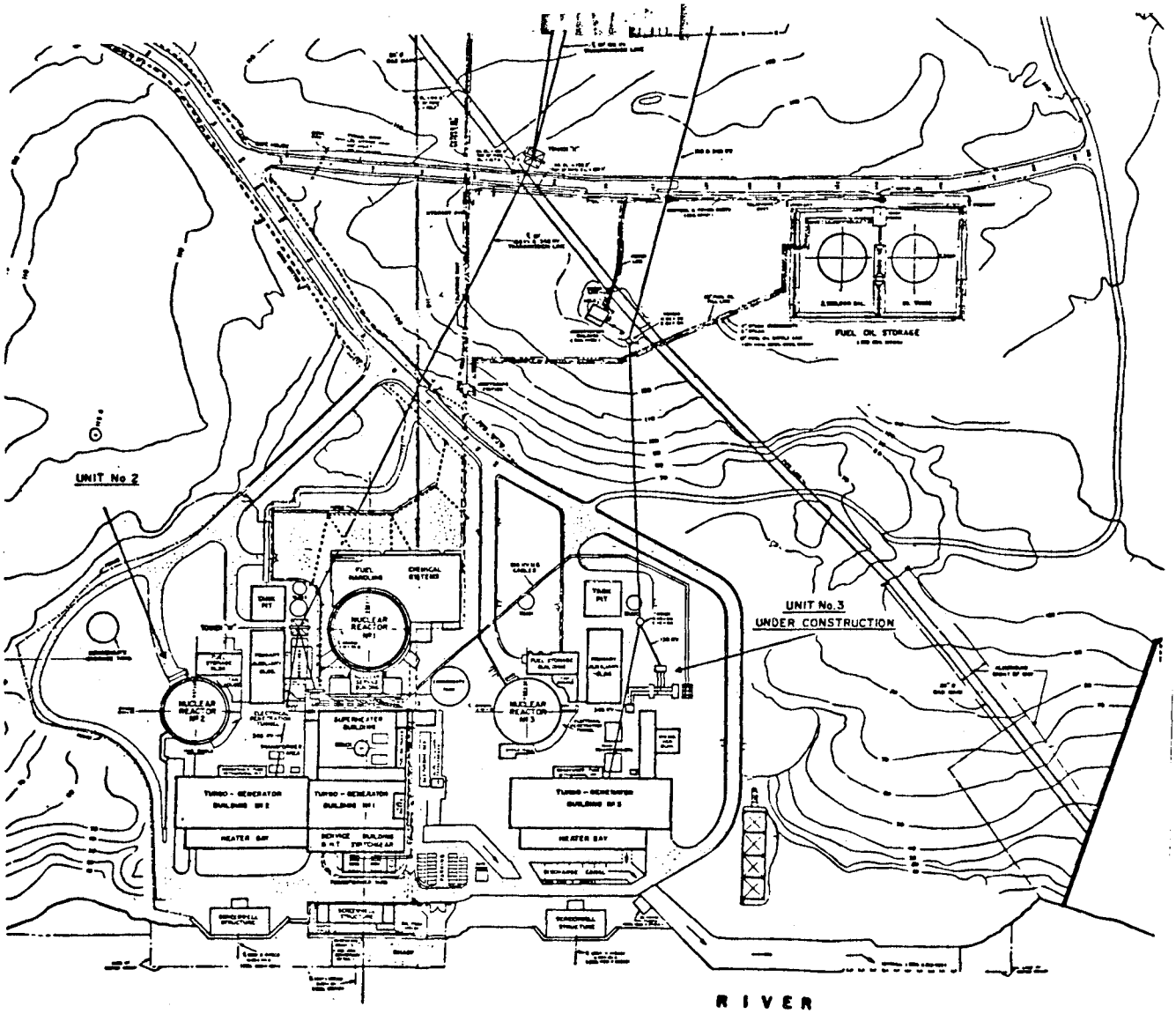


Figure 6.3

Indian Point Site [Ref. 6.6.4]

From Ref. 6.6.3, a gas pipeline leak rate of 0.23 per 1000 km-years for gas pipeline systems in the United Kingdom is quoted. This leakage rate is also quoted to be half the leak rate for gas pipelines in the United States. Converting this leak rate for the U.S. gives:

$$\left(\frac{0.23}{1000 \text{ km yrs}}\right) \times 2 \times \left(\frac{1 \text{ km}}{0.62137 \text{ mi.}}\right) = 7.4 \times 10^{-4} \text{ leaks/mi.yrs.}$$

The definition of what constitutes a leak is not given nor is the type of pipeline defined that this leak rate applies to.

Since the gas pipeline accident rate from the Indian Point PRA [Ref. 6.6.2] is greater than the gas pipeline leak rate from the U.K. paper [Ref. 6.6.3], it can be considered more conservative and will be used for this screening analysis.

The methodology presented in Appendix 6.A.1 is used to perform the initial screening analysis. The gas pipelines is about 400 feet from the plant so D is set to be zero. Since pipelines cannot move, the use of the term `vehicle miles' is obviously inappropriate. Ref. 6.6.3 defines the term `interaction length', which is the length of pipeline over which failures of a given size and type could affect a specified area, such as a town or a plant. (Implicit in this analysis is the interchangability of the terms `leak', `failure', and `accidents'. In an actual analysis, these terms will have to be more rigidly defined). Pipeline failure rates are estimated on a per length basis so the frequency of a pipeline failure may be calculated by multiplying the failure rate by the interaction length. Using 10 miles as the interaction length (5 mile radius from the Indian Point site), gives a pipeline failure frequency (within 5 miles of the Indian Point site) of:

$$1.6 \times 10^{-3} \text{ accidents/mi.yr.} \times 10 \text{ miles} = 1.6 \times 10^{-2} \text{ accs./yr.}$$

6.6.3 Power Plant Response to Pipeline Accidents

The Indian Point PRA [Ref. 6.6.3] dismisses the hazard from pipeline accidents on the basis of the frequency of the initiating event. Taken into consideration was the leakage detection program of the local gas transmission pipeline company, the possibility of gas pipeline isolation, and the failure of gas to ignite. A frequency of 5.0×10^{-7} was assigned for a gas pipeline fire which "threatens" the plant.

6.6.4 Pipeline Accident References

[6.6.1] Energy Technology Handbook, edited by Douglas M. Considine, McGraw-Hill Book Company, New York, NY, 1977.

[6.6.2] Indian Point Probabilistic Safety Study, Consolidated Edison Company and Power Authority of State of New York, Pickard, Lowe & Garrick, Inc. 1983.

[6.6.3] Low-Probability/High Consequence Risk Analysis, Issues, Methods, and Case Studies, Ray A. Waller and Vincent T. Covello, editors, Plenum Press, New York, NY, 1984, "State-of-the-Art of Risk Assessment of Chemical Plants in Europe", R.A. Cox and D.H. Slater, pg. 257-283.

[6.6.4] Indian Point 3 Nuclear Power Plant Final Safety Analysis Report Update, Power Authority of the State of New York, Revision 0. July 1982.

6.7 RAILROAD ACCIDENTS

Railroad accidents pose a hazard to a nuclear power plant due to the possible release of hazardous material towards the plant and/or the possibility of explosion and fire with resulting physical damage to the plant due to blast, debris and fire. Physical damage to plant due to actual collision with plant structures is considered minimal due to the distance between main rail lines and plant structures. The movement of trains within the plant exclusion areas is infrequent and controlled except for those plant sites that have main rail lines going through them. Except for those plant sites with main rail lines going through the exclusion area, only a limited amoung of hazardous material is carried in each shipment. Sabotage and deliberate crashes are not considered in this subsection.

6.7.1 Railroad Safety Regulations

Hazardous materials regulations for rail transport appear in Title 49 of the Code of Federal Regulations, Section 174 (49 CFR 174) [Ref. 6.7.1]. Reference 6.7.2 describes the regulations as containing general operating, handling, and loading and unloading requirements, as well as detailed requirements for various hazard classes. There are specific requirements for segregating hazardous materials in a car and for the placement of cars containing certain types of material. Carriers are also instructed to forward shipments of hazardous materials within 48 hours after acceptance at the originating point, or receipt at any yard, transfer station, or interchange point. Special loading and bracing requirements are provided, for container-on-flatcar, trailer-on-flatcar, and portable tanks and procedures for unloading tank cars are also specified.

According to Reference 6.7.2, the Federal Railroad Administration (FRA) enforces regulations pertaining to the transportation of hazardous materials by rail, including those governing the manufacture and maintenance of tank cars used to ship hazardous materials. Additionally, the FRA has jurisdiction over all areas of rail safety such as track maintenance, equipment standards, and operating practices. Rail safety regulations are published in 49 CFR Parts 209 to 236.

6.7.2 Hazard to Nuclear Power Plants from Railroad Accidents

The current NRC regulations regarding hazards to nuclear power plants from railroad accidents, as listed in Sections 2.2 and 6.2, give guidance on shipment frequency, control room design assumptions, explosion overpressure, etc., but do not relate these factors to a probabilistic determination of core damage. Because core damage probability from railroad accidents is not related to the licensing guidance for nuclear power plant siting, the risk analysis method given in Appendix 6.A.1 is used to determine how a currently licensed operating plant would meet the first figure-of-merit of 10^{-5}/yr. for

core damage frequency.

Only power plant sites with railroad lines that go through or are near the power plant exclusion area (see Section 6.3.2 for further discussion of exclusion area and off-site transportation accidents) and also carry large amounts of hazardous material traffic are exposed to the hazards of railroad accidents. Note that this is a conditional "and" statement, that is, a plant may have a railroad line through or near the plant exclusion area but if it carries little hazardous material traffic, it is of little concern. Conversely, a railroad line that carries large volumes of traffic, much of which may consist of hazardous material shipments, but is far enough away, may not be of concern. It is those unique situations of a railroad line carrying large volumes of hazardous material shipment traffic through or near a power plant exclusion area which are of concern.

The Waterford 3 nuclear power plant site in St. Charles Parish, Louisiana near the town of Taft, will be used as an example in an initial screening analysis to demonstrate a method by which plants could determine if further analysis will be necessary in order to meet the first figure-of-merit for core damage frequency.

According to the Waterford Safety Evaluation Report (SER) [Ref. 6.7.3], the Waterford site exclusion area is traversed by a Missouri Pacific Railroad line that runs about 2,400 ft. southwest of the reactor building. The line services all the chemical plants on the west bank of the Mississippi River. On the average, 18 trains per day, each train averaging about 100 cars, pass the site, shipping products such as caustic soda, chlorine, and various industrial chemicals. There are several other railroad lines near the plant, the closest being 2.25 miles. See Figure 6.2 in Section 6.5.2 for layout of Waterford site.

The methodology outlined in Appendix 6.A.1 is used to perform the initial screening analysis. The Missouri Pacific Railroad line is about 0.5 miles from the plant, so D = 0.5 miles. Using Table 6.A.1.2 with 18 shipments (considering each train as one shipment) per day, the total vehicle (train) miles per year is about 6.9×10^4. From Table 6.A.5.1 in Appendix 6.A.5, a train accident rate of 10.1×10^{-6} accidents/train mile is obtained. Multiplying these two numbers together gives the frequency of a railroad accident within five miles of the plant site per year on this particular line is:

$$(6.9 \times 10^4) \times (10.1 \times 10^{-6}) = 0.70 \text{ train accidents/year.}$$

From Table 6.A.5.3 of Appendix 6.A.5, it was found that 36,988 cars carrying hazardous material cargo were in trains involved in accidents from 1975 through 1984. 1,452 cars carrying hazardous materials in these trains were damaged and released hazardous materials in these accidents. The probability of release of hazardous materials from a train accident is then:

$$1{,}452 \;/\; 36{,}988 = 0.0393$$

This is assuming that if any car carrying hazardous material releases hazardous material, there is a release from the train accident. Assuming that all of the cars of the 18 trains that pass by the Waterford plant daily carry hazardous materials, probably a conservative assumption, then the frequency of an railroad accident within five miles of the Waterford plant releasing a hazardous material on this particular line is:

$$0.70 \text{ accidents/year} \times 0.0393 = 2.74 \times 10^{-2}/\text{year}.$$

If the distribution of cars carrying nonhazardous cargoes to cars carrying hazardous cargoes that travel this railroad line were known, then this frequency could be reduced by the ratio of cars carrying hazardous cargoes to the total number of cars.

Similar calculations should be done for the other railroad lines that pass within five miles of the Waterford plant. The total frequency for a train accident within five miles of the Waterford plant releasing hazardous materials is then, the summation over all the railroad lines of the frequency for each of the railroad lines.

6.7.3 Power Plant Response to Railroad Accidents

In the case of Waterford, assuming that the Missouri Pacific Railroad line was the predominant risk contributor for railroad accidents, then in order to meet the first figure-of-merit of 1×10^{-5}/yr. for core damage, the probability of core damage of less than 10^{-3} given a train accident within 5 miles of the plant that releases hazardous material should be demonstrated. That is,

probability of core damage given railroad accident	*	probability of railroad accident within 5 miles of Waterford site that releases hazardous material	<=	probability of core damage due to railroad accident
10^{-3}	*	10^{-2}	<=	10^{-5}

<= defined as less than or equal to.

A suggested way that a probability of core damage given a railroad accident of less than 10^{-3} could be demonstrated is by determining the probability of diffusion of hazardous materials from the accident site towards the plant site. Another possibility is to determine the probability of the plant to successfully isolate the control room from the hazardous material released if the hazardous material should reach the plant site. Another calculation would be to estimate the probability of damage to the plant from possible explosions of the hazardous material. Appendix 6.A.1 gives additional information on the factors that could be determined to demonstrate the probability of core damage given a railroad accident is less than a certain value.

6.7.4 Railroad Accident References

[6.7.1] Title 49 Code of Federal Regulations, Transportation (49 CFR) Parts 100 to 177, Revised as of October 1, 1986, Office of the Federal Register, National Archives and Records Administration, Washington, D.C.

[6.7.2] OTA-SET-304 Transportation of Hazardous Materials, U.S. Congress, Office of Technology Assessment, U.S. Goverment Printing Office, Washington, D.C. July 1986.

[6.7.3] NUREG-0787 Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. July 1981.

6.8 TRUCK ACCIDENTS

Truck accidents pose a hazard to a nuclear power plant due to the possible release of hazardous material towards the plant and/or the possibility of explosion and fire with resulting physical damage to the plant due to blast, debris and fire. Physical damage to a plant due to actual collision with plant structures is considered minimal due to the distance between main highways and plant structures. The movement of trucks with hazardous material within the plant exclusion area is infrequent and controlled except for those plant sites that have major highway going through them. Except for plant sites with major highways going through the exclusion area, only a limited amount of hazardous material are carried in each shipment. Sabotage and deliberate crashes are not considered in this subsection.

According to Ref. 6.8.1, of the four modes of bulk transport of hazardous materials, the highway mode is the most versatile and widely used. It also has the highest probability of an accident. The reasons the highway mode has the highest frequency of accident are: 1) it has the the most miles of network, 2) it has the largest number of individual shipments, 3) it has the largest number of operators, 4) it has the greatest traffic density in an unrestricted right-of-way, and 5) it has the highest average traffic speed. Fortunately, while highway transport has the highest frequency of accidents, the consequences of a release are less because less hazardous material is shipped per shipment than for the other modes of bulk transport.

6.8.1 Truck Safety Regulations

According to Ref. 6.8.1, only interstate commerce is regulated by the U.S. Federal Government under the Hazardous Materials Transportation Act. This is an important distinction because the truck accident data from the Bureau of Motor Carrier Safety (BMCS) only records truck accidents involving trucks in interstate commerce. It is estimated that a large percentage of hazardous materials truck transport is intrastate commerce, with gasoline, fuel oil, and propane deliveries comprising the bulk of it.

Carriers operating solely intrastate do not need to meet Federal standards except for those transporting hazardous wastes, hazardous substances, and flammable cryogenics whose transport is regulated by the Federal Government regardless of whether the commerce is intrastate or interstate [Ref. 6.8.1]. Certain states have regulations similar to the Federal regulations. The other states do not. In some of these states which lack the restrictions of the Federal regulations, the intrastate carriers have become the market for used equipment that no longer meets Federal standards [Ref. 6.8.1].

6.8.2 Hazard to Nuclear Power Plants from Truck Accidents

The current NRC regulations regarding hazards to nuclear power plants from truck accidents as listed in Sections 2.2 and 6.2, gives guidance on

shipment frequency, control room design assumptions, explosion overpressure, etc., but does not relate these factors to a probabilistic determination of core damage. Because core damage probability from truck accidents is not related to the licensing guidance for nuclear power plant siting, the risk analysis method given in Appendix 6.A.1 is used to determine how a currently licensed operating plant would meet the first figure-of-merit of 10^{-5}/yr. for core damage frequency.

Only nuclear power plant sites with major highways that go through or are near the plant exclusion area (see Section 6.3.2 for further discussion of exclusion area and off-site transportation accidents) and also carry large amounts of traffic are exposed to this hazard. This is a conditional "and" statement, that is, a plant may have a highway in or near the plant exclusion area but if it carries little traffic, it is of little concern. Conversely, a highway that carries large volume of traffic, much of which may consist of hazardous material shipments, but is far enough away, may not be of concern. It is those unique situations of a high carrying large volume of traffic with hazardous material shipments in or near a power plant exclusion area which are of concern.

The San Onofre nuclear power plant site in San Diego County, California near the town of San Clemente, will be used as an example in an initial screening analysis to demonstrate a method by which plants could determine if further analysis will be necessary in order to meet the first figure-of-merit for core damage frequency.

According to the San Onofre 2 and 3 Safety Evaluation Report (SER) [Ref. 6.8.2], an Atchinson, Topeka and Santa Fe railroad line, the San Diego Freeway (Interstate 5) and U.S. Highway 101 pass through the San Onofre site exclusion area, approximately 600 to 700 feet east of the Units 1, 2 and 3 reactor containment buildings. This analysis will examine only the hazard to the San Onofre site from the truck traffic on the San Diego Freeway but to determine the total risk to the San Onofre site from transportation accidents, the risk calculation should be summed over all transportation modes over all transportation routes. Figure 6.4 shows the San Onofre site and the distance from the reactor containments to the major transportation routes stated above [Ref. 6.8.3].

According to the San Onofre 2 and 3 Final Safety Analysis Report (FSAR) [Ref. 6.8.3], the truck accident data for the San Diego Freeway (Interstate 5) for a 10 mile segment (milepost R61.38 to R71.38) extending 5 miles in each direction from the San Onofre plant site was found from the California Department of Transportation for 1974 to 1977. The truck traffic rates were based on weighted sample counting, extrapolated to annual counts, and combined northbound and southbound data. Traffic accidents were reported to the state if property damage was $200 or greater or there had been personal injury or death. The truck accident rate was given as 0.566×10^{-6} accidents per truck mile based on 48 accidents and 84.74 million truck miles on the San Diego Freeway from 1974 to 1978. The highest observed truck accident rate was 0.687×10^{-6} in 1976 and the lowest observed truck accident rate was 0.453×10^{-6} in 1975.

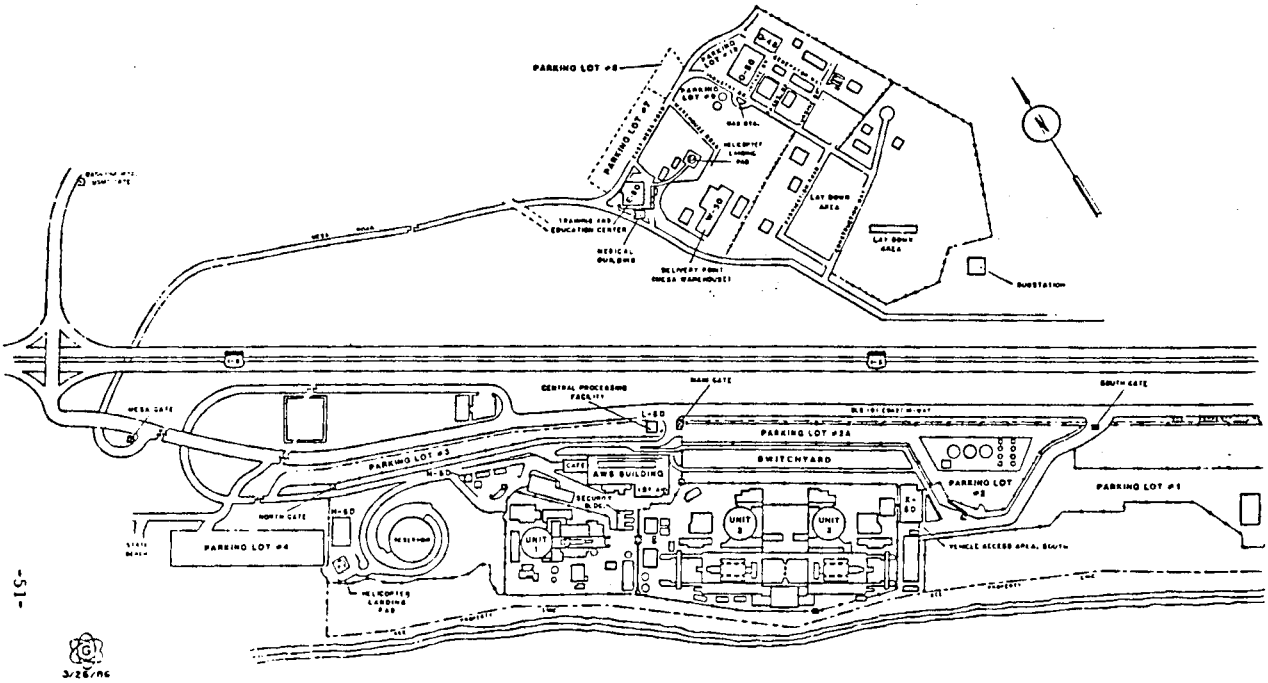


Figure 6.4

San Onofre Nuclear Generation Site (SONGS)
[Ref. 6.8.3]

From Table 6.A.6.1 of Appendix 6.A.6, the average accident rate for U.S. Intercity For-Hire Motor Carriers was found from the Bureau of Motor Carrier Safety (BMCS) to be 2.48×10^{-6} accidents per truck mile for the period 1960 to 1972. The truck accident property damage threshold was $250 for this period. After 1973, changes in reporting rules, redefinition of the damage threshold limit from $250 to $2000, and the lack of truck miles tabulation has made it so that truck accident data from the BMCS after 1973 is not comparable to data before 1973. Relying strictly on BMCS truck accident data prior to 1973 is probably overconservative because of the institution of the 55 mile per hour national speed limit, the effects of inflation, new types of equipment, etc. On the other hand, increased traffic density, decreased safety inspections of truck and decreased enforcement of the truck regulations, decreased railroad traffic resulting in increased truck traffic, has probably served to decrease the amount of conservatism inherent in the pre-1973 BMCS accident data. In other words, a truly representative estimate of the truck accident rate for all truck traffic consisting of large tractor/semitrailer truck combinations (the traffic of interest) could not be found from U.S. Government reference sources. Therefore, one must turn to industry sources in order to find more recent truck accident data.

Table 6.A.6.3 of Appendix 6.A.6 gives the truck accident rate for trucks in the petroleum industry from 1968 to 1981. The average accident rate for this period was found to be 6.45×10^{-6} accidents/truck mile. This data was taken from American Petroleum Institute (API) publications and covers the period when the national gasoline shortage crisis occurred and the 55 mile per hour national speed limit was imposed.

As the previous paragraphs have indicated, truck accident rates may vary widely depending on the sources they are taken from. The truck accident rate found from the California Department of Transportation for the San Diego Freeway from 1974 to 1977 based on actual traffic and accident counts, is probably the most realistic, although now its data may be slightly dated. The truck accident rate for the San Diego Freeway of 0.566×10^{-6} accidents per truck mile compares reasonably well with the truck accident rate from the API of 6.45×10^{-6} accidents/truck mile if one considers that the truck accident rate from API is a national average and includes all oil industry truck traffic over all roads. The San Diego Freeway is a large 10-lane (at least) major freeway capable of carrying a large volume of traffic. It seems reasonable that the truck accident rate on the San Diego Freeway within five miles of the San Onofre site is a factor of about 11 less than the oil industry national truck accident rate due to the characteristics of the San Diego Freeway. Therefore, for this initial screening analysis, the truck accident rate on the San Diego Freeway within five miles of the San Onofre site of 0.566×10^{-6} accidents per truck mile will be used.

The methodology presented in Appendix 6.A.1 is used to perform the initial screening analysis. The San Diego Freeway is located between 600 to 700 feet from San Onofre Units 1, 2 and 3 containment buildings, so $D \doteq 0.11$ to 0.13 miles. Since this is simply an initial screening analysis, D will be set to be zero so the vehicle hazard distance, L, is 10 miles.

The number of hazardous material shipments that traveled by truck (assuming one shipment per truck) on the San Diego Freeway near the San Onofre plant site was found from Ref. 6.8.3 to be 29,912 in 1978 and 24,986 in 1984. Year-by-year information was not given so no trends could be found, therefore, the highest number of yearly hazardous material shipments, 29,912 will be used.

The total number of vehicle (truck) miles of vehicles carrying hazardous material within five miles of the San Onofre plant site is then:

$$(29{,}912 \text{ trucks/yr.}) \times (10 \text{ miles}) = 3.0 \times 10^5 \text{ truckmiles/yr.}$$

Multiplying the accident rate of trucks on the San Diego Freeway with the number of truckmiles of trucks carrying hazardous material within five miles of the San Onofre plant site gives the truck accident frequency rate of trucks carrying hazardous material within five miles of the San Onofre plant site on the San Diego Freeway:

$$\left(3.0 \times 10^5 \frac{\text{trkmiles}}{\text{year}}\right) \times \left(0.566 \times 10^{-6} \frac{\text{trk acc.}}{\text{trkmile}}\right) = 1.7 \times 10^{-2}$$

If the proportion of trucks involved in accidents carrying and releasing hazardous material to the total number of trucks carrying hazardous materials involved in accidents could be found, then the frequency rate above could be reduced by this proportion. Unfortuately, this information is unavailable presently.

Similar calculations should be done for the other highways with fives miles of the San Onofre plant site. The total frequency for a truck accident involving a truck carrying hazardous material (but not necessarily involving a release) is then the summation over all the highways of the frequency for each highway that carries hazardous material truck traffic.

6.8.3 Power Plant Response to Truck Accidents

In the case of San Onofre, assuming the San Diego Freeway as the predominant risk contributor for truck accidents, then in order to meet the first figure-of-merit of 1×10^{-5}/yr. for core damage, the probability of core damage of less than 10^{-3} given a truck accident involving a truck carrying hazardous material within 5 miles of the plant should be demonstrated. That is,

probability of core damage given truck accident	*	probabilty of truck accident within 5 miles of San Onofre site involving truck carrying hazardous material	<=	probability of core damage due to truck accident
10^{-3}	*	10^{-2}	<=	10^{-5}

<= defined as less than or equal to.

A suggested way that a probability of core damage given a truck accident of less than 10^{-3} could be demonstrated is by determining the probability of diffusion of hazardous materials from the accident site towards the plant site. Another possibility is to determine the probability of the plant to successfully isolate the control room from the hazardous material released if the hazardous material should reach the plant site. Another calculation would be the probability of damage to the plant from possible explosions of the hazardous material. Appendix 6.A.1 gives additional information on the factors that could be determined to demonstrate the probability of core damage given a truck accident is less than a certain value.

6.8.4 Truck Accident References

[6.8.1] OTA-SET-304 Transportation of Hazardous Materials, U.S. Congress, Office of Technology Assessment, U.S. Government Printing Office, Washington, D.C. July 1986.

[6.8.2] NUREG-0712 Safety Evaluation Report Related to the Operation of San Onofre Nuclear Generating Station, Units 2 and 3, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. February 1981.

[6.8.3] San Onofre 2 and 3 Updated Final Safety Analysis Report, Southern California Edison, Revision 3. February 1987.

6.9 GENERAL OBSERVATIONS OF TRANSPORTATION ACCIDENT RISK ANALYSIS

The accident frequency data for commercial aviation, general aviation and railroads was found to be good. Federal agencies tabulate accident data and operating data on a relatively consistent basis and it is possible to make comparisons of accident and operating data year-by-year.

For military aviation, ship/barge traffic and pipeline operation, current reliable accident frequency data has yet to be found. Research continues in these areas.

Up-to-date accident frequency data could not be found for trucks. This information does not appear to be available at the present time. Some of the reasons why truck accident data is less than satisfactory are that Federal agencies no longer tabulate the desired data; State agencies may tabulate the desired data but they can only cover a limited portion of the total fleet population; and industry sources are usually done for specific purposes and are generally not done year-by-year on a consistent basis.

6.9.1 Recommendations for Plant-Specific Transportation Accident Risk Analysis

For most reactor sites, the risk from transportation accidents is small and can be easily shown to be small. However, there are a few sites where more extensive analyses may be required in order to screen out this hazard. It is important to emphasize that the scrrening must be probabilistic in character, in order to show that the probability (or frequency) of poentential accidents is low enough to satisfy the first figure-of-merit of core damage.

A recommended step-by-step approach is as follows:

1. The first obvious step would be to identify the transportation modes that are present near (within about five miles) the plant site. Those modes that are not present can be screened out easily.

2. The next step is to identify the transportation routes and determine the traffic traveling each route. Those transportation routes with little traffic may be screened out at this point. Those routes with sufficient traffic to be of concern should proceed to the next step.

3. A survey of the hazardous material traffic carried over each transportation route should be done at this point. If this cannot be done, then an estimate based on industry surveys may be appropriate if sufficient justification and conservatism is provided.

4. An estimate of the risk from tranpsortation accidents can be made by multiplying the hazardous material traffic carried over each transportation route by the appropriate mode accident rate. Caution should be exercised that the proper transportation accident rate for each mode be used because, as this section has shown, accident rates may vary according to their source. A site-

specific accident rate based on actual accident experience would be the most desirable situation but usually there is not enough data or the operational experience is inappropriate for the current situation. If that is the case, then national, regional or industry accident rates may be usable, again if sufficient justificationa and conservatism is provided.

5. If the transportation accidents for each mode cannot be dismissed on the basis of the initiating event frequency, then analysis of the plant response to the transportation accident should be done at this point. This analysis may take the form of determining the probability of the hazardous material detonating, catching fire, diffusing to the plant site, forcing the plant operators to evacuate or isolate the control room, damage to plant equipment resulting in a transient event, etc. Such analysis may be quite elaborate, but need not be if a simplified analysis can accomplish the same objective. The most important aspect for this step is that the analysis be done probabilistically.

6.9.2 Future Areas of Research Needed

A nuclear power plant whose site was found to be acceptable when it was originally licensed may find its site open to question particularly in regards to meeting the first figure-of-merit for core damge. The reasons for this are two fold.

The first is that the original licensing decision was not based on a strictly probabilistic basis, if at all. Most of the NRC licensing guidance was deterministic and the probabilistic portion gave guidance only on the acceptable limits (i.e., number of hazardous material shipments, distance to an airport, etc.) and did not relate to the core damage frequency.

The second reason is that the transportation industry changes with time. For example, minor transportation routes may develop into major transportation routes with increased traffic and greater traffic density. Vehicle speeds may increse due to regulatory changes and the introduction of new technology. Vehicle weights may increase due to the introduction of new vehicle models. New industries near a plant site may increase hazardous material shipments. Cargo types may change due to changes in the regional or national economy.

For the above reasons, transportation accident analysis that was considered very conservative at the time a plant was licensed may not be as conservative presently. Demonstrating that a plant meets the first figure-of-merit for core damge would necessitate a reanalysis of the plant-specific transportation hazards to the power plant site if the transportation hazards cannot be easily dismissed because of distance between the routes and the site and/or the lack of traffic on these routes.

If plant-specific reanalysis of transportation hazards to the site is necessary, then current estimates of the transportation accident rate for each applicable mode are needed. Complete up-to-date accident data and rates could be found for commercial aviation, general aviation and railroads. Research continues for military aviation, pipelines and marine accident data. Current, reliable truck accident data can be considered less than desired. The reasons for this is that the federal agencies charged with compiling truck accident

data no longer gather the needed data. Some state agencies may compile some of the needed data but their database comprise only a portion of the total truck population. Industry sources are not done on a consistent year to year basis and thus historical trends are difficult to draw. Since many plant sites dismiss the hazard from transportation accidents on the basis of the initiating event frequency, complete, reliable, and current accident data is required. More research should be done on accident data and rates for those modes for which it is less than desirable, i.e., military aviation, pipeline, marine and trucks.

For those plant sites where probabilistic analysis of the plant response to the transportation accident is necessary, much research remains as to how this can be accomplished. For example, detonation, flammability and toxicity limits for hazardous materials vary widely according to the type of material, concentration, etc. Each of these limits may have to be expressed as probability distributions. The diffusion of volatile hazardous materials is another area where much research remains to be done. Some studies are available but these are deterministic in nature and lack guidance on how these studies may be converted into probabilities. Research may need to be done to probabilistically determine the plant's ability to isolate the control room from the outside environment should the volatile hazardous material reach the plant. Another field of research is the plant's ability to run without operation intervention due to the operators being forced to evacuate the control room or the operators being overcome by toxic fumes. The plant's response to a nearby explosion or fire which damages plant structures or equipment is another area where probabilistic analysis remains. For many of these areas, PRA methodology might be modified or even used directly to provide some of the answers. Other areas may require other forms of probabilistic risk analysis.

6.10 SUMMARY AND CONCLUSIONS

Transportation accidents may need to be considered among the external hazards to nuclear power plants. This is primarily due to the lack of probabilistic analysis of the plant's response to a nearby transportation accident. If it can be shown that the probability of a transportation occuring nearby is sufficiently small, then it may be possible to end the analysis at that point. Otherwise, additional work would need to be done in the area of probabilistic analysis of a plant's response to a nearby transportation accident. This additional work is probably needed for only a few select sites.

It is possible to separate the probabilistic analysis of the initiating event (the transportation accident) from the probabilistic analysis of the plant response to the initiating event. That is, the probability of a transportation accident occuring near a plant site does not depend on the plant design.

6.A APPENDICES

6.A.1 TRANSPORTATION ACCIDENT EVENT MODEL

Transportation accidents are among the factors that are considered in choosing a nuclear power plant site. The movement of hazardous (toxic/flammable/explosive) materials in or near the power plant site whose release, detonation or burning due to an accident involving the vehicle or pipeline, could affect the plant to the degree that damage to the reactor or release of radioactive material from the reactor could result must be considered. Another possible scenario involves the collision of the vehicle itself, with or without hazardous materials, into the power plant, causing sufficent damage to plant to cause damage to the reactor core or release of radioactive material from the reactor core.

For the purposes of this report, transportation accidents are defined to be accidents involving aviation traffic of all types, ship and barge traffic, gas/oil/chemical pipelines, railroad traffic and truck traffic. The definition of an accident is defined for each mode of transportation in its relevant section.

The evaluation of the risk of core damage at a nuclear power plant due to transportation accidents has been accomplished using the event sequence trees presented in Figures 6.A.1 to 6.A.10. These event trees have been developed so that they may be applied to all modes of transportation if new information makes it desirable to do so in the future.

It should be noted that the event sequence trees presented in Figures 6.A.1 to 6.A.10 are not intended to be a detailed risk assessment procedure for evaluating transportation accident hazards, but rather a guide that shows how the various factors fit within the overall risk analysis method.

The first term needed to evaluate transportation accident risks is the frequency of transportation accidents near the plant site. Generally, accident rates are tabulated in terms of vehicle-miles. So, in order to determine the frequency of transportation accidents near the plant site, the number of vehicle miles per year near the plant site must be determined. The first question is how near is near? SRP Sections 2.2.1 - 2.2.2 and 2.2.3 specifies an area of radius 5 miles or 8.05 kilometers must be used in considering a location for a nuclear power plant site. For the purposes of this study, a transportation accident within a 5 mile radius to a nuclear power plant is considered a potential hazard. Next, since few major transportation routes actually go through a nuclear power site, the question of how many miles of the transportation route is within a 5 mile radius of the site if the route's nearest approach to the site is a certain offset distance, D, must be determined. This is shown in Figure 6.A.11. The vehicle hazard distance, L, is defined as the distance that is traveled by vehicles which are a potential hazard to the plant and are within 5 miles of the plant site. If the offset distance, D, of a transportation route is zero, then the vehicle hazard distance is 10 miles because the transportation route goes through the plant site and all vehicles that travel on that route within 5 miles of the

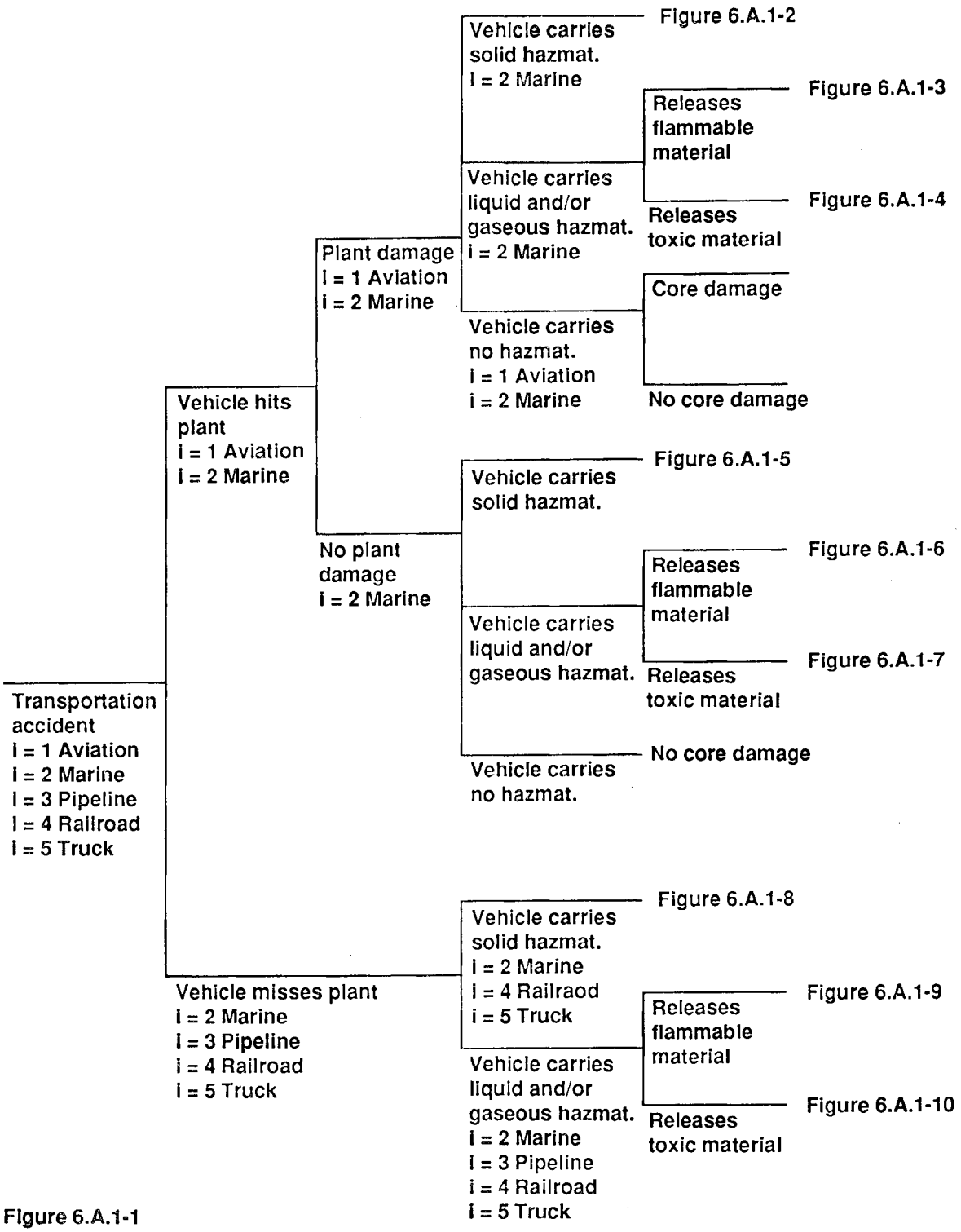

Transportation accident
i = 1 Aviation
i = 2 Marine
i = 3 Pipeline
i = 4 Railroad
i = 5 Truck
Vehicle hits plant
i = 1 Aviation
i = 2 Marine
Plant damage
i = 1 Aviation
i = 2 Marine
Vehicle carries solid hazmat.
i = 2 Marine
Figure 6.A.1-2
Vehicle carries liquid and/or gaseous hazmat.
i = 2 Marine
Releases flammable material
Figure 6.A.1-3
Releases toxic material
Figure 6.A.1-4
Vehicle carries no hazmat.
i = 1 Aviation
i = 2 Marine
Core damage
No core damage
No plant damage
i = 2 Marine
Vehicle carries solid hazmat.
Figure 6.A.1-5
Vehicle carries liquid and/or gaseous hazmat.
Releases flammable material
Figure 6.A.1-6
Releases toxic material
Figure 6.A.1-7
Vehicle carries no hazmat.
No core damage
Vehicle misses plant
i = 2 Marine
i = 3 Pipeline
i = 4 Railroad
i = 5 Truck
Vehicle carries solid hazmat.
i = 2 Marine
i = 4 Railraod
i = 5 Truck
Figure 6.A.1-8
Vehicle carries liquid and/or gaseous hazmat.
i = 2 Marine
i = 3 Pipeline
i = 4 Railroad
i = 5 Truck
Releases flammable material
Figure 6.A.1-9
Releases toxic material
Figure 6.A.1-10

Figure 6.A.1-1

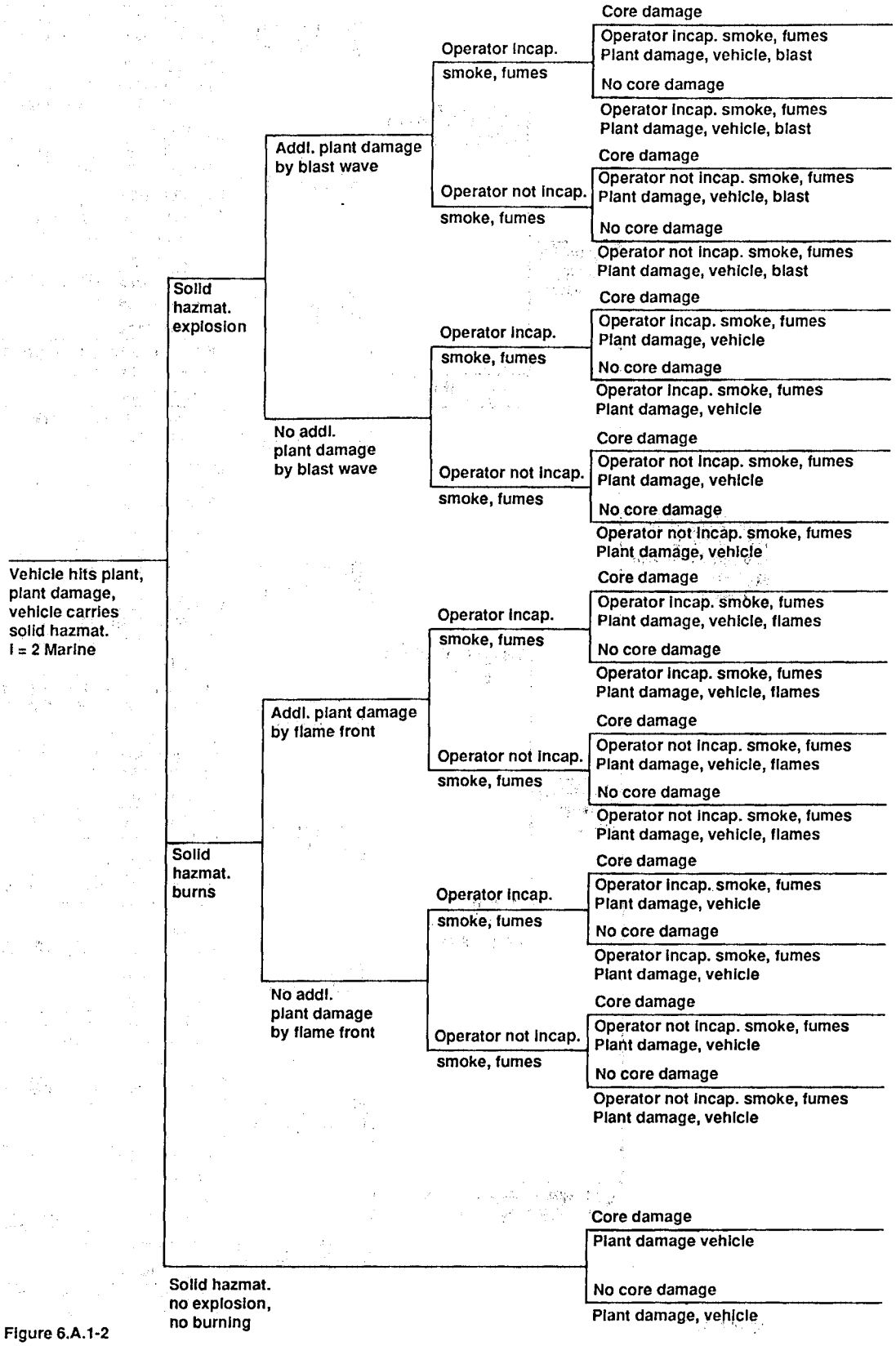

Vehicle hits plant, plant damage, vehicle carries solid hazmat. I = 2 Marine
Solid hazmat. explosion
Addl. plant damage by blast wave
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes Plant damage, vehicle, blast
No core damage
Operator incap. smoke, fumes Plant damage, vehicle, blast
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes Plant damage, vehicle, blast
No core damage
Operator not incap. smoke, fumes Plant damage, vehicle, blast
No addl. plant damage by blast wave
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes Plant damage, vehicle
No core damage
Operator incap. smoke, fumes Plant damage, vehicle
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes Plant damage, vehicle
No core damage
Operator not incap. smoke, fumes Plant damage, vehicle
Solid hazmat. burns
Addl. plant damage by flame front
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes Plant damage, vehicle, flames
No core damage
Operator incap. smoke, fumes Plant damage, vehicle, flames
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes Plant damage, vehicle, flames
No core damage
Operator not incap. smoke, fumes Plant damage, vehicle, flames
No addl. plant damage by flame front
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes Plant damage, vehicle
No core damage
Operator incap. smoke, fumes Plant damage, vehicle
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes Plant damage, vehicle
No core damage
Operator not incap. smoke, fumes Plant damage, vehicle
Solid hazmat. no explosion, no burning
Core damage
Plant damage vehicle
No core damage
Plant damage, vehicle

Figure 6.A.1-2

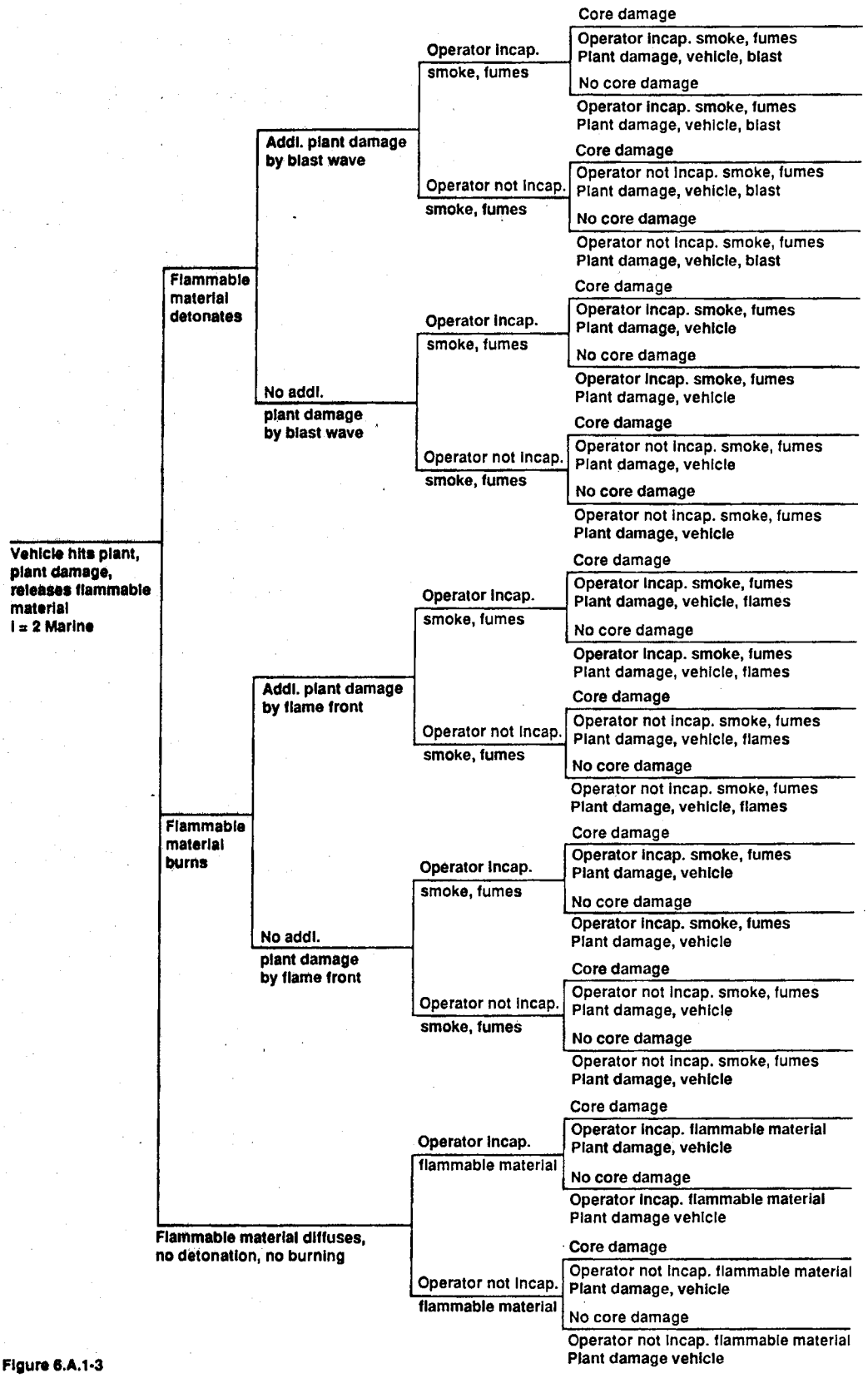

Vehicle hits plant, plant damage, releases flammable material
I = 2 Marine
Flammable material detonates
Addl. plant damage by blast wave
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes
Plant damage, vehicle, blast
No core damage
Operator incap. smoke, fumes
Plant damage, vehicle, blast
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes
Plant damage, vehicle, blast
No core damage
Operator not incap. smoke, fumes
Plant damage, vehicle, blast
No addl. plant damage by blast wave
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes
Plant damage, vehicle
No core damage
Operator incap. smoke, fumes
Plant damage, vehicle
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes
Plant damage, vehicle
No core damage
Operator not incap. smoke, fumes
Plant damage, vehicle
Flammable material burns
Addl. plant damage by flame front
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes
Plant damage, vehicle, flames
No core damage
Operator incap. smoke, fumes
Plant damage, vehicle, flames
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes
Plant damage, vehicle, flames
No core damage
Operator not incap. smoke, fumes
Plant damage, vehicle, flames
No addl. plant damage by flame front
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes
Plant damage, vehicle
No core damage
Operator incap. smoke, fumes
Plant damage, vehicle
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes
Plant damage, vehicle
No core damage
Operator not incap. smoke, fumes
Plant damage, vehicle
Flammable material diffuses, no detonation, no burning
Operator incap. flammable material
Core damage
Operator incap. flammable material
Plant damage, vehicle
No core damage
Operator incap. flammable material
Plant damage vehicle
Operator not incap. flammable material
Core damage
Operator not incap. flammable material
Plant damage, vehicle
No core damage
Operator not incap. flammable material
Plant damage vehicle

Figure 6.A.1-3

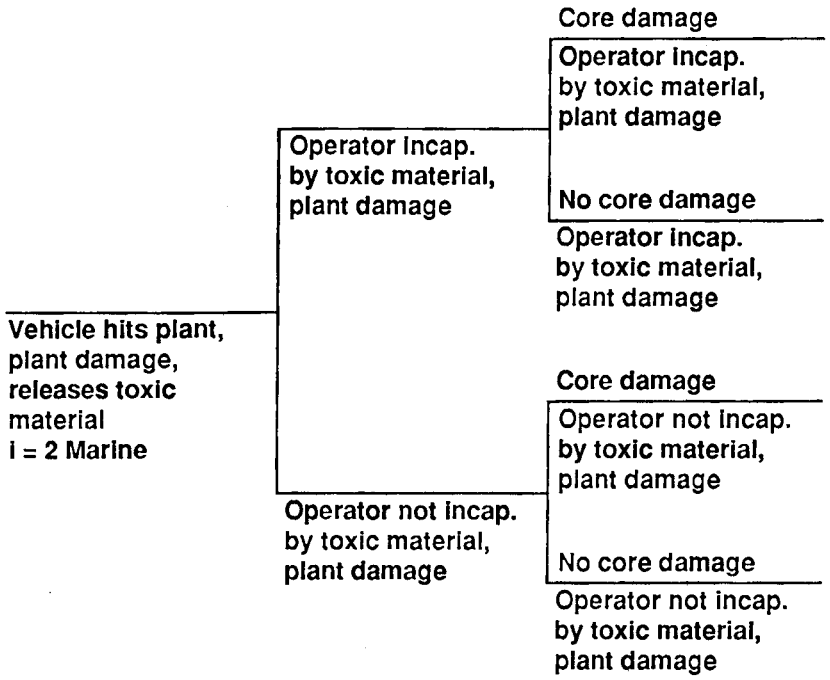

Core damage
Operator incap.
by toxic material,
plant damage
Operator incap.
by toxic material,
plant damage
No core damage
Operator incap.
by toxic material,
plant damage
Vehicle hits plant,
plant damage,
releases toxic
material
i = 2 Marine
Core damage
Operator not incap.
by toxic material,
plant damage
Operator not incap.
by toxic material,
plant damage
No core damage
Operator not incap.
by toxic material,
plant damage

Figure 6.A.1-4

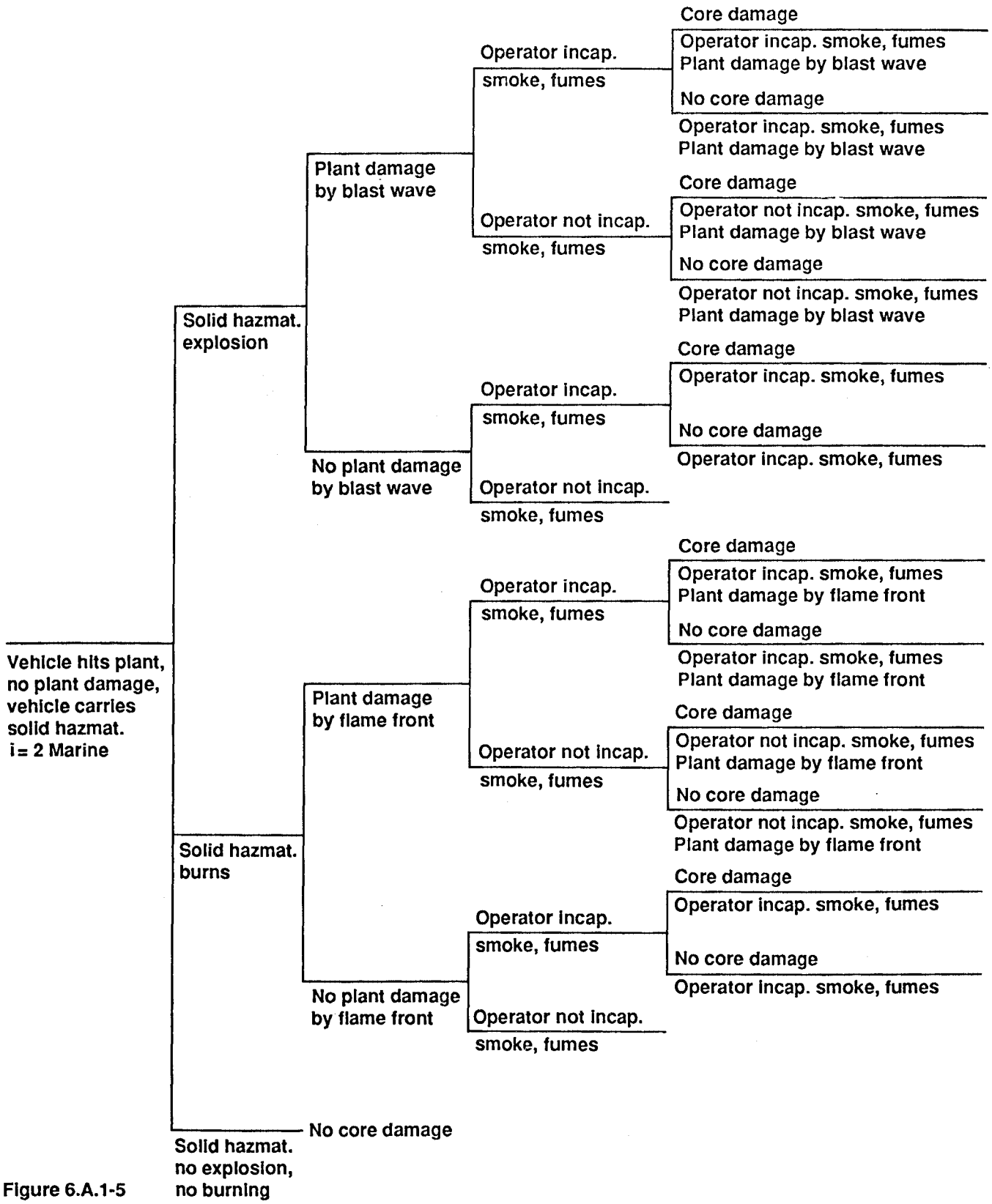

Vehicle hits plant, no plant damage, vehicle carries solid hazmat. i= 2 Marine
Solid hazmat. explosion
Plant damage by blast wave
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes Plant damage by blast wave
No core damage
Operator incap. smoke, fumes Plant damage by blast wave
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes Plant damage by blast wave
No core damage
Operator not incap. smoke, fumes Plant damage by blast wave
No plant damage by blast wave
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes
No core damage
Operator incap. smoke, fumes
Operator not incap. smoke, fumes
Solid hazmat. burns
Plant damage by flame front
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes Plant damage by flame front
No core damage
Operator incap. smoke, fumes Plant damage by flame front
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes Plant damage by flame front
No core damage
Operator not incap. smoke, fumes Plant damage by flame front
No plant damage by flame front
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes
No core damage
Operator incap. smoke, fumes
Operator not incap. smoke, fumes
Solid hazmat. no explosion, no burning
No core damage

Figure 6.A.1-5

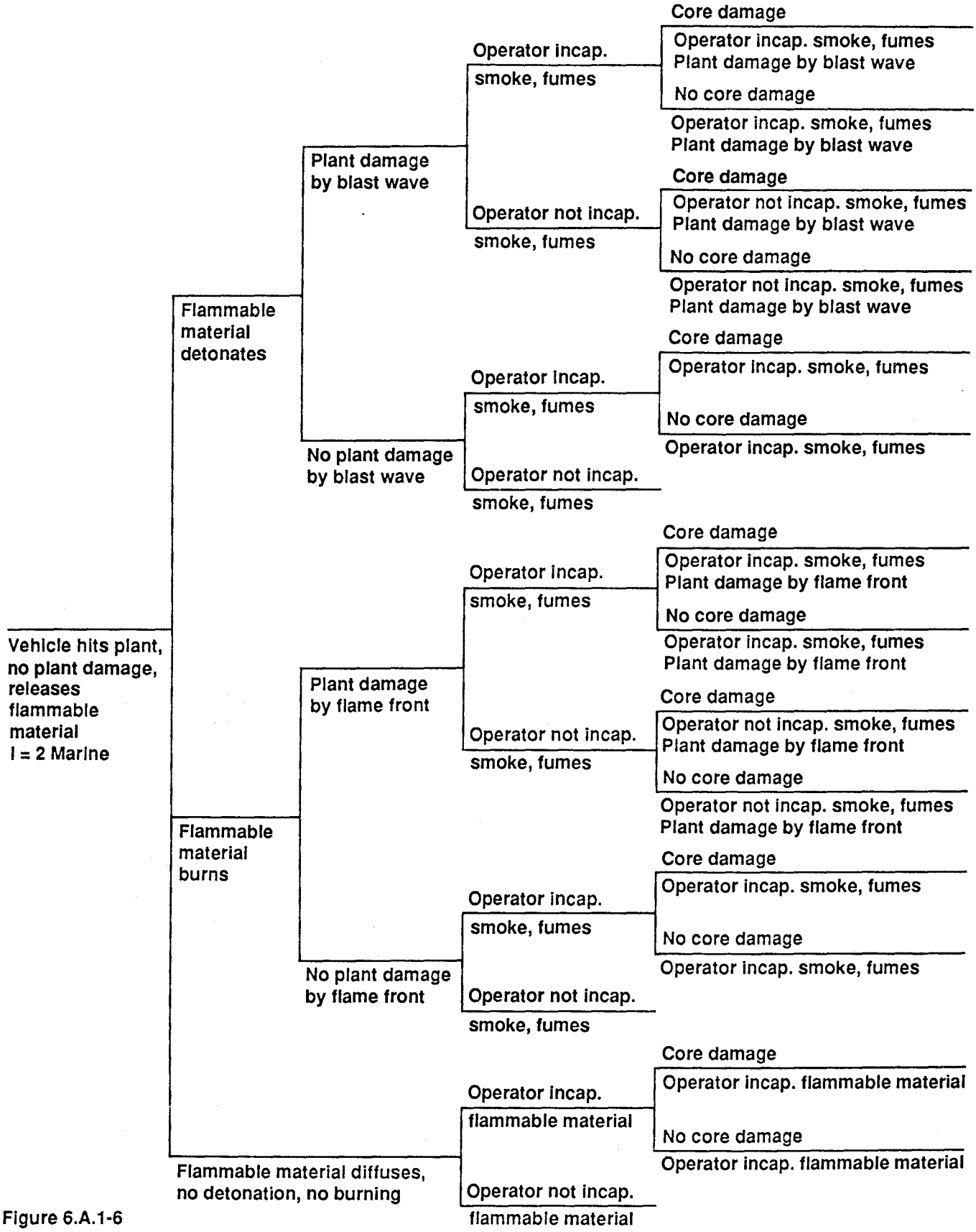

Vehicle hits plant, no plant damage, releases flammable material I = 2 Marine
Flammable material detonates
Plant damage by blast wave
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes Plant damage by blast wave
No core damage
Operator incap. smoke, fumes Plant damage by blast wave
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes Plant damage by blast wave
No core damage
Operator not incap. smoke, fumes Plant damage by blast wave
No plant damage by blast wave
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes
No core damage
Operator incap. smoke, fumes
Operator not incap. smoke, fumes
Flammable material burns
Plant damage by flame front
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes Plant damage by flame front
No core damage
Operator incap. smoke, fumes Plant damage by flame front
Operator not incap. smoke, fumes
Core damage
Operator not incap. smoke, fumes Plant damage by flame front
No core damage
Operator not incap. smoke, fumes Plant damage by flame front
No plant damage by flame front
Operator incap. smoke, fumes
Core damage
Operator incap. smoke, fumes
No core damage
Operator incap. smoke, fumes
Operator not incap. smoke, fumes
Flammable material diffuses, no detonation, no burning
Operator incap. flammable material
Core damage
Operator incap. flammable material
No core damage
Operator incap. flammable material
Operator not incap. flammable material

Figure 6.A.1-6

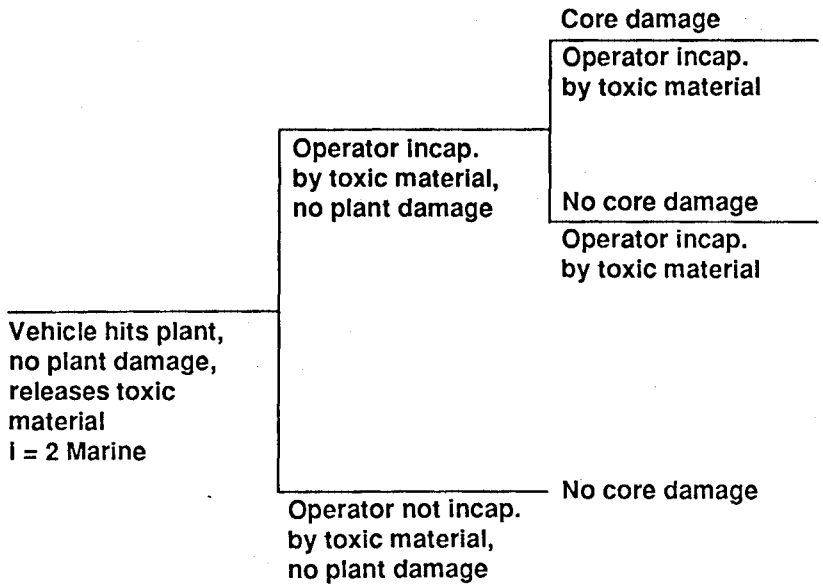

Core damage
Operator incap.
by toxic material
Operator incap.
by toxic material,
no plant damage
No core damage
Operator incap.
by toxic material
Vehicle hits plant,
no plant damage,
releases toxic
material
i = 2 Marine
No core damage
Operator not incap.
by toxic material,
no plant damage

Figure 6.A.1-7

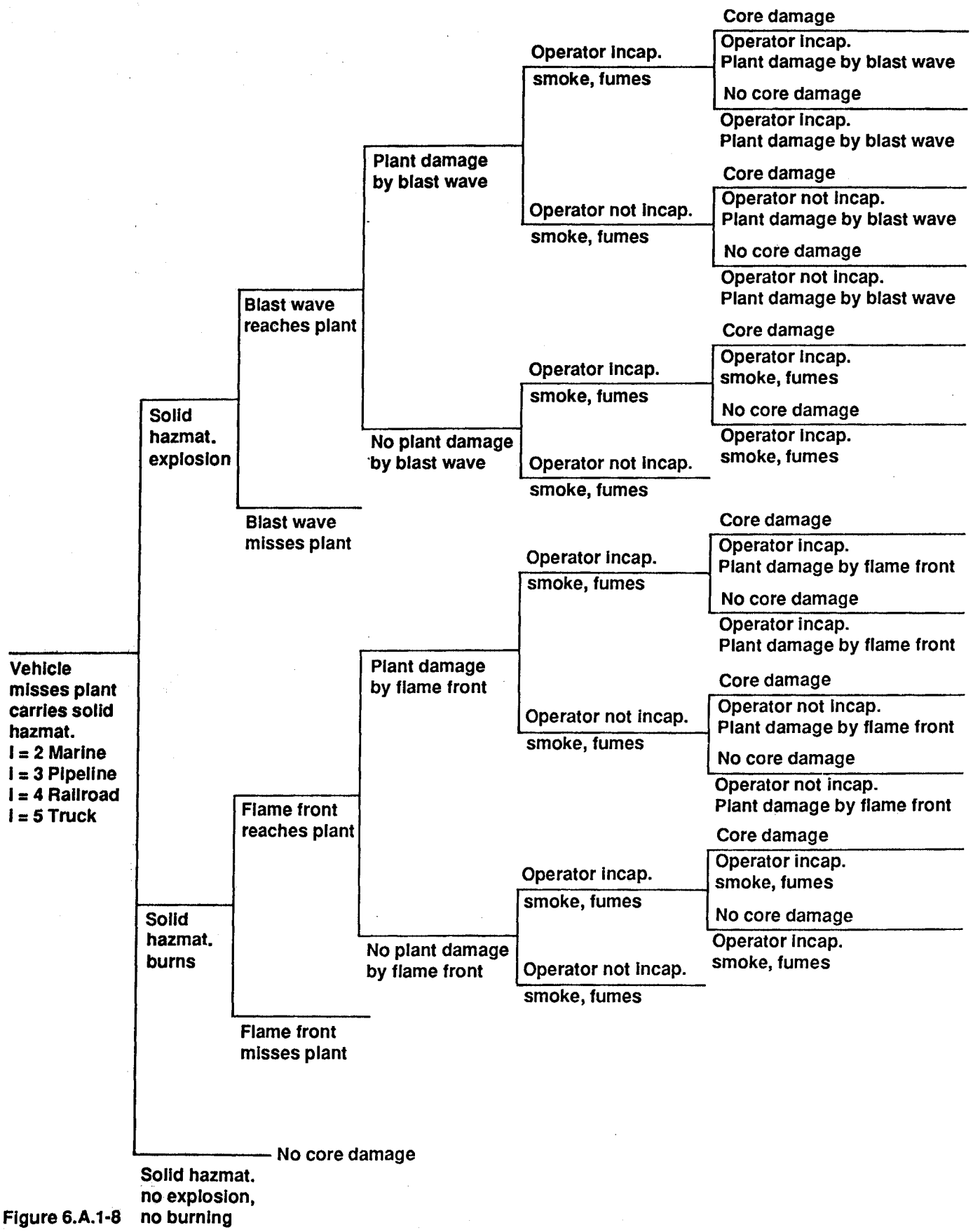

Vehicle misses plant carries solid hazmat.
I = 2 Marine
I = 3 Pipeline
I = 4 Railroad
I = 5 Truck
Solid hazmat. explosion
Blast wave reaches plant
Plant damage by blast wave
Operator incap.
smoke, fumes
Core damage
Operator incap.
Plant damage by blast wave
No core damage
Operator incap.
Plant damage by blast wave
Operator not incap.
smoke, fumes
Core damage
Operator not incap.
Plant damage by blast wave
No core damage
Operator not incap.
Plant damage by blast wave
No plant damage by blast wave
Operator incap.
smoke, fumes
Core damage
Operator incap.
smoke, fumes
No core damage
Operator incap.
smoke, fumes
Operator not incap.
smoke, fumes
Blast wave misses plant
Solid hazmat. burns
Flame front reaches plant
Plant damage by flame front
Operator incap.
smoke, fumes
Core damage
Operator incap.
Plant damage by flame front
No core damage
Operator incap.
Plant damage by flame front
Operator not incap.
smoke, fumes
Core damage
Operator not incap.
Plant damage by flame front
No core damage
Operator not incap.
Plant damage by flame front
No plant damage by flame front
Operator incap.
smoke, fumes
Core damage
Operator incap.
smoke, fumes
No core damage
Operator incap.
smoke, fumes
Operator not incap.
smoke, fumes
Flame front misses plant
No core damage
Solid hazmat. no explosion, no burning

Figure 6.A.1-8

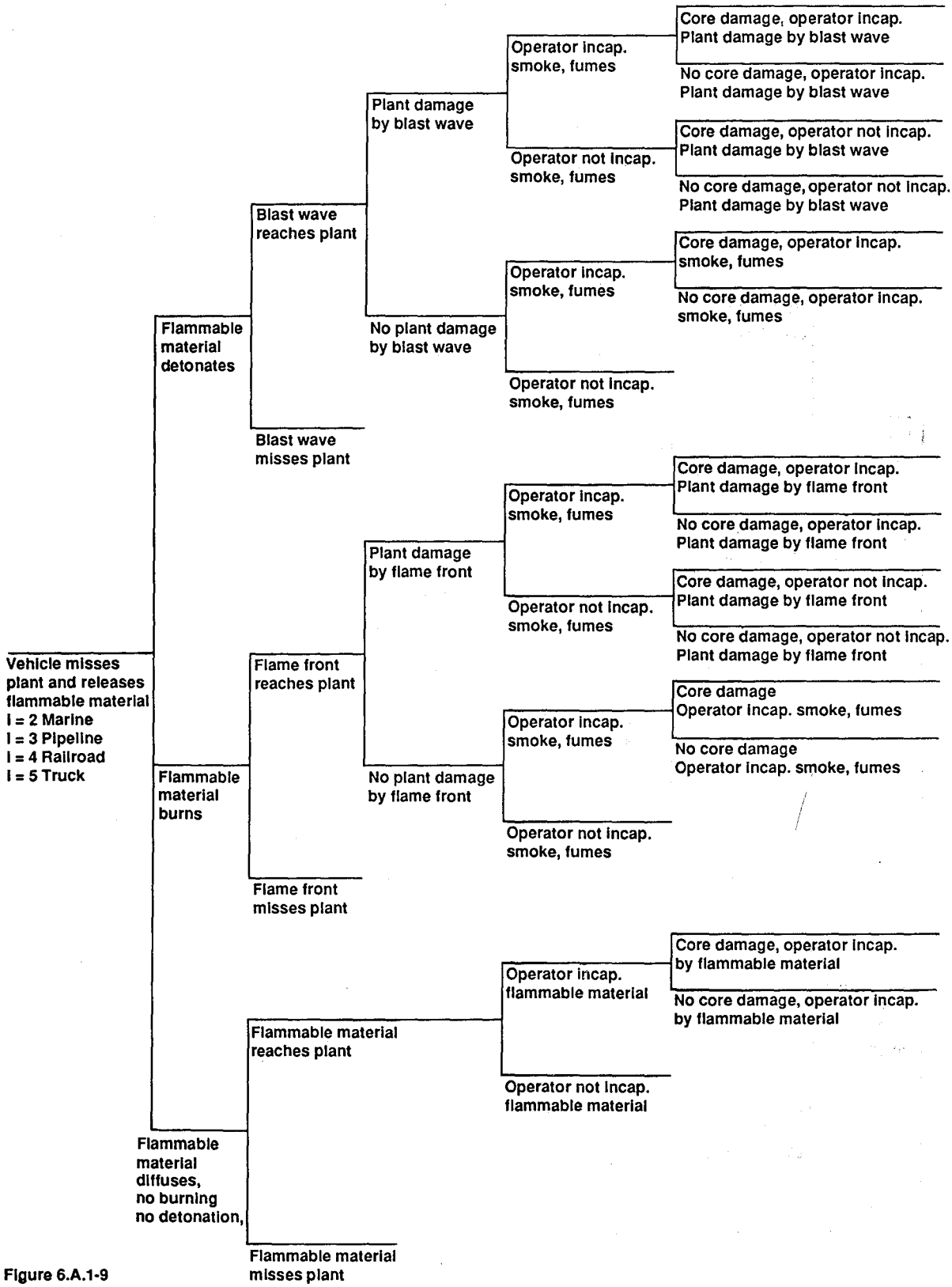

Vehicle misses plant and releases flammable material
I = 2 Marine
I = 3 Pipeline
I = 4 Railroad
I = 5 Truck
Flammable material detonates
Blast wave reaches plant
Plant damage by blast wave
Operator incap. smoke, fumes
Core damage, operator incap. Plant damage by blast wave
No core damage, operator incap. Plant damage by blast wave
Operator not incap. smoke, fumes
Core damage, operator not incap. Plant damage by blast wave
No core damage, operator not incap. Plant damage by blast wave
No plant damage by blast wave
Operator incap. smoke, fumes
Core damage, operator incap. smoke, fumes
No core damage, operator incap. smoke, fumes
Operator not incap. smoke, fumes
Blast wave misses plant
Flammable material burns
Flame front reaches plant
Plant damage by flame front
Operator incap. smoke, fumes
Core damage, operator incap. Plant damage by flame front
No core damage, operator incap. Plant damage by flame front
Operator not incap. smoke, fumes
Core damage, operator not incap. Plant damage by flame front
No core damage, operator not incap. Plant damage by flame front
No plant damage by flame front
Operator incap. smoke, fumes
Core damage Operator incap. smoke, fumes
No core damage Operator incap. smoke, fumes
Operator not incap. smoke, fumes
Flame front misses plant
Flammable material diffuses, no burning no detonation,
Flammable material reaches plant
Operator incap. flammable material
Core damage, operator incap. by flammable material
No core damage, operator incap. by flammable material
Operator not incap. flammable material
Flammable material misses plant

Figure 6.A.1-9

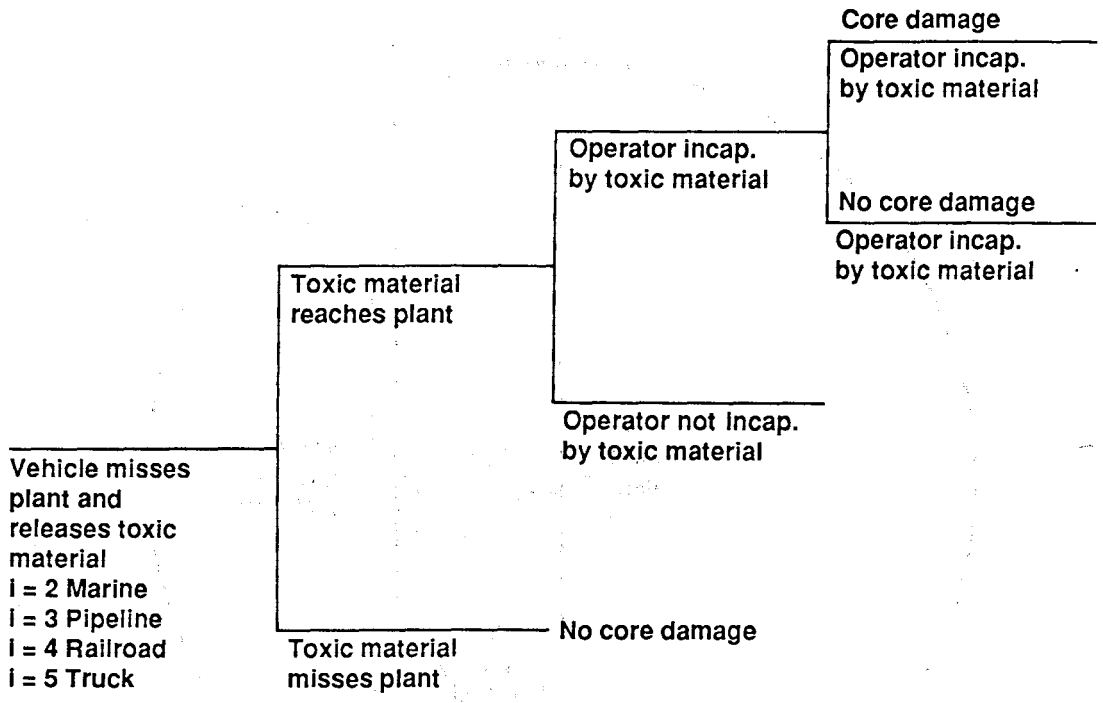
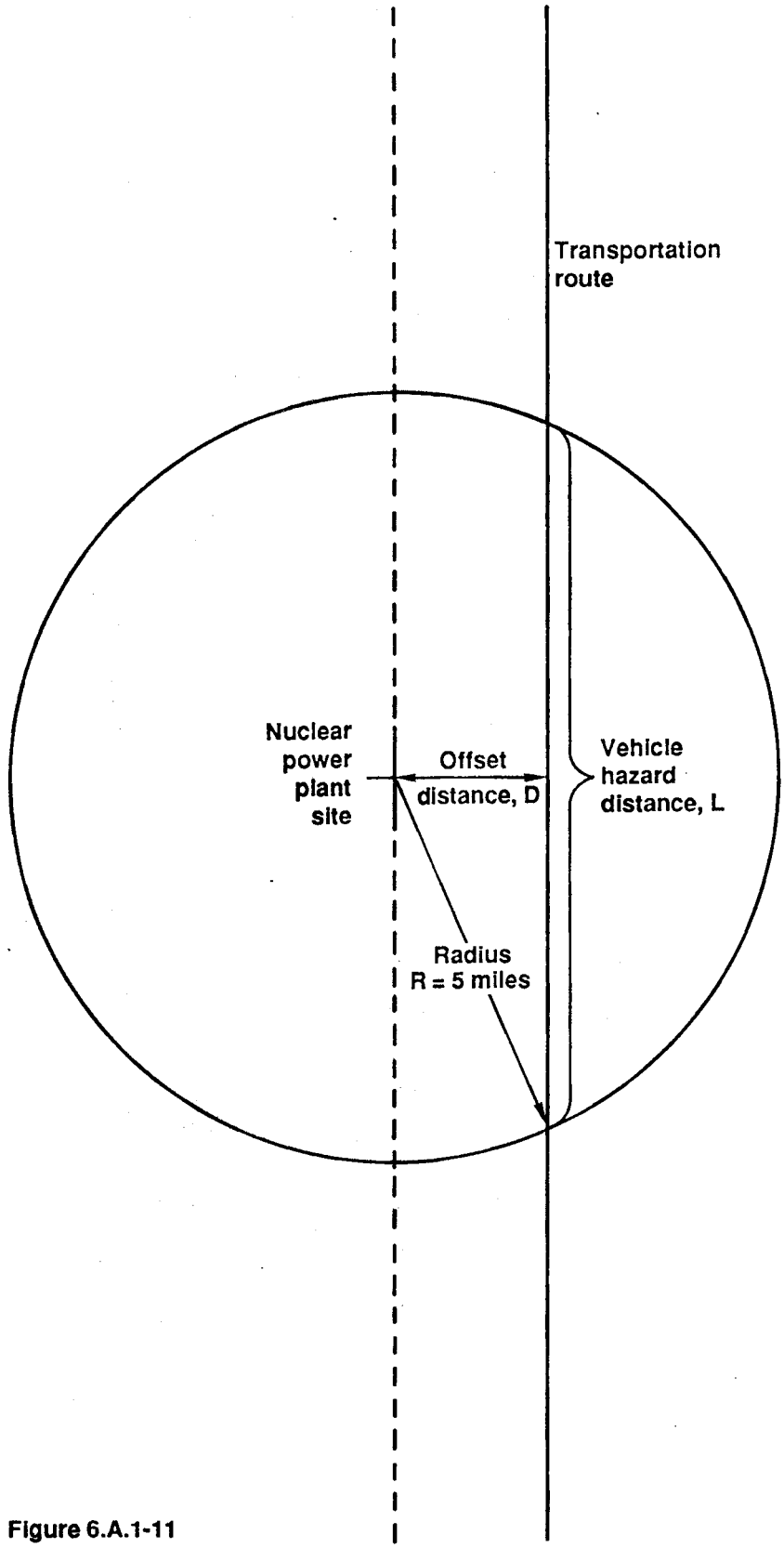


Figure 6.A.1-10



$$R^2 = \left(\frac{L}{2}\right)^2 + D^2$$

$$25 = \frac{L^2}{4} + D^2$$

$$L = (100 - 4D^2)^{1/2}$$

Figure 6.A.1-11

plant must be considered a hazard. If the offset distance, D, of a transportation route is 5 miles, then the vehicle hazard distance is zero miles because no vehicles travel within 5 miles of the plant. Solving for the vehicle hazard distance, L, in terms of the offset distance and radius:

Eq. 6.A.1 $$R^2 = (L/2)^2 + D^2$$

where R = 5 miles,
L = vehicle hazard distance,
D = offset distance.

Substituting for R:

$$25 = L^2/4 + D^2$$

Solving for L:

$$100 = L^2 + 4*D^2$$

Eq. 6.A.2 $$L = (100 - 4*D^2)^{1/2}$$

Table 6.A.1.1 presents the offset distance, D, and vehicle hazard distance, L, for D between zero and five miles.

Once the vehicle hazard distance for a transportation route near a plant site has been determined, then the number of vehicles that travel that route per year must be determined. The number of vehicle miles is then:

Eq. 6.A.3 Vehicle Miles per year = L * No. of Shipments per year.

Table 6.A.1.2 presents the vehicle miles per year for various vehicle hazard distances, L, versus the number of shipments per day. It is assumed that there is only one shipment per vehicle. 0.5 shipments per day corresponds to one shipment every two days or 3.5 shipments per week. 0.72 shipments per day corresponds to approximately five shipments per week. The total number of vehicle miles per year for all modes of transportation over all routes within five miles of a plant site is the summation of vehicle miles for all transportation modes over all transportation routes.

Finally, the frequency of transportation accidents within five miles of a plant site is then:

Eq. 6.A.4 Transportation Accidents per year w/i 5 miles of site = Vehicle Miles per year w/i 5 miles of site * Vehicle Accident Rate

The vehicle accident rate can be found for each mode of transportation in its relevant sections.

After determining the transportation accident frequency within five miles of a plant site, then the risk analysis methodology presented by Figures 6.A.1 to 6.A.10 can be used. The next factor to be determined is whether the vehicle hits the plant or misses the plant. Because only small shipments of hazardous materials are transported by aircraft, its potential for release of hazardous materials is discounted and therefore its risk to the plant is discounted should it miss the plant. For railroad and truck traffic, their movement within plant boundaries are controlled and infrequent so their potential for risk to the plant by actual collision with plant structures are discounted. However, if additional information reveals that this risk cannot be discounted, then these modes can be accomodated in this risk analysis model. Finally, a pipeline cannot move so its potential for collision with the plant is nil.

Should a vehicle hit a plant, then the possibility of damage to the plant must be determined. The definition of plant damage will probably need to be defined on a plant specific basis since only plant damage which can ultimately lead to core damage needs to be considered.

Next, the contents of the cargo carried by the vehicle must be determined. There are three possibilities. The vehicle carries a solid hazardous material such as dynamite or explosives. The vehicles carries a liquid or gaseous hazardous material such as petroleum products or chlorine. The last possibility is that the vehicle carries no hazardous cargo. The definition of hazardous material will need to be established at this point. The U.S. Department of Transportation (U.S. DOT) has an extensive definition for materials that must be considered hazardous [49 CFR Section 172]. This definition is probably not suitable since it include materials that are known or potential carcinogens, and hazardous wastes. Additional work and agreement within the industry will be needed on this issue.

For solid hazardous materials involved in a transportation accident, three possibilies exist. The material explodes, burns, or does neither of the first two. If the material explodes, then the possibility of damage to the plant must be explored. The guidance given by Reg. Guide 1.91 may be useful at this point although the possibility that important plant equipment may be damaged by overpressures less than one psi cannot be discounted. Even if the the plant is not damaged by the blast wave of the explosion, the possibility of toxic fumes and/or smoke that incapcitates the operators must be investigated. The possibility of operator incapacitation and/or damage to important plant equipment must also be studied if the solid material burns. The guidance given by Reg. Guide 1.78 could be used to aid determining the probability of operator incapacitation.

For liquid and gaseous hazardous materials, the methodology is basically the same as for solid hazardous materials except that the question of whether the material is flammable or toxic must be determined. The reason for this is that flammable materials will behave much like solid hazardous materials, e.g. they will detonate, burn or do neither of the first two. But, unlike solid hazardous materials, if the flammable material does not explode or burn,

the possibility of incapacitating the operator due to its volatility still exists. Toxic materials will be considered to be all materials that cannot detonate or burn yet still possess the capability to incapacitate the operator. Toxic materials that are capable of detonating or burning will be considered flammable.

Finally, the probability of core damage given damage or no damge to the plant caused by the vehicle, and given damage or no damage to the plant caused by explosions or fire, and given operator incapacitation or no operator incapacitation, can be determined at this point.

It should be noted that the risk analysis calculation does not have to be carried to its final step if at any point, the probability is less than the first figure of merit of 10-5 discussed in Chapter One of this report.

Table 6.A.1.1

Offset Distance Versus Vehicle Hazard Distance

Offset Distance, D		Vehicle Hazard Distance, L	
Miles	Feet	Miles	Feet
0.00	0	10.0	52,800
0.25	1,320	9.99	52,734
0.50	2,640	9.95	52,535
1.00	5,280	9.80	51,733
1.50	7,920	9.54	50,368
2.00	10,560	9.17	48,392
2.50	13,200	8.66	45,726
3.00	15,840	8.00	42,240
3.50	18,480	7.14	37,707
4.00	21,120	6.00	31,680
4.50	23,760	4.36	23,015
5.00	26,400	0.0	0

Table 6.A.1.2

Vehicle Miles Per Year as a Function of
Shipments Per Day and Vehicle Hazard Distance

Offset Distances, D (miles)

Shipments per day	0.0	0.5	1.0	1.5	2.0	2.5	3.0	3.5	4.0	4.5	5.0
	Vehicle Hazard Distance, L (miles)										
	10.00	9.95	9.80	9.54	9.17	8.66	8.00	7.14	6.00	4.36	0
	Vehicle Miles per Year										
0.5	1820	1811	1784	1736	1669	1576	1456	1299	1092	794	0
0.72	2621	2608	2568	2500	2403	2270	2097	1871	1572	1143	0
1	3640	3622	3567	3473	3338	3152	2912	2599	2184	1587	0
2	7280	7244	7134	6945	6676	6304	5824	5198	4368	3174	0
3	10920	10865	10702	10418	10014	9457	8736	7797	6552	4761	0
4	14560	14487	14269	13890	13352	12609	11648	10396	8736	6348	0
5	18200	18109	17836	17363	16689	15761	14560	12995	10920	7935	0
6	21840	21731	21403	20835	20027	18913	17472	15594	13104	9522	0
7	25480	25353	24970	24308	23365	22066	20384	18193	15288	11109	0
8	29120	28974	28538	27780	26703	25218	23296	20792	17472	12696	0
9	32760	32596	32105	31253	30041	28370	26208	23391	19656	14283	0
10	36400	36218	35672	34726	33379	31522	29120	25990	21840	15870	0
11	40040	39840	39239	38198	36717	34675	32032	28589	24024	17457	0
12	43680	43462	42806	41671	40055	37827	34944	31188	26208	19044	0
13	47320	47083	46374	45143	43392	40979	37856	33786	28392	20632	0
14	50960	50705	49941	48616	46730	44131	40768	36385	30576	22219	0
15	54600	54327	53508	52088	50068	47284	43680	38984	32760	23806	0
16	58240	57949	57075	55561	53406	50436	46592	41583	34944	25393	0
17	61880	61571	60642	59034	56744	53588	49504	44182	37128	26980	0
18	65520	65192	64210	62506	60082	56740	52416	46781	39312	28567	0
19	69160	68814	67777	65979	63420	59893	55328	49380	41496	30154	0
20	72800	72436	71344	69451	66758	63045	58240	51979	43680	31741	0
21	76440	76058	74911	72924	70095	66197	61152	54578	45864	33328	0
22	80080	79680	78478	76396	73433	69349	64064	57177	48048	34915	0
23	83720	83301	82046	79869	76771	72502	66976	59776	50232	36502	0
24	87360	86923	85613	83341	80109	75654	69888	62375	52416	38089	0
25	91000	90545	89180	86814	83447	78806	72800	64974	54600	39676	0
26	94640	94167	92747	90287	86785	81958	75712	67573	56784	41263	0
27	98280	97789	96314	93759	90123	85110	78624	70172	58968	42850	0
28	101920	101410	99882	97232	93461	88263	81536	72771	61152	44437	0
29	105560	105032	103449	100704	96799	91415	84448	75370	63336	46024	0
30	109200	108654	107016	104177	100136	94567	87360	77969	65520	47611	0

Table 6.A.1.2 (continued)

Vehicle Miles Per Year as a Function of
Shipments Per Day and Vehicle Hazard Distance

Offset Distances, D (miles)

Shipments per day	0.0	0.5	1.0	1.5	2.0	2.5	3.0	3.5	4.0	4.5	5.0
	Vehicle Hazard Distance, L (miles)										
	10.00	9.95	9.80	9.54	9.17	8.66	8.00	7.14	6.00	4.36	0
	Vehicle Miles per Year										
31	112840	112276	110583	107649	103474	97719	90272	80568	67704	49198	0
32	116480	115898	114150	111122	106812	100872	93184	83167	69888	50785	0
33	120120	119519	117718	114594	110150	104024	96096	85766	72072	52372	0
34	123760	123141	121285	118067	113488	107176	99008	88365	74256	53959	0
35	127400	126763	124852	121540	116826	110328	101920	90964	76440	55546	0
36	131040	130385	128419	125012	120164	113481	104832	93563	78624	57133	0
37	134680	134007	131986	128485	123502	116633	107744	96162	80808	58720	0
38	138320	137628	135554	131957	126839	119785	110656	98760	82992	60308	0
39	141960	141250	139121	135430	130177	122937	113568	101359	85176	61895	0
40	145600	144872	142688	138902	133515	126090	116480	103958	87360	63482	0
41	149240	148494	146255	142375	136853	129242	119392	106557	89544	65069	0
42	152880	152116	149822	145848	140191	132394	122304	109156	91728	66656	0
43	156520	155737	153390	149320	143529	135546	125216	111755	93912	68243	0
44	160160	159359	156957	152793	146867	138699	128128	114354	96096	69830	0
45	163800	162981	160524	156265	150205	141851	131040	116953	98280	71417	0
46	167440	166603	164091	159738	153542	145003	133952	119552	100464	73004	0
47	171080	170225	167658	163210	156880	148155	136864	122151	102648	74591	0
48	174720	173846	171226	166683	160218	151308	139776	124750	104832	76178	0
49	178360	177468	174793	170155	163556	154460	142688	127349	107016	77765	0
50	182000	181090	178360	173628	166894	157612	145600	129948	109200	79352	0
51	185640	184712	181927	177101	170232	160764	148512	132547	111384	80939	0
52	189280	188334	185494	180573	173570	163916	151424	135146	113568	82526	0
53	192920	191955	189062	184046	176908	167069	154336	137745	115752	84113	0
54	196560	195577	192629	187518	180246	170221	157248	140344	117936	85700	0
55	200200	199199	196196	190991	183583	173373	160160	142943	120120	87287	0
56	203840	202821	199763	194463	186921	176525	163072	145542	122304	88874	0
57	207480	206443	203330	197936	190259	179678	165984	148141	124488	90461	0
58	211120	210064	206898	201408	193597	182830	168896	150740	126672	92048	0
59	214760	213686	210465	204881	196935	185982	171808	153339	128856	93635	0
60	218400	217308	214032	208354	200273	189134	174720	155938	131040	95222	0

6.A.2 AVIATION ACCIDENT DATA

The Federal Aviation Administration (FAA) of the U.S. Department of Transportation (U.S. DOT) is responsible for the regulation of commercial and general aviation in the United States. The National Transportation Safety Board (NTSB), also apart of the U.S. DOT, investigates aviation accidents which have significant safety implications and issues accident reports on commercial and general aviation safety.

The NTSB defines an aircraft accident as occurrences incident to flight in which as a result of operation of the aircraft, any person receives fatal (an injury which results in death within 7 days of the accident) or serious injury or any aircraft receives substantial damage [Ref. 6.A.2.1 - 6.A.2.13, 6.A.2.25 - 6.A.2.42].

Substantial damage is defined by the NTSB for commercial and general aviation accidents after January 1, 1968 [Ref. 6.A.2.1 - 6.A.2.13, 6.A.2.27 - 6.A.2.42] as damage or structural failure which adversely affects the structural strength, performance, or flight characteristics of the aircraft and which would normally require major repair or replacement of the affected component. Engine failure, damage limited to an engine, bent fairings or cowling, dented skin, small punctured holes in the skin of fabric, ground damage to rotor or propeller blades, damage to landing gear, wheels, tires, flaps, engine accessories, brakes, or wingtips are not considered substantial damage. Prior to 1968, for general aviation accidents only, the NTSB defined substantial damage as damage or structural failure reasonably estimated to $300 or more to repair [Ref. 6.A.2.25 - 6.A.2.26].

Prior to 1978, the following definitions were used by the NTSB in compiling U.S. commercial aviation accident data.

An Air Carrier is defined as any air carrier operating under Title 14 of the Code of Federal Regulations, Section 121 (14 CFR 121). Included within this definition are operators who have been issued a Certificate of Public Convenience and Necessity (authorizes the carrier to engage in air transportation) by the Civil Aeronautics Board (CAB), i.e., certificated route air carriers and supplemental air carriers. Commercial operators of large aircraft are also classifed as air carriers [Ref. 6.A.2.1 - 6.A.2.8].

A Certificated Route Air Carrier is an air carrier holding a Certificate of Public Convenience and Necessity (authorizes the carrier to engage in air transportation) issued by the Civil Aeronautics Board (CAB) to conduct scheduled services over specified routes. Certain nonscheduled charter operations may also be conducted by these carriers [Ref. 6.A.2.1 - 6.A.2.8].

A Supplemental Air Carrier is an air carrier holding operating certificates isued by the Civil Aeronautics Board (CAB) authorizing it to perform charter passenger and cargo service to supplement the scheduled service of the certificated route air carriers [Ref. 6.A.2.1 - 6.A.2.8]. The FAA further expands this definition by stating that both international and

domestic charter operations are for a temporary period. The authority of Supplemental Air Carriers to engage in military charters is of an indefinite period. In addition, Supplemental Air Carriers can perform on an emergency basis, as may be authorized by the CAB, scheduled operations including the transportation of individually ticketed passengers and individually waybilled cargo [Ref. 6.A.2.43].

A Commercial Operator of Large Aircraft is an operator who has been granted a commercial operator operating certificate by the Federal Aviation Administration (FAA) to engage in the transportation of persons or property in commerce for compensation or hire with large aircraft (over 12,500 lbs.) under regulations specified by Title 14 of the Code of Federal Regulations, Section 121 (14 CFR 121) [Ref. 6.A.2.1 - 6.A.2.8].

In 1978, the Airline Deregulation Act was passed by Congress. This Act substantially changed the requirements by which commercial aviation was required to operate. The definition of an aircraft accident for commercial aviation was left unchanged but the accident data compilation was now done according to which section of 14 CFR the commercial air carrier operated under.

Section 121 of Title 14 of the Code of Federal Regulations (14 CFR 121) regulates the operation of aircraft used for commerce that have a maximum payload of greater than 7,500 lbs. or a maximum passenger seating configuration of greater than 30 seats.

Section 125 of Title 14 of the Code of Federal Regulations (14 CFR 125) regulates the operation of aircraft used for commerce that have a maximum payload of 7,500 lbs. or less but greater than 6,000 lbs. or a maximum passenger seating configuration of 30 seats or less but greater than 20 seats.

Section 135 of Title 14 of the Code of Federal Regulations (14 CFR 135) regulates the operation of aircraft used for commerce that have a maximum payload of less than 6,000 lbs. or a maximum passenger seating configuration of less than 20 seats.

An Air Taxi is defined by the FAA as that classification of air carriers which transports persons, property, and mail using small aircraft (under 30 seats or a maximum payload of 7,500 pounds). An air taxi does not hold a Certificate of Public Convenience and Necessity nor economic authority as issued by the Civil Aeronautics Board (CAB) [6.A.2.43].

A Commuter Air Carriers is defined by the FAA as an air taxi which performs at least five round trips per week between two or more points and publishes flight schedules which specify the times, days of the week, and points between which such flights are performed.

The FAA defines General Aviation as that portion of civil aviation (nonmilitary aviation) which encompasses all facets of aviation except air carriers [6.A.2.43]. Section 91 of Title 14 of the Code of Federal Regulations (14 CFR 91) regulates general aviation.

Table 6.A.2.1

Total U.S. Certificated Route, Supplemental, and Commercial Operators of Large Aircraft Accident Rate [Ref. 6.A.2.1 - 6.A.2.8]

Year	Total Acc.	Fatal Acc.*	Fatal.	A/C Hours Flown	A/C Miles (000)**	Avg. Miles /Hour	Accident Rate (per A/C Hour)	Accident Rate (per A/C Mile)
1963	77	13	264	4,126,399	1,231,312	298.40	1.866E-05	6.253E-08
1964	79	13	238	4,312,764	1,336,867	309.98	1.832E-05	5.909E-08
1965	83	9	261	4,690,882	1,536,395	327.53	1.769E-05	5.402E-08
1966	75	8	272	5,104,984	1,768,458	346.42	1.469E-05	4.241E-08
1967	70	12	286	5,868,842	2,179,739	371.41	1.193E-05	3.211E-08
1968	71	15	349	6,404,260	2,498,848	390.19	1.109E-05	2.841E-08
1969	63	10	158	6,740,199	2,736,596	406.01	9.347E-06	2.302E-08
1970	55	8	146	6,470,351	2,684,552	414.90	8.500E-06	2.049E-08
1971	48	8	203	6,386,662	2,660,731	416.61	7.516E-06	1.804E-08
1972	50	8	190	6,302,160	2,619,043	415.58	7.934E-06	1.909E-08
1973	43	9	227	6,504,819	2,646,669	406.88	6.610E-06	1.625E-08
1974	47	9	467	5,978,480	2,464,295	412.19	7.862E-06	1.907E-08
1975#	45	3	124	6,040,841	2,477,764	410.17	7.449E-06	1.816E-08
1976#	28	4	45	6,228,487	2,568,113	412.32	4.495E-06	1.090E-08
1977#	26	5	656	6,541,168	2,684,072	410.34	3.975E-06	9.687E-09
1978#	24	6	163	6,794,009	2,793,697	411.20	3.533E-06	8.591E-09
1979@	32	6	355	7,259,487	2,967,766	408.81	4.408E-06	1.078E-08
Total	916	146	4404	101,754,794	39,854,917			
Avg.	53.88	8.59	259.06	5,985,576	2,344,407		9.647E-06	2.663E-08
Std.	19.11	3.20	138.20	870,355	520,975		4.872E-06	1.699E-08

Accident Rate (/AC hour) based on totals: 9.002E-06

Accident Rate (/AC mile) based on totals: 2.298E-08

*Includes midair collisions nonfatal to air carrier occupants (1968--2, 1969--1, 1971--2).

**Nonrevenue miles of supplemental air carriers not reported 1966-78. Nonrevenue miles of supplemental air carriers included 1979.

#Accidents involving commercial operators of large aircraft included.

@Accidents involving commercial operators of large aircraft and All-Cargo Air Service Carriers included.

Table 6.A.2.2

U.S. Air Carriers Operating Under 14 CFR 121, 125, and 127, All Operations Accident Rate [Ref. 6.A.2.9 - 6.A.2.13]

Year	Total Acc.	Fatal Acc.	Fatal.	A/C Hours Flown	A/C Miles (000)	Avg. Miles /Hour	Accident Rate (per A/C Hour)	Accident Rate (per A/C Mile)
1975	37	3	124	5,607,358	N/A		6.598E-06	
1976	23	2	38	5,806,729	N/A		3.961E-06	
1977	24	5	655	6,039,707	N/A		3.974E-06	
1978	22	5	160	6,234,628	2,678,308	429.59	3.529E-06	8.214E-09
1979	29	5	354	6,878,911	2,922,226	424.81	4.216E-06	9.924E-09
1980	19	1	1	7,067,468	2,924,234	413.76	2.688E-06	6.497E-09
1981	26	4	4	6,810,255	2,798,575	410.94	3.818E-06	9.290E-09
1982	19	4	234	6,702,251	2,804,475	418.44	2.835E-06	6.775E-09
1983	24	4	15	6,930,564	2,922,583	421.69	3.463E-06	8.212E-09
1984	16	1	4	7,763,557	3,264,196	420.45	2.061E-06	4.902E-09
Total	239	34	1589	65,841,428				
Avg.	23.90	3.40	158.90	6,584,143			3.714E-06	7.688E-09
Std.	5.63	1.50	199.82	622,386			1.156E-06	1.610E-09

N/A = information not available at time of table preparation.

Table 6.A.2.3

U.S. Air Carriers Operating Under 14 CFR 135 Scheduled (Commuter) and Nonscheduled (On-Demand Air Taxi) Service Accident Rate [Ref. 6.A.2.9 - 6.A.2.13]

Year	Total Acc.	Fatal Acc.	Fatal.	A/C Hours Flown	A/C Miles (000)	Avg. Miles /Hour	Accident Rate (per A/C Hour)	Accident Rate (per A/C Mile)
1975	200	36	97	3,462,583	N/A		5.776E-05	
1976	172	40	127	3,668,499	N/A		4.689E-05	
1977	202	40	150	4,454,470	N/A		4.535E-05	
1978	259	68	203	4,847,889	N/A		5.343E-05	
1979	212	45	143	4,854,242	N/A		4.367E-05	
1980	208	53	140	4,793,312	N/A		4.339E-05	
1981	188	49	128	4,136,591	N/A		4.545E-05	
1982	159	36	86	4,556,511	N/A		3.490E-05	
1983	158	29	73	4,085,791	N/A		3.867E-05	
1984	168	30	100	4,824,769	N/A		3.482E-05	
Total	1926	426	1247	43,684,657				
Avg.	192.60	42.60	124.70	4,368,466			4.443E-05	
Std.	29.23	11.16	35.92	483,894			6.961E-06	

N/A = information not available at time of table preparation.

Table 6.A.2.4

U.S. Air Carriers Operating Under 14 CFR 135
Scheduled (Commuter) Service Accident Rate
[Ref. 6.A.2.9 - 6.A.2.13]

Year	Total Acc.	Fatal Acc.	Fatal.	A/C Hours Flown	A/C Miles (000)	Avg. Miles /Hour	Accident Rate (per A/C Hour)	Accident Rate (per A/C Mile)
1975	48	12	28	936,312	N/A		5.126E-05	
1976	35	9	27	965,296	N/A		3.626E-05	
1977	44	9	32	1,150,250	N/A		3.825E-05	
1978	61	14	48	1,302,136	N/A		4.685E-05	
1979	52	15	66	1,169,921	192,493	164.54	4.445E-05	2.701E-07
1980	38	8	37	1,175,588	192,200	163.49	3.232E-05	1.977E-07
1981	31	9	34	1,240,764	193,001	155.55	2.498E-05	1.606E-07
1982	27	5	14	1,299,748	222,355	171.08	2.077E-05	1.214E-07
1983	18	2	11	1,510,908	253,572	167.83	1.191E-05	7.099E-08
1984	22	7	48	1,745,762	291,460	166.95	1.260E-05	7.548E-08
Total	376	90	345	12,496,685				
Avg.	37.60	9.00	34.50	1,249,669			3.197E-05	1.494E-07
Std.	13.02	3.74	15.62	228,142			1.326E-05	7.004E-08

N/A = information not available at time of table preparation.

Table 6.A.2.5

U.S. Air Carriers Operating Under 14 CFR 135
Nonscheduled (On-Demand Air Taxi) Service Accident Rate
[Ref. 6.A.2.9 - 6.A.2.13]

Year	Total Acc.	Fatal Acc.	Fatal.	A/C Hours Flown	A/C Miles (000)	Avg. Miles /Hour	Accident Rate (per A/C Hour)	Accident Rate (per A/C Mile)
1975	152	24	69	2,526,271	N/A		6.017E-05	
1976	137	31	100	2,703,203	N/A		5.068E-05	
1977	158	31	118	3,304,220	N/A		4.782E-05	
1978	198	54	155	3,545,753	N/A		5.584E-05	
1979	160	30	77	3,684,321	N/A		4.343E-05	
1980	171	46	105	3,617,724	N/A		4.727E-05	
1981	157	40	94	2,895,827	N/A		5.422E-05	
1982	132	31	72	3,256,763	N/A		4.053E-05	
1983	140	27	62	2,574,883	N/A		5.437E-05	
1984	146	23	52	3,079,007	N/A		4.742E-05	
Total	1551	337	904	31,187,972				
Avg.	155.10	33.70	90.40	3,118,797			5.017E-05	
Std.	18.20	9.45	29.17	409,317			5.712E-06	

N/A = information not available at time of table preparation.

Table 6.A.2.6

Air Carrier Aircraft Characteristics
[Ref. 6.A.2.14 - 6.A.2.15]

Aircraft Model	Empty Weight (1000 lbs.)	T/O Weight (1000 lbs.)	Land. Weight (1000 lbs.)	Maximum Fuel Ld. (U.S. Gal.)	Max. Seats	Maximum Payload (1000 lbs)	Range w/ Max. Payload (nmiles)
Max. Wt. > 500,000 lbs.							
B-747SP	333	700	450	50,356	440	77.03	7,150
B-747-300	384.5	833	574	53,983	660	150.52	5,660
B-747-200F	341.02	833	630	53,983	Cargo	248.98	4,430
B-747-200C	387.67	833	630	53,983	550	202.33	4,430
B-747-200B	384.06	833	564	53,983	550	151.33	5,920
B-747-100B	374.91	750	564	48,072	500	151.59	4,430
DC-10-40	271	580	403	40,539	380	97	5,330
DC-10-30	267	580	403	40,539	380	101	5,350
Max. Wt. > 400,000 lbs., < 500,000 lbs.							
DC-10-10	243	440	363.5	26,566	380	92	2,950
L-1011-500	245.7	496	368	31,641	330	92.25	4,580
L-1011-200	248.6	466	368	26,419	400	71.41	4,260
L-1011-100	246.5	466	368	26,419	400	73.53	4,030
L-1011-1	241.7	430	358	23,582	400	83.27	2,950
Widebodies, Max. Wt. > 300,000 lbs., < 400,000 lbs.							
A300-600	191.26	375.88	308.64	19,167	375	95.34	2,310
A300B4-200	194.04	363.75	295.42	16,381	345	83.74	2,880
A300B4-100	194.2	347.2	295.42	16,381	345	79.17	2,470
A300B2-300	194.3	313.06	299.8	11,604	345	83.48	1,190
A300B2-200	192.2	313.06	295.42	11,604	345	81.17	1,350
A310-300	169.84	337.31	271.17	18,014	280	79.29	3,750
A310-200	169.2	313.06	267.86	16,381	280	76.61	2,790
B-767-300	188.8	345	300	16,693	330	90.1	2,780
B-737-200ER	180.6	351	278	20,452	290	73.3	4,340
B-767-200	176.2	315	272	16,693	290	74.7	3,030
Narrowbodies, Max. Wt. > 300,000 lbs., < 400,000 lbs.							
B-707-320C	147.8	333.6	247	23,853	219	53.9	5,175
DC-8-73	166.5	355	258	24,275	269	64.5	4,190
DC-8-72	151.7	350	240	24,275	201	44.3	5,820
DC-8-71	165.3	325	240	23,392	269	59.7	3,400
DC-8-63	158.3	355	245	24,277	259	67.74	4,245
DC-8-50		325					
Max. Wt. > 200,000 lbs., < 300,000 lbs.							
B-757-200	126.47	240	198	11,253	239	58.09	3,320
B-727-200	101.77	209.5	160	10,585	189	41	2,900
B-720B	115	234	175	14,892	149	12	5,600
CV-990A		253					
CV-880							

Table 6.A.2.6 (continued)

Aircraft Model	Empty Weight (1000 lbs.)	T/O Weight (1000 lbs.)	Land. Weight (1000 lbs.)	Maximum Fuel Ld. (U.S. Gal.)	Max. Seats	Maximum Payload (1000 lbs)	Range w/ Max. Payload (nmiles)
Max. Wt. > 100,000 lbs., < 200,000 lbs.							
A320-100	84.17	158.7	138.8	9,788	179	41.53	1,860
B-737-300	69.73	124.5	114	6,121	149	35.27	1,580
B-737-200	60.67	115.5	105	5,970	130	34.33	1,550
B-737-100	60.66	115.5	105	5,970	130	34.33	1,550
B-727-100	88	169.5	142.5	10,706	131	35.5	2,250
BAC-111-500	55.04	99.65	87	4,546	119	25.93	1,500
BAe-146-300	56.5	104	90	3,408	122	26.5	1,660
MD-87	73.16	140	128	5,840	139	38.84	1,915
MD-83	80.23	160	139.5	7,003	172	41.27	2,370
MD-82	78.55	149.5	130	5,840	172	41.45	1,860
MD-81	78.42	140	128	5,840	172	39.58	1,385
DC-9-50	65	121	110	5,032	139	33	1,375
DC-9-40	59	114	102	3,675	125	36.5	1,175
DC-9-30	56	108	99	3,675	115	30.2	1,370
DC-9-20	52.5	100	95.3	3,675	90	31.5	1,090
L-188 Electra		116					
L-100-30 Hercules	76.13	155	135	9,675	Cargo	50.82	1,770
L-100-20 Hercules	75.07	155	130	9,675	Cargo	51.93	1,770
Max. Wt. < 100,000 lbs., Part 121 Aircraft							
BAe-146-200	50.5	93	81	3,408	106	23	1,610
BAe-146-100	49	84	77.5	3,408	88	19.5	1,370
CV-580/600		50.14					
CV-240/340/440		49.7					
DHC-8-300	24.2	39.6	39	1,505	56	12	N/A
DHC-7-100	27.69	47	45	1,480	50	11.38	420
F.100	51.26	91.5	84.5	3,446	107	25.24	1,090
F.28 Mk4000	38.68	69.5	62	3,446	85	23.31	940
F.28 Mk3000	36.99	64	56	3,446	65	19.01	1,570
F.27/FH-227	25.99	45	42	1,976	56	13.16	825
M-404	29.13	44.9					
Max. Wt. < 100,000 lbs., Part 125 Aircraft							
CASA 212-300	10.65	16.98	16.42	528	26	5	346
DHC Twin Otter	7.44	12.5	12.3	471	20	4.86	50
EMB-120	15.5	25.35	24.8	875	30	7.65	350
FH Metro III (H)	9.22	16	15.5	649	20	4.68	940
IAI Arava 101B	8.82	15	15	980	20	5.18	237
N262/MO-298	15.93	23.81	23.72	681	29	6.78	270
Max. Wt. < 100,000 lbs., Part 135 Aircraft							
BH-1900C	9.5	16.6	16.1	510	19	4.5	905
BH-C99	7.04	11.3	11.3	448	15	3.0	4,475
EMB-123	11.02	16.98	16.64	346	19	4.41	300
EMB-110 P1/41	8.56	13	12.57	440	19	3.45	310

Table 6.A.2.6 (continued)

A = Airbus Industrie
B = Boeing
BAe = British Aerospace
BH = Beechcraft
CV = Convair
DC = Douglas
DHC = De Havilland Canada
EMB = Embraer
F = Fokker
FH = Fairchild
IAI = Israel Aircraft Industries
L = Lockheed
M = Martin
MD = McDonnell Douglas
MO = Mohawk
N = Nord Aviation

Table 6.A.2.7

Harrisburg International Airport, Harrisburg, PA
Total (Commercial) Departures Performed
[Ref. 6.A.2.16 - 6.A.2.24]

Aircraft Type	CY 1985	CY 1984	CY 1983	CY 1982	CY 1981	CY 1980	CY 1979	CY 1978	CY 1977	Total
> 500,000 lbs.										
B-747	1	0	2	0	0	0	0	0	0	3
DC-10	0	0	0	0	0	0	0	0	0	0
Subtotal	1	0	2	0	0	0	0	0	0	3
Pct.	.0%	0.0%	0.1%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	.0%
> 400,000 lbs., < 500,000 lbs.										
L-1011	0	0	0	0	0	0	0	0	2	2
Subtotal	0	0	0	0	0	0	0	0	2	2
Pct.	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%
> 300,000 lbs., < 400,000 lbs.										
B-707	0	0	31	379	362	222	301	240	106	1641
B-767	0	0	0	0	0	0	0	0	0	0
DC-8	0	0	2	2	0	0	0	0	0	4
Subtotal	0	0	33	381	362	222	301	240	106	1645
Pct.	0.0%	0.0%	0.6%	6.6%	6.3%	3.8%	5.2%	4.1%	1.8%	28.4%
> 200,000 lbs., < 300,000 lbs.										
B-757	0	0	0	0	0	0	0	0	0	0
B-727	2481	2075	933	895	838	1226	1058	1146	956	11608
B-720	0	0	0	0	0	0	0	0	0	0
CV-990	0	0	0	0	0	0	0	0	0	0
CV-880	0	0	0	0	0	0	0	0	0	0
Subtotal	2481	2075	933	895	838	1226	1058	1146	956	11608
Pct.	42.8%	48.4%	27.5%	28.5%	24.0%	27.8%	17.9%	18.4%	17.3%	27.5%
> 100,000 lbs., < 200,000 lbs.										
B-737	1799	890	984	215	0	0	0	0	0	3888
BAC-111	135	70	230	123	138	363	354	135	280	1828
DC-9/MD-80	1375	1248	1214	1529	2147	2327	2558	2618	2929	17945
Subtotal	3309	2208	2428	1867	2285	2690	2912	2753	3209	23661
Pct.	57.1%	51.6%	71.5%	59.4%	65.6%	61.0%	49.3%	44.2%	57.9%	56.1%
< 100,000 lbs.										
BH-99	0	0	0	0	0	273	992	0	0	1265
CV-580	0	0	0	0	0	0	0	8	333	341
N-262/MO-298	0	0	0	0	0	0	646	2082	933	3661
Subtotal	0	0	0	0	0	273	1638	2090	1266	5267
Pct.	0.0%	0.0%	0.0%	0.0%	0.0%	6.2%	27.7%	33.6%	22.9%	12.5%
Total	5791	4283	3396	3143	3485	4411	5909	6229	5539	42186
Average:										4687
Std. Dev.										1133

Table 6.A.2.8

General Aviation Accident Rate 1959-1983
All Operations [Ref. 6.A.2.25 - 6.A.2.42]

Year	Total Acc.	Fatal Acc.	Fatal. (1)	A/C Hours Flown (000)	Aircraft Miles Flown (000)	Avg. Miles/ Hours	Accident Rate (per A/C Hour)	Accident Rate (per A/C Mile)
1959	4,576	450	823	12,903	1,716,019	132.99	3.546E-04	2.667E-06
1960	4,793	429	787	13,121	1,768,704	134.80	3.653E-04	2.710E-06
1961	4,625	426	761	13,602	1,857,946	136.59	3.400E-04	2.489E-06
1962	4,840	430	857	14,500	1,964,586	135.49	3.338E-04	2.464E-06
1963	4,690	482	893	15,106	2,048,574	135.61	3.105E-04	2.289E-06
1964	5,069	526	1,083	15,738	2,180,818	138.57	3.221E-04	2.324E-06
1965	5,196	538	1,029	16,733	2,562,380	153.13	3.105E-04	2.028E-06
1966	5,712	573	1,149	21,023	3,336,138	158.69	2.717E-04	1.712E-06
1967	6,115	603	1,229	22,153	3,439,964	155.28	2.760E-04	1.778E-06
1968	4,968	692	1,399	24,053	3,700,864	153.86	2.065E-04	1.342E-06
1969	4,767	647	1,413	25,351	3,926,461	154.88	1.880E-04	1.214E-06
1970	4,712	641	1,310	26,030	3,207,127	123.21	1.810E-04	1.469E-06
1971	4,648	661	1,355	25,512	3,143,181	123.20	1.822E-04	1.479E-06
1972	4,256	695	1,426	26,974	3,317,100	122.97	1.578E-04	1.283E-06
1973	4,255	723	1,412	29,974	3,686,802	123.00	1.420E-04	1.154E-06
1974	4,425	729	1,438	31,413	3,863,799	123.00	1.409E-04	1.145E-06
1975	4,237	675	1,345	32,024	3,938,952	123.00	1.323E-04	1.076E-06
1976	4,193	695	1,320	33,922	4,172,406	123.00	1.236E-04	1.005E-06
1977	4,286	702	1,436	35,791	4,402,126	123.00	1.198E-04	9.736E-07
1978	4,494	793	1,770	39,409	4,964,400	125.97	1.140E-04	9.052E-07
1979	4,023	678	1,367	43,340	5,590,883	129.00	9.282E-05	7.196E-07
1980#	3,597	622	1,252	36,402	4,695,858	*129.00	9.881E-05	7.660E-07
1981#	3,502	654	1,282	36,803	4,747,587	*129.00	9.516E-05	7.376E-07
1982#	3,231	589	1,182	32,095	4,140,255	*129.00	1.007E-04	7.804E-07
1983#	3,075	555	1,064	31,048	4,005,192	*129.00	9.904E-05	7.678E-07
Tot.	112,285	15,208	30,382	655,020	86,378,122			
Avg.	4,491	608	1,215	26,201	3,455,125		2.024E-04	1.491E-06
Std.	679	103	245	8,911	1,062,121		9.495E-05	6.534E-07

(1) Fatalities exclude air carrier deaths when in collision with general aviation aircraft (1966-2, 1967-104, 1969-82, 1972-5, 1978-142).

#Aircraft miles flown and aircraft accident rate (per A/C Mile) calculated from assumed value.

*Assumed value.

Data compiled prior to 1975 included commercial operators of large aircraft.

Data compiled prior to 1980 included U.S. Air Carriers operating under 14 CFR 135, both scheduled (commuter) and nonscheduled (on-demand air taxi) service.

Table 6.A.2.9

General Aviation Accidents, By Flight Phase
[Ref. 6.A.2.30 - 6.A.2.38]

Year	Static	Taxi	Takeoff	Inflight, Cruise, Descent, Mnuvrng.	Landing, Approach	Unknown, Not Reported	Total
1971	43	180	896	1,412	2,121	24	4,676
1972	47	169	752	1,342	1,937	30	4,277
1973	36	171	770	1,418	1,858	28	4,281
1974	52	163	888	1,433	1,880	9	4,425
1975	21	177	828	1,431	1,760	20	4,237
1976	30	152	781	1,469	1,766	16	4,214
1977	25	147	873	1,451	1,768	22	4,286
1978	37	174	889	1,514	1,854	26	4,494
1979	37	129	805	1,392	1,621	39	4,023
Total	328	1,462	7,482	12,862	16,565	214	38,913
Avg.	36.44	162.44	831.33	1429.11	1840.56	23.78	4323.67
Std.	9.48	15.76	53.49	45.55	131.45	8.09	176.00
Pct.	0.84%	3.76%	19.23%	33.05%	42.57%	0.55%	100.00%

Table 6.A.2.10

General Aviation Accidents
Aircraft Involved, By Flight Phase
[Ref. 6.A.2.30 - 6.A.2.42]

Year	Static	Taxi	Takeoff	Inflight, Cruise, Descent, Mnuvrng.	Landing, Approach	Unknown, Not Reported	Total
1971	43	184	896	1,415	2,134	27	4,699
1972	47	171	752	1,348	1,949	33	4,300
1973	38	173	771	1,425	1,869	29	4,305
1974	52	166	888	1,444	1,904	30	4,484
1975	21	178	828	1,440	1,776	45	4,288
1976	30	152	782	1,481	1,780	16	4,241
1977	25	150	873	1,464	1,793	32	4,337
1978	38	175	889	1,528	1,880	47	4,557
1979	37	131	807	1,402	1,643	43	4,063
1980	23	109	706	1,241	1,519	30	3,628
1981	11	89	727	1,267	1,415	25	3,534
1982	24	109	682	1,513	881	61	3,270
1983	39	82	661	1,075	1,198	52	3,107
Total	428	1,869	10,262	18,043	21,741	470	52,813
Avg.	32.92	143.77	789.38	1387.92	1672.38	36.15	4062.54
Std.	11.23	34.37	78.67	122.03	328.62	12.02	489.09
Pct.	0.81%	3.54%	19.43%	34.16%	41.17%	0.89%	100.00%

Table 6.A.2.11

General Aviation Accidents, Airport Proximity
[Ref. 6.A.2.25 - 6.A.2.38]

Year	Air-port	Oth. Land. Areas (1)	Traf. Pat-tern	<0.5 Mile	0.5 to 1 Mile	1-2 Mi.	2-3 Mi.	3-4 Mi.	4-5 Mi.	>5.0 Mi.	Unkn. Other	Total
1966	3392	41	240	254	92	143	106	94	28	1268	54	5712
1967	3613	29	394	157	102	150	128	91	30	1326	95	6115
1968	2538	18	602	96	82	151	119	72	25	1211	54	4968
1969	2347	15	455	140	90	136	116	88	39	1283	58	4767
1970	2318	20	296	234	116	137	116	100	40	1270	65	4712
1971	2344	9	241	212	117	131	129	108	27	1293	37	4648
1972	1994	18	234	254	113	151	129	107	51	1148	58	4257
1973	1983	20	167	276	135	143	117	66	50	1135	163	4255
1974	2056	21	174	340	112	128	76	51	33	1193	241	4425
1975	1838	30	227	275	121	116	69	47	43	1266	205	4237
1976	1814	23	249	282	117	99	62	36	38	1293	182	4195
1977	1906	22	152	360	135	112	70	43	36	1180	270	4286
1978	2019	35	145	348	128	112	77	48	48	1250	284	4494
1979	1789	33	114	332	105	141	90	45	37	1104	233	4023
Tot.	31951	334	3690	3560	1565	1850	1404	996	525	17220	1999	65094
Avg.	2282	24	264	254	112	132	100	71	38	1230	143	4650
Std.	545.9	8.3	130.6	77.8	15.6	16.0	24.2	25.3	8.1	65.8	88.4	578.3
Pct.	49.1%	0.5%	5.7%	5.5%	2.4%	2.8%	2.2%	1.5%	0.8%	26.5%	3.1%	100.0%

(1) Other Land. Areas include seaplane base, heliport, and barge/ship platform.

Table 6.A.2.12

General Aviation Accidents, Airport Proximity
Aircraft Involved, 1973-1981 [Ref. 6.A.2.32 - 6.A.2.40]

Year	Air port	Other Land. Areas (1)	Traf. Pat-tern	>0.5 Mile	0.5 to 1 Mile	1-2 Mi.	2-3 Mi.	3-4 Mi.	4-5 Mi.	>5.0 Mi.	Unkn. Other	Total
1973	2015	20	170	277	136	143	117	67	50	1143	167	4305
1974	2085	21	178	341	114	131	78	53	36	1200	246	4483
1975	1867	30	233	275	121	116	72	48	44	1275	207	4288
1976	1839	23	252	284	118	99	62	37	38	1306	183	4241
1977	1933	22	155	363	135	112	71	44	38	1189	275	4337
1978	2056	36	148	351	128	115	78	48	48	1260	289	4557
1979	1810	33	116	333	106	142	90	46	39	1110	238	4063
1980	1529	24	128	345	114	103	64	40	32	1004	245	3628
1981	1434	20	215	281	119	135	75	43	31	930	251	3534
Tot.	16568	229	1595	2850	1091	1096	707	426	356	10417	2101	37436
Avg.	1841	25	177	317	121	122	79	47	40	1157	233	4160
Std.	213.2	5.7	44.3	34.4	9.5	15.5	15.7	8.2	6.2	118.8	38.1	336.9
Pct.	44.3%	0.6%	4.3%	7.6%	2.9%	2.9%	1.9%	1.1%	1.0%	27.8%	5.6%	100.0%

(1) Other Land. Areas include seaplane base, heliport, and barge/ship platform.

Table 6.A.2.13

General Aviation Accidents, Airport Proximity
Aircraft Involved, 1982-1983 [Ref. 6.A.2.41 - 6.A.2.42]

Year	Air-port	On Air-strip	<5.0 Mile (1)	>5.0 Mile (1)	Other	Total
1982	1314	158	702	1031	65	3270
1983	1213	177	417	251	1049	3107
Tot.	2527	335	1119	1282	1114	6377
Avg.	1264	168	560	641	557	3189
Std.	50.5	9.5	142.5	390	492	81.5
Pct.	39.6%	5.3%	17.5%	20.1%	17.5%	100.0%

(1) 1982 data given for nautical miles, 6076 ft.
1983 data given for statute miles, 5280 ft.

Commercial Aviation Accident Data References

[6.A.2.1] NTSB-ARC-74-1 Annual Reviews of Aircraft Accident Data, U.S. Air Carrier Operations 1970-72, Bureau of Aviation Safety, National Transportation Safety Board, Washington, D.C. April 11, 1974.

[6.A.2.2] NTSB-ARC-74-2 Annual Reviews of Aircraft Accident Data, U.S. Air Carrier Operations 1973, Bureau of Aviation Safety, National Transportation Safety Board, Washington, D.C. October 24, 1974.

[6.A.2.3] NTSB-ARC-76-1 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations 1974, Bureau of Aviation Safety, National Transportation Safety Board, Washington, D.C. December 30, 1975.

[6.A.2.4] NTSB-ARC-77-1 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations 1975, Bureau of Technology, National Transportation Safety Board, Washington, D.C. January 25, 1977.

[6.A.2.5] NTSB-ARC-78-1 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations 1976, Bureau of Technology, National Transportation Safety Board, Washington, D.C. January 5, 1978.

[6.A.2.6] NTSB-ARC-78-2 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations 1977, Bureau of Technology, National Transportation Safety Board, Washington, D.C. September 6, 1978.

[6.A.2.7] NTSB-ARC-80-1 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations 1978, Bureau of Technology, National Transportation Safety Board, Washington, D.C. October 6, 1980.

[6.A.2.8] NTSB-ARC-81-1 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations 1979, Bureau of Technology, National Transportation Safety Board, Washington, D.C. November 16, 1981.

[6.A.2.9] NTSB/ARC-83/01 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations 1980, Bureau of Safety Programs, National Transportation Safety Board, Washington, D.C. January 14, 1983.

[6.A.2.10] NTSB/ARC-85/01 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations Calender Year 1981, Bureau of Safety Programs, National Transportation Safety Board, Washington, D.C. February 1, 1985.

[6.A.2.11] NTSB/ARC-86/01 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations Calender Year 1982, Bureau of Safety Programs, National Transportation Safety Board, Washington, D.C. (No Date).

[6.A.2.12] NTSB/ARC-87/01 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations Calender Year 1983, Bureau of Safety Programs, National Transportation Safety Board, Washington, D.C. February 13, 1987.

[6.A.2.13] NTSB/ARC-87/02 Annual Review of Aircraft Accident Data, U.S. Air Carrier Operations Calender Year 1984, Bureau of Safety Programs, National Transportation Safety Board, Washington, D.C. April 15, 1987.

[6.A.2.14] "Commercial Aircraft of the World", Flight International, various issues.

[6.A.2.15] An Illustrated Guide to the World's Airliners, William Green, Gordon Swanborough, Arco Publishing, Inc., New York, NY, c1982.

[6.A.2.16] Airport Activity Statistics of Certificated Route Air Carriers, 12 Months Ending Dec. 31, 1977, U.S. Dept. of Transportation, Federal Aviation Administration, Center for Transportation Information, Cambridge, MA.

[6.A.2.17] Airport Activity Statistics of Certificated Route Air Carriers, 12 Months Ending Dec. 31, 1978, U.S. Dept. of Transportation, Federal Aviation Administration, Center for Transportation Information, Cambridge, MA.

[6.A.2.18] Airport Activity Statistics of Certificated Route Air Carriers, 12 Months Ending Dec. 31, 1979, U.S. Dept. of Transportation, Federal Aviation Administration, Center for Transportation Information, Cambridge, MA.

[6.A.2.19] Airport Activity Statistics of Certificated Route Air Carriers, 12 Months Ending Dec. 31, 1980, U.S. Dept. of Transportation, Federal Aviation Administration, Center for Transportation Information, Cambridge, MA.

[6.A.2.20] Airport Activity Statistics of Certificated Route Air Carriers, 12 Months Ending Dec. 31, 1981, U.S. Dept. of Transportation, Federal Aviation Administration, Center for Transportation Information, Cambridge, MA.

[6.A.2.21] Airport Activity Statistics of Certificated Route Air Carriers, 12 Months Ending Dec. 31, 1982, U.S. Dept. of Transportation, Federal Aviation Administration, Center for Transportation Information, Cambridge, MA.

[6.A.2.22] Airport Activity Statistics of Certificated Route Air Carriers, 12 Months Ending Dec. 31, 1983, U.S. Dept. of Transportation, Federal Aviation Administration, Center for Transportation Information, Cambridge, MA.

[6.A.2.23] Airport Activity Statistics of Certificated Route Air Carriers, 12 Months Ending Dec. 31, 1984, U.S. Dept. of Transportation, Federal Aviation Administration, Center for Transportation Information, Cambridge, MA.

[6.A.2.24] Airport Activity Statistics of Certificated Route Air Carriers, 12 Months Ending Dec. 31, 1985, U.S. Dept. of Transportation, Federal Aviation Administration, Center for Transportation Information, Cambridge, MA.

General Aviation Accident Data References

[6.A.2.25] Annual Review of U.S. General Aviation Accidents Occuring in Calender Year 1966, National Transportation Safety Board, Washington, D.C. November 1967.

[6.A.2.26] Annual Review of U.S. General Aviation Accidents Occuring in Calender Year 1967, National Transportation Safety Board, Washington, D.C. October 1968.

[6.A.2.27] Annual Review of U.S. General Aviation Accidents Occuring in Calender Year 1968, National Transportation Safety Board, Washington, D.C. September 1969.

[6.A.2.28] NTSB-ARG-71-1 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1969, Bureau of Aviation Safety, National Transportation Safety Board, Washington, D.C. April 28, 1971.

[6.A.2.29] NTSB-ARG-74-1 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1970, Bureau of Aviation Safety, National Transportation Safety Board, Washington, D.C. April 18, 1974.

[6.A.2.30] NTSB-ARG-74-2 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1971, Bureau of Aviation Safety, National Transportation Safety Board, Washington, D.C. May 29, 1974.

[6.A.2.31] NTSB-ARG-74-3 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1972, Bureau of Aviation Safety, National Transportation Safety Board, Washington, D.C. November 14, 1974.

[6.A.2.32] NTSB-ARG-75-1 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1973, Bureau of Aviation Safety, National Transportation Safety Board, Washington, D.C. July 25, 1975.

[6.A.2.33] NTSB-ARG-76-1 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1974, Bureau of Aviation Safety, National Transportation Safety Board, Washington, D.C. January 31, 1976.

[6.A.2.34] NTSB-ARG-77-1 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1975, Bureau of Technology, National Transportation Safety Board, Washington, D.C. January 25, 1977.

[6.A.2.35] NTSB-ARG-78-1 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1976, Bureau of Technology, National Transportation Safety Board, Washington, D.C. March 8, 1978.

[6.A.2.36] NTSB-ARG-78-2 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1977, Bureau of Technology, National Transportation Safety Board, Washington, D.C. November 16, 1978.

[6.A.2.37] NTSB-ARG-80-1 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1978, Bureau of Technology, National Transportation Safety Board, Washington, D.C. May 20, 1980.

[6.A.2.38] NTSB-ARG-81-1 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1979, Bureau of Technology, National Transportation Safety Board, Washington, D.C. November 5, 1981.

[6.A.2.39] NTSB/ARG-84/01 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1980, Bureau of Safety Programs, National Transportation Safety Board, Washington, D.C. April 12, 1984.

[6.A.2.40] NTSB/ARG-84/02 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1981, Bureau of Safety Programs, National Transportation Safety Board, Washington, D.C. June 8, 1984.

[6.A.2.41] NTSB/ARG-86/01 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1982, Bureau of Safety Programs, National Transportation Safety Board, Washington, D.C. February 16, 1986.

[6.A.2.42] NTSB/ARG-87/01 Annual Review of Aircraft Accident Data, U.S. General Aviation Calender Year 1983, Bureau of Safety Programs, National Transportation Safety Board, Washington, D.C. January 8, 1987.

6.A.3 MARINE ACCIDENT DATA

Table 6.A.3.1

U.S. Water Transport Industry Vehicles
[Ref. 6.A.3.1]

Year	Non-self-propelled Vessels			Self-propelled Vessels, Towboats & Tugs	Total Inland Water Vessels	Ocean-going Vessels, 1000 gross tons & over
	Dry Cargo Barges & Scows	Tank Barges	Total			
1974	19,772	3,375	23,147	4,035	27,182	922
1975	21,876	3,534	25,410	4,100	29,510	857
1976	23,164	3,623	26,787	4,240	31,027	842
1977	24,937	3,770	28,707	4,379	33,086	840
1978	24,037	3,946	27,983	4,380	32,363	879
1979	25,420	4,000	29,420	4,492	33,912	865
1980	27,426	4,166	31,592	4,693	36,285	864
1981	*	*	*	*	*	853
1982	29,479	4,413	33,892	4,890	38,782	832
1983	*	*	*	*	*	788
1984	29,730	4,114	33,844	4,993	38,837	744

* Beginning in 1981 data collected every 2 years.

Table 6.A.3.2

U.S. Water Transport Industry
[Ref. 6.A.3.1]

Year	Inland Waterways (Miles)	Freight Revenue Ton-Miles (000,000) Inland Waterways	Freight Revenue Ton-Miles (000,000) Total Domestic Waterways	Crude Oil Transported Ton-Miles (1 E9)	Refined Petroleum Products Transported Ton-Miles (1 E9)	Total Petroleum Products Transported Ton-Miles (1 E9)
1974	25,543	354,882	586,345	53.0	244.0	297.0
1975	25,543	342,210	565,984	40.6	257.4	298.0
1976	25,543	372,865	591,853	37.8	269.1	306.9
1977	25,543	368,275	599,000	63.1	270.2	333.3
1978	25,543	409,316	827,263	261.3	269.3	530.6
1979	25,543	424,569	828,760	265.5	257.4	522.9
1980	25,543	406,879	921,836	387.4	230.4	617.8
1981	25,543	410,240	929,413	404.9	212.3	617.2
1982	25,543	351,280	886,469	432.7	184.2	616.9
1983	25,543	359,013	919,566	471.2	159.3	630.5
1984	25,543	398,879	887,720	412.6	158.1	570.7
Avg.		381,673	776,746	257.3	228.3	485.6
Std.		27,504	147,962	169.0	41.5	138.0
Max.		424,569	929,413	471.2	270.2	630.5
Min.		342,210	565,984	37.8	158.1	297.0

Marine Accident References

[6.A.3.1] DOT-TSC-RSPA-86-3 National Transporation Statistics Annual Report, 1986, Kathleen Bradley, U.S. Department of Transportation, Research and Special Programs Administration, Transportation Systems Center, Cambridge, MA. July 1986.

6.A.4 PIPELINE ACCIDENT DATA

Table 6.A.4.1

U.S. Petroleum Pipeline Mileage
[Ref. 6.A.4.1]

Year	Refined Oil Trunk Lines (miles)		Crude Oil Gathering Lines (miles)		Crude Oil Trunk Lines (miles)		Total Petroleum Pipelines (miles)	
	FERC	All	FERC	All	FERC	All	FERC	All
1970	59,335	72,396	46,587	71,132	63,030	75,143	168,952	218,671
1971	61,525	74,277	45,759	70,110	60,946	75,512	168,230	219,899
1972	64,701	76,158	42,893	69,088	59,757	75,881	167,351	221,127
1973	64,919	78,038	41,655	69,247	57,435	76,250	164,009	223,535
1974	68,609	79,124	41,577	68,764	57,602	76,824	167,788	224,712
1975	66,620	80,210	42,582	68,281	54,658	77,398	163,860	225,889
1976	67,913	81,296	39,235	67,798	58,544	77,972	165,692	227,066
1977	60,099	74,995	34,703	66,580	59,739	78,483	154,541	220,058
1978	65,114	77,314	36,539	65,368	59,981	75,483	161,634	218,165
1979	74,261	85,905	36,927	58,179	58,606	71,876	169,794	215,960
1980	74,510	88,562	35,279	58,263	59,560	71,568	169,349	218,393
1981	76,353	89,456	38,558	57,099	57,904	68,486	172,815	215,041
1982	77,402	90,727	35,580	53,421	59,567	69,529	172,549	213,677
1983	79,387	93,054	30,966	47,688	57,466	67,077	167,819	207,819
1984	80,875	94,822	36,072	47,288	56,975	66,540	173,922	208,650
Avg.	69,442	82,422	38,994	62,554	58,785	73,601	167,220	218,577
Std.	6,901	7,112	4,263	7,941	1,885	3,931	4,736	5,514
Max.	80,875	94,822	46,587	71,132	63,030	78,483	173,922	227,066
Min.	59,335	72,396	30,966	47,288	54,658	66,540	154,541	207,819

FERC = pipelines regulated by Federal Energy Regulatory Commission. Prior to 1976, these pipelines were regulated by Interstate Commerce Commission (ICC).

Table 6.A.4.2

U.S. Gas Pipeline Mileage
[Ref. 6.A.4.1]

Year	Field and Gathering Lines (miles)	Transmission Pipelines (miles)	Distribution Main Lines (miles)	Total Gas Pipelines (miles)
1965	61,700	211,300	494,500	767,500
1966	63,000	217,000	519,600	799,600
1967	63,700	225,400	539,200	828,300
1968	64,400	234,500	562,700	861,600
1969	64,900	248,100	578,600	891,600
1970	66,300	252,200	594,800	913,300
1971	66,200	254,800	610,400	931,400
1972	66,900	258,100	623,100	948,100
1973	65,900	263,100	633,800	962,800
1974	66,400	262,200	645,600	974,200
1975	68,500	262,600	648,200	979,300
1976	70,300	258,200	659,100	987,600
1977	71,500	260,500	666,900	998,900
1978	74,900	260,600	677,500	1,013,000
1979	77,800	263,500	688,500	1,029,800
1980	83,500	266,500	701,800	1,051,800
1981	86,200	269,500	714,100	1,069,800
1982	90,500	271,700	721,200	1,083,400
1983	91,884	273,506	729,728	1,095,118
1984	93,780	271,811	736,654	1,102,245
Avg.	72,913	254,256	637,299	964,468
Std.	10,275	17,706	69,610	95,386
Max.	93,780	273,506	736,654	1,102,245
Min.	61,700	211,300	494,500	767,500

Table 6.A.4.3

U.S. Pipeline Industry Products Transported [Ref. 6.A.4.1]

Year	Crude Oil Ton-Miles (1 E9)	Refined Petroleum Products Ton-Miles (1 E9)	Total Petroleum Products Ton-Miles (1 E9)	Domestic Natural Gas Cubic Ft. (1 E9)
1974	303.0	203.0	506.0	22,111
1975	288.0	219.0	507.0	20,410
1976	303.0	212.0	515.0	20,801
1977	326.6	219.4	546.0	19,521
1978	359.5	226.3	585.8	19,627
1979	372.2	236.1	608.3	20,241
1980	362.6	225.6	588.2	19,877
1981	333.1	230.6	563.7	19,404
1982	335.1	230.6	565.7	18,001
1983	332.4	223.7	556.1	16,385
1984	333.0	235.1	568.1	17,951
Avg.	331.7	223.8	555.4	19,484
Std.	25.2	9.5	32.5	1,489
Max.	372.2	236.1	608.3	22,111
Min.	288.0	203.0	506.0	16,385

Pipeline Accident References

[6.A.4.1] DOT-TSC-RSPA-86-3 National Transportation Statistics Annual Report, 1986, Kathleen Bradley, U.S. Department of Transportation, Research and Special Programs Administration, Transportation Systems Center, Cambridge, MA. July 1986.

6.A.5 Railroad Accident Data

All railroads in the United States are required by Federal regulations (49 CFR 225) to file monthly accident/incident reports with the Office of Safety, Federal Railroad Administration (FRA) of the U.S. Department of Transportation (U.S. DOT). The regulations define a railroad as a system of surface transportation of persons and/or property over rails. It include line-haul freight and passenger railroads, switching and terminal railroads, and passenger-carrying railroads including, but not limited to, rapid transit, commuter, scenic, street, subway, elevated, cable, and cog railways [Refs. 6.A.5.1 - 6.A.5.9].

A train accident is defined by the FRA Office of Safety as any event involving on-track railroad equipment that results in damage to railroad on-track equipment, signals, track or track structures, and roadbed at or exceeding the dollar damage threshold limit. Prior to 1975, the dollar damage threshold limit was $750. In 1975, it was adjusted upwards to $1750 to account for the effects of inflation. This limit was adjusted upwards every two years thereafter to reflect inflationary changes. In 1977, the limit was $2300. In 1979, the limit was $2900. In 1981, the limit was changed to $3700. In 1983, the limit was $4500. This change improved the comparability of the accident data, but comparisons were valid only for those years when the threshold was adjusted. Generally, the number of accidents reported in the year following an adjustment was higher. Biennial threshold adjustment made year-to-year statistical comparisons more difficult and imposed an unnecessary reporting burden on the railroads. Therefore, the FRA began to adjust the second year accident data by limiting the count of accidents in 1978 to those resulting in reportable damge of more than $2600, in 1980 to those over $3200, in 1982 to those over $4100, and in 1984 to those over $4700. For all other years, the actual number of accidents was reported [Refs. 6.A.5.1 - 6.A.5.9]. The net result has been that the dollar damage threshold limit has been steadily increasing at an almost linear rate.

Table 6.A.5.1 presents the railroad accident rate for 1965 through 1984 [Refs. 6.A.5.1 - 6.A.5.9]. The average railroad accident rate for this period is 10.1 per million train miles (10.1 E-6/train mile or 1.01 E-5/train mile) with a standard deviation of 2.49 E-6/train mile. Train miles are defined as the sum of the locomotive miles, yard switching miles and motor train miles as tabulated by the FRA for each year. The FRA defines locomotive mile as the movement under its own power of a locomotive a distance one mile whether coupled with or without cars. This item covers miles run by locomotives in road service and in train and yard-switching service. Switching miles are computed at the rate of six miles per hour for the time actually engaged in such service. A motor train mile is a movement under its own power of a motor train a distance of one mile [Refs. 6.A.5.1 - 6.A.5.9].

Table 6.A.5.2 presents railroad accident data on trains transporting hazardous materials (commonly referred to as hazmat.) by accident type [Refs. 6.A.5.1 - 6.A.5.9]. Note that this accident data gives the number of

trains involved in each accident type rather than the actual number of accidents.

Table 6.A.5.3 presents railroad accident data on trains transporting hazardous materials and their subsequent fate [Refs. 6.A.5.1 - 6.A.5.9].

Table 6.A.5.1

U.S. Railroad Accident Rate 1965-1984
[Refs. 6.A.5.1 - 6.A.5.9]

Year	Train Miles (000)	Accidents	Accidents/ Train Mile	Acc. Damage Threshold	Accident Fatalities	Accident Injuries
1965	930,437	5,967	6.41E-06	$750.00	191	864
1966	941,416	6,793	7.22E-06	$750.00	214	900
1967	895,028	7,294	8.15E-06	$750.00	170	754
1968	876,489	8,028	9.16E-06	$750.00	142	1,293
1969	864,081	8,543	9.89E-06	$750.00	203	1,173
1970	838,674	8,095	9.65E-06	$750.00	210	627
1971	783,844	7,304	9.32E-06	$750.00	171	694
1972	781,408	7,532	9.64E-06	$750.00	171	777
1973	831,347	9,698	1.17E-05	$750.00	149	758
1974	833,261	10,694	1.28E-05	$750.00	139	911
1975	755,033	8,041	1.06E-05	$1,750.00	82	1,220
1976	774,764	10,248	1.32E-05	$1,750.00	152	1,279
1977	750,042	10,362	1.38E-05	$2,300.00	108	985
1978	751,964	11,277	1.50E-05	$2,600.00	139	1,911
1979	763,429	9,740	1.28E-05	$2,900.00	100	1,275
1980	717,662	8,451	1.18E-05	$3,200.00	97	860
1981	676,216	5,781	8.55E-06	$3,700.00	63	562
1982	573,369	4,589	8.00E-06	$4,100.00	49	472
1983	558,191	3,906	7.00E-06	$4,500.00	56	502
1984	592,599	3,900	6.58E-06	$4,700.00	63	893
Total	15,489,254	156,243	2.013E-04		2,669	18,710
Avg.	774,463	7,812	1.006E-05		133.45	935.50
Std.	107,447	2,136	2.491E-06		52.04	335.58

Average Accident Rate Based on Totals: 1.009E-05

1975-1984 Total Accidents:	76,295
1975-1984 Average No. of Accidents:	7,630
1975-1984 Std. Deviation:	2,709

Table 6.A.5.2

Accidents Involving Trains Transporting Hazardous Material (Hazmat.)
1975-84, by Type of Accident [Refs. 6.A.5.1 - 6.A.5.9]

Year	No. Trains w/ Hazmat. Collisions	No. Trains w/ Hazmat. Derailmnts	No. Trains w/ Hazmat. Rail-Hghwy Grade-Xing	No. Trains w/ Hazmat. Explosion, Detonation, Fire or Violent Rupture	No. Trains w/ Hazmat. Other Accidents	No. Trains w/ Hazmat. All Accidents
1975	82	637	16	0	1	736
1976	122	627	18	0	1	768
1977	122	673	22	24	23	864
1978	154	815	21	13	32	1,035
1979	141	765	20	5	26	957
1980	118	673	22	7	22	842
1981	81	479	15	4	22	601
1982	58	394	15	1	36	504
1983	49	349	11	1	21	431
1984	46	347	20	2	28	443
Total	973	5,759	180	57	212	7,181
Avg.	97.30	575.90	18.00	5.70	21.20	718.10
Std.	37.12	162.63	3.46	7.18	11.07	203.62
Pct.	13.55%	80.20%	2.51%	0.79%	2.95%	100.00%

Table 6.A.5.3

Accidents Involving Trains Transporting Hazardous Material (Hazmat.)
1975-84, by Cars Involved, Damaged, Releasing Hazmat.
[Ref. 6.A.5.1 - 6.A.5.9]

Year	No. Trains Involved	Cars in Train	Cars w/ Hazmat.	Avg. Pct. Cars w/ Hazmat. in Train	Cars w/ Hazmat. Damaged	Cars Releasing Hazmat.
1975	736	54,079	5,668	10.48%	976	135
1976	768	52,702	3,347	6.35%	847	166
1977	864	60,838	3,848	6.32%	1,072	173
1978	1,035	73,324	4,944	6.74%	1,229	232
1979	957	67,085	4,693	7.00%	1,060	165
1980	842	59,697	4,139	6.93%	989	173
1981	601	41,197	2,770	6.72%	773	109
1982	504	35,268	2,297	6.51%	671	137
1983	431	28,872	2,456	8.51%	543	62
1984	443	30,384	2,826	9.30%	581	100
Total	7,181	503,446	36,988		8,741	1,452
Avg.	718.10	50344.60	3698.80	7.49%	874.10	145.20
Std.	203.62	14818.42	1090.44	1.36%	217.03	45.11

Avg. No. Cars per Train = 70.11
Avg. No. Cars w/ Hazmat. per Train = 5.15
Avg. No. Cars w/ Hazmat. Damaged per Train = 1.22

Railroad Accident Data References

[6.A.5.1] Accident/Incident Bulletin No. 145, Calender Year 1976, Office of Safety, Federal Railroad Administration, U.S. Department of Transportation, Washington, D.C. December 1977.

[6.A.5.2] Accident/Incident Bulletin No. 146, Calender Year 1977, Office of Safety, Federal Railroad Administration, U.S. Department of Transportation, Washington, D.C. August 1978.

[6.A.5.3] Accident/Incident Bulletin No. 147, Calender Year 1978, Office of Safety, Federal Railroad Administration, U.S. Department of Transportation, Washington, D.C. October 1979.

[6.A.5.4] Accident/Incident Bulletin No. 148, Calender Year 1979, Office of Safety, Federal Railroad Administration, U.S. Department of Transportation, Washington, D.C. July 1980.

[6.A.5.5] Accident/Incident Bulletin No. 149, Calender Year 1980, Office of Safety, Federal Railroad Administration, U.S. Department of Transportation, Washington, D.C. June 1981.

[6.A.5.6] Accident/Incident Bulletin No. 150, Calender Year 1981, Office of Safety, Federal Railroad Administration, U.S. Department of Transportation, Washington, D.C. June 1982.

[6.A.5.7] Accident/Incident Bulletin No. 151, Calender Year 1982, Office of Safety, Federal Railroad Administration, U.S. Department of Transportation, Washington, D.C. June 1983.

[6.A.5.8] Accident/Incident Bulletin No. 152, Calender Year 1983, Office of Safety, Federal Railroad Administration, U.S. Department of Transportation, Washington, D.C. June 1984.

[6.A.5.9] Accident/Incident Bulletin No. 153, Calender Year 1984, Office of Safety, Federal Railroad Administration, U.S. Department of Transportation, Washington, D.C. June 1985.

6.A.6 Truck Accident Data

This section includes information furnished on accident report forms submitted to the U.S. Department of Transportation (U.S. DOT) Federal Highway Administration (FHWA) Bureau of Motor Carrier Safety (BMCS) by Motor Carriers of Property (vs. passengers) who operate in interstate or foreign commerce [Ref. 6.A.6.1 - 6.A.6.14].

The BMCS defines Motor Carriers of Property as falling broadly into two categories: (1) Private and (2) For-hire. Private carrers are defined as manufacturers, wholesalers, retailers, and others who use their own (or leased) vehicles as part of their commercial operation to transport their own goods. The For-hire carriers are defined as trucking firms which haul freight owned by another party [Ref. 6.A.6.1 - 6.A.6.14].

For-hire carriers are further divided by the BMCS into three categories: (1) Common, (2) Contract, and (3) Exempt. Common carriers provide transportation services to the public in general. Contract carriers are in business to meet the needs of individual customers. Exempt carriers conduct stipulated types of operations (e.g., hauling agricultural commodities from farm to processor) and are not subject to Interstate Commerce Commission (ICC) economic regulation. Interstate common and contract carriers are subject to the economic regulations of the ICC. In order to operate interstate, common or contract carriers must have its specific services certified (or authorized) by the ICC. These certified for-hire carriers are also known as regulated or authorized carriers [Ref. 6.A.6.1 - 6.A.6.14].

Truck accidents are defined by the BMCS as occurrences involving a motor vehicle operated by a motor carrier subject to the Federal Motor Carrier Safety Regulations (FMCSR) (49 CFR 390-397) resulting in (1) the death of one or more human beings; (2) bodily injury to one or more persons who, as result, receives medical treatment away from the scene of the accident; and/or (3) total damage to all property aggregating dollar damage at or above the dollar damage threshold limit based on actual cost of reliable estimates [Ref. 6.A.6.1 - 6.A.6.14].

Prior to 1973, only truck accidents involving for-hire carriers with accident dollar damage of $250 or greater was tabulated by the BMCS [Ref. 6.A.6.1 - 6.A.6.3]. Table 6.A.6.1 presents the intercity accident summary for U.S. for-hire motor carriers from 1960 through 1972 with the $250 damage threshold. Total vehicle miles were included in the BMCS reports so an accident rate per vehicle mile could be calculated year by year. The average accident rate for this period was 2.50 accidents per million truck miles (2.50×10^{-6} accidents/vehicle mile) with a standard deviation of 2.76×10^{-7} and with a maximum accident rate of 3.17×10^{-6} recorded in 1964. When the totals of this period are used to calculate an average accident rate, a value of 2.48

x 10^{-6} accident/vehicle mile is obtained [Ref. 6.A.6.1 - 6.A.6.3].

In 1973, the BMCS began tabulating truck accidents of private as well as for-hire carriers. This change resulted in a greatly expanded accident data base. In addition, the dollar damage threshold limit was increased to $2000 to account for the effects of inflation [Ref. 6.A.6.4 - 6.A.6.14]. The dollar damage threshold limit has not been changed since then. Because of these changes in truck accident tabulating rules, BMCS accident data before 1973 is not comparable to accident data after 1973. Another shortcoming of the BMCS accident data is that a precise definition of a truck is not given. This is important because inclusion of vehicles such as pickups, light utility vans, in short, any truck that is not a large tractor/semitrailer truck will tend to obscure the actual accident rate of interest. Table 6.A.6.2 presents the accident data summary for U.S. for-hire and private motor carriers from 1973 through 1984 with the $2000 damage threshold. Total vehicle miles were not included in the BMCS reports so an accident rate per vehicle mile could not be calculated [Ref. 6.A.6.4 - 6.A.6.14].

The American Petroleum Institute maintains an accident data base which consists of information submitted by petroleum industry companies to the American Petroleum Institute. The number of companies submitting information changes from year to year. As a result, this accident data base does not remain constant. Subsidiaries of the reporting petroleum industry companies are usually but not always included. Also, the precise accident definition has not been given in these reports. Nevertheless, this accident data is included in this appendix because it gives a truck accident rate for an industry dominated by private carriers [Ref. 6.A.6.15 - 6.A.6.19].

Table 6.A.6.3 presents the accident data summary for the U.S. petroleum industry from 1968 through 1981 with trucks reported separately [Ref. 6.A.6.15 - 6.A.6.19]. Vehicle miles for these trucks were recorded separately so an accident rate per truck mile could be calculated year by year. The average accident rate for this period for trucks in the petroleum industry was 6.41 accidents per million truck miles (6.41 x 10^{-6} accidents/truck mile) with a standard deviation of 9.50 x 10^{-7} and a maximum accident rate of 8.35 x 10^{-6} recorded in 1968. When the totals of this period are used to calculate an average accident rate, a value of 6.45 x 10^{-6} is obtained. It should be noted that this accident data covers the period of 1968 through 1981 which includes the imposition of the 55 miles/hour national speed limit. The vehicle accident rate and truck accident rate for private carriers of the petroleum industry continued its downward trend following the imposition of the national speed limit and has remained lower except for occasional fluctuations.

Table 6.A.6.1

U.S. Motor Carriers Intercity Accident Summary 1960-72
For-Hire Carriers, $250 Accident Damage Threshold
[References 6.A.6.1 - 6.A.6.3]

Year	Fatal Acc.	Injury Accs.	Prop. Damage Accs.	Total Accs.	Fatal.	Inj.	Truck Miles (000,000)	Accident Rate (/trk.mile)
1960	NA	NA	NA	22,616	1,353	14,336	9,087	2.49E-06
1961	NA	NA	NA	19,922	1,130	13,024	8,839	2.25E-06
1962	NA	NA	NA	22,566	1,437	15,296	9,545	2.36E-06
1963	NA	NA	NA	28,628	1,451	18,517	9,884	2.90E-06
1964	NA	NA	NA	30,267	1,492	19,297	9,536	3.17E-06
1965	NA	NA	NA	24,794	1,471	15,158	10,862	2.28E-06
1966	NA	NA	NA	26,606	1,361	15,478	10,956	2.43E-06
1967	NA	NA	NA	25,981	1,291	14,882	10,705	2.43E-06
1968	NA	NA	NA	29,209	1,421	16,124	11,704	2.50E-06
1969	1,108	10,414	19,150	30,672	1,361	16,232	12,461	2.46E-06
1970	1,128	10,468	21,607	33,203	1,375	15,793	12,390	2.68E-06
1971	944	9,208	20,429	30,581	1,163	13,988	13,951	2.19E-06
1972	1,065	10,383	25,234	36,682	1,294	15,822	15,883	2.31E-06
Tot.	NA	NA	NA	361,727	17,600	203,947	145,803	
Avg.	NA	NA	NA	27,825	1,354	15,688	11,216	
Std.	NA	NA	NA	4,497	107	1,629	1,970	

Average Accident Rate Based on Totals: 2.48E-06

NA = information not available at time of table preparation.

Table 6.A.6.2

U.S. Motor Carriers Accident Data Summary 1973-84
For-Hire and Private Carriers, $2000 Accident Damage Threshold
[References 6.A.6.4 - 6.A.6.14]

Year	Col. Accs.	Noncol. Accs.	Total Accs.	Fatal.	Inj.
1973	23,805	7,106	30,911	3,058	35,245
1974	18,682	6,676	25,358	2,429	26,911
1975	17,845	6,429	24,274	2,232	26,374
1976	19,074	6,592	25,666	2,520	26,794
1977	22,173	7,763	29,936	2,983	31,698
1978	24,982	9,016	33,998	2,998	32,757
1979	25,946	9,595	35,541	3,072	32,126
1980	23,056	8,333	31,389	2,528	27,149
1981	24,095	8,211	32,306	2,810	28,533
1982	23,511	8,248	31,759	2,479	25,779
1983	23,480	8,148	31,628	2,528	26,692
1984	28,118	8,736	36,854	2,721	29,149
Tot.	274,767	94,853	369,620	32,358	349,207
Avg.	22,897	7,904	30,802	2,697	29,101
Std.	2,920	971	3,796	271	2,966

Table 6.A.6.3

U.S. Petroleum Industry Accident Data Summary,
Trucks Reported Separately 1968-1981
[References 6.A.6.15 - 6.A.6.19]

Year	No. of Companies	No. of Vehicles	Total Accs.	Vehicle Miles (000)	Accident Rate /(veh.mile)
1968	93	22,697	4,421	529,687	8.35E-06
1969	80	29,821	5,330	754,496	7.06E-06
1970	76	29,442	5,567	757,348	7.35E-06
1971	75	28,465	5,064	774,570	6.54E-06
1972	72	31,039	5,500	827,551	6.65E-06
1973	73	20,046	3,804	508,783	7.48E-06
1974	73	20,147	3,151	469,804	6.71E-06
1975	69	29,071	4,089	779,260	5.25E-06
1976	70	22,748	3,528	585,609	6.02E-06
1977	69	21,508	2,784	519,446	5.36E-06
1978	68	19,113	2,562	404,748	6.33E-06
1979	63	21,414	2,889	467,939	6.17E-06
1980	62	21,970	2,391	455,324	5.25E-06
1981	81	21,158	2,445	465,571	5.25E-06
Tot.	1,024	338,639	53,525	8,300,136	
Avg.	73.14	24,189	3,823	592,867	6.41E-06
Std.	7.62	4,148	1,138	144,860	9.15E-07

Average Accident Rate Based on Totals: 6.45E-06

Table 6.A.6.4

U.S. Truck Fleet Population [Ref. 6.A.6.20]

Year	Combination Truck (000)	Single Unit Truck (000)	Total Trucks (000)	Intercity Freight Revenue Ton-Miles (000,000)
1974	1085.0	23545.2	24630.2	495,000
1975	1131.0	23644.7	24775.7	454,000
1976	1224.8	26554.1	27778.9	510,000
1977	1264.1	28298.4	29562.5	555,000
1978	1366.6	30366.0	31732.6	599,000
1979	1339.0	32010.7	33349.7	608,000
1980	1405.0	32232.2	33637.2	555,000
1981	1255.5	33195.6	34451.1	527,000
1982	1232.2	33911.7	35143.9	520,000
1983	1273.4	35815.1	37088.5	575,000
1984	1259.5	36787.6	38047.1	605,000
Avg.	1257.8	30578.3	31836.1	545,727
Std.	89.5	4345.4	4399.3	47,227
Max.	1405.0	36787.6	38047.1	608,000
Min.	1085.0	23545.2	24630.2	454,000

Truck Accident Data References

[6.A.6.1] 1969 Accidents of Large Motor Carriers of Property, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. December 1970.

[6.A.6.2] 1970 Accidents of Large Motor Carriers of Property, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. March 1972.

[6.A.6.3] 1971-1972 Accidents of Large Motor Carriers of Property, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. May 1974.

[6.A.6.4] 1973 Accidents of Motor Carriers of Property, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. July 1975.

[6.A.6.5] 1974 Accidents of Motor Carriers of Property, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. (No Date).

[6.A.6.6] 1975 Accidents of Motor Carriers of Property, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. (No Date).

[6.A.6.7] 1976 Accidents of Motor Carriers of Property, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. October 26, 1977.

[6.A.6.8] 1977 Accidents of Motor Carriers of Property, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. (No Date).

[6.A.6.9] Accidents of Motor Carriers of Property 1978, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. May 1980.

[6.A.6.10] Accidents of Motor Carriers of Property 1979, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. (No Date).

[6.A.6.11] Accidents of Motor Carriers of Property 1980-1981, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. August 27, 1982.

[6.A.6.12] Accidents of Motor Carriers of Property 1982, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. May 1983.

[6.A.6.13] Accidents of Motor Carriers of Property 1983, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. October 1984.

[6.A.6.14] Accidents of Motor Carriers of Property 1984, Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, Washington, D.C. May 1986.

[6.A.6.15] Summary of Motor Vehicle Accidents in the Petroleum Industry for 1977, American Petroleum Institute, Washington, D.C. May 1978.

[6.A.6.16] Summary of Motor Vehicle Accidents in the Petroleum Industry for 1978, American Petroleum Institute, Washington, D.C. August 1979.

[6.A.6.17] Summary of Motor Vehicle Accidents in the Petroleum Industry for 1979, American Petroleum Institute, Washington, D.C. June 1980.

[6.A.6.18] Summary of Motor Vehicle Accidents in the Petroleum Industry for 1980, American Petroleum Institute, Washington, D.C. September 1981.

[6.A.6.19] Summary of Motor Vehicle Accidents in the Petroleum Industry for 1981, American Petroleum Institute, Washington, D.C. August 1982.

[6.A.6.20] DOT-TSC-RSPA-86-3 National Transportation Statistics Annual Report, 1986, Kathleen Bradley, U.S. Department of Transportation, Research and Special Programs Administration Systems Center, Cambridge, MA. July 1986.

CHAPTER 7 -- SUMMARY AND CONCLUSIONS

7.1 COMPARISON WITH FIGURE-OF-MERIT #1

Of the four external initiating events examined, internal fires, high wind/tornadoes, external floods, and transportation accidents, all were found to be important with respect to the first figure-of-merit of 1×10^{-5} per reactor year for core damage and should be included in any vulnerability search.

For internal fires, the importance is generic for all plants, that is, all plants will require a vulnerability study of internal fires.

For high winds/tornadoes, many plants will be able to screen out this issue based on either the absence of wind-vulnerable structures or the determination that high wind/tornado frequencies are sufficiently low. Some plants may find it necessary to perform a more detailed plant response analysis.

For external floods, many plants will be able to screen out this issue based on site characteristics, if they are located high enough or are sufficiently protected. Some sites may require more detailed plant response analysis, if the initiating event frequency cannot be shown to be low enough.

Finally, for transportation accidents, most sites will be able to screen out this issue based on frequency and distance considerations. Only a very few sites may require further analysis of one or more transportation modes, possibly including some site response analysis.

7.2 COMPARISON WITH FIGURE-OF-MERIT #2

Of the four external initiators studied, only internal fires had enough literature available to provide insight into how plants might perform with respect to the second figure-of-merit of 1×10^{-6} per reactor year for a large radioactive release. The five internal fire PRAs revealed that there were no fire initiated sequences leading to large releases with frequencies above about 10^{-9} per year. Whether this finding is generally applicable to all reactors could not be determined.

7.3 RISK ANALYSIS TECHNIQUES

It was found possible to separate the probabilistic risk analysis into two parts for all of the external initiating events examined except for internal fires. The first part is the probability of the initiating event itself. The second part is the response of the plant to the external initiating event. For internal fires, the probability of the initiating event was found to depend on the plant design which also affects the plant response.

For three of the external initiators examined, internal fires, high wind/tornadoes, and transportation accidents, the techniques and data are generally available to allow a relatively good determination of the frequency of the initiating events that might compromise the safety of nuclear power plants. Site specific data may not be available and thus generic information may be all that is obtainable for most plants. For external floods, methods for determining the frequency of very large initiating events at vulnerable sites are only fair to poor, depending on the site and its characteristics.

For those initiating events that had full-scope PRAs performed for them, internal fires, high wind/tornadoes, and external floods, enough analysis of the plant response has been done to demonstrate that the PRA approach will work and is acceptable. For transportation accidents, for which no full-scope PRAs are available, the probabilistic analysis methodology of plant response is less than adequate and much more work remains to be done in this area should it be necessary to perform plant response analysis.

7.4 SUGGESTED APPROACHES TO VULNERABILITY ANALYSIS

A full-scope PRA type analysis is always an acceptable approach to studying vulnerabilities due to any of the four external initiator categories covered in this report.

An abbreviated form of a full-scope PRA, such as a screening type PRA vulnerability search, may be useful in providing the information needed to determine a plant's response to external initiating events especially when plant systems models and a support system dependency matrix have been developed for other purposes (such as for an internal initiator PRA study).

For internal fires, no simplified approach currently exists, although advanced methods of screening vulnerable areas now allow a full-scope PRA to be accomplished relatively efficiently.

For high wind/tornadoes, the screening out of all but a few plant sites with wind vulnerable structures may allow a relatively straightforward PRA vulnerability analysis.

For external floods, identification of those few plant sites vulnerable to flooding may allow PRA vulnerability analysis of this issue for most plants without large additional effort.

For transportation accidents, frequency analysis is usually adequate for screening purposes if allowance is made for future trends in transportation. For those few plant sites for which plant response is needed, the analysis may be difficult to do.

An approximate probabilistic plant response analysis may be sufficient to screen out a category of an external initiator if the frequency is small enough. It should be feasible to perform such an analysis for the four external initiators considered in this report.

7.5 OTHER EXTERNAL INITIATORS

In Chapter Two, a list of external initiators other than those explicitly considered in this study (the "others" category) was presented. The list ranges from hailstorms to meteorite strikes to nearby industrial accidents. No study was done of these initiators, but broad conclusion about them can nevertheless be stated here, to round out the discussion of external initiators.

Other external initiators can be broadly divided into three subcategories: 1) those external initiators which are applicable to all U.S. reactor sites and cannot be easily dismissed on the basis of their initiating event frequency, 2) those external initiators which are applicable to all U.S. reactors sites and can be easily dismissed on the basis of their initiating event frequency, and 3) those external initiators which are applicable to only a few U.S. reactor sites and may not be easily dismissed on their initiating event frequency without further study. Examples of the first subcategory would be lightning strikes, industrial sabotage, accidents from nearby industrial/military facilities, damage or destruction due to military action, and onsite hazardous material release. Examples of the second subcategory would be extraterrestrial activity such as meteorite strikes. All of the other external initiators would be considered to be in the third subcategory unless additional information allows them to be placed in one of the other two subcategories.

If any specific reactor site finds itself potentially vulnerable to one or more of these other external initiators, then it will be necessary to perform some sort of analysis. The initial analysis may be of the scoping type, e.g., a scoping analysis to ascertain using conservative arguments whether or not plant safety might be significantly compromised. A more detailed analysis would be needed in those cases where a scoping analysis could not convincingly dismiss an external initiator. Whatever type of analysis is performed, it will be necessary for it to be probabilistic in nature. Specifically, arguments must be based on showing that the frequency of a postulated initiating event is very small, or that the contingent likelihood of adverse plant response is very small. Deterministic type arguments without probabilistic numbers (even crude numbers) will not be sufficient when compared with probabilistic criteria.

Finally, no "check list" should be needed for these potential external initiators. An individual nuclear power plant that may be potentially vulnerable to one or more of these external initiators should be responsible for identifying these initiators as part of its broader vulnerability review.

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG/CR-5042
UCID-21223

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Evaluation of External Hazards to Nuclear Power Plants in the United States

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
C. Y. Kimura, and R. J. Budnitz* *Future Resources Assoc., Inc.

5. DATE REPORT COMPLETED

MONTH	YEAR
October	1987

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Lawrence Livermore National Laboratory
Post Office Box 808, L-196
Livermore, California 94550

DATE REPORT ISSUED

MONTH	YEAR
December	1987

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Reactor and Plant Systems
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.
A0448/A0815

13. TYPE OF REPORT
Technical

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less) As part of the research program supporting the implementation of the NRC Policy Statement on Severe Accidents, the Lawrence Livermore National Laboratory (LLNL) has performed a study of the risk of core damage to nuclear power plants in the United States due to externally initiated events. The broad objective has been to gain an understanding of whether or not each external initiator is among the major potential accident initiators that may pose a threat of severe reactor core damage or of large radioactive release to the environment from the reactor.

Four external hazards were investigated in this report. These external hazards are internal fires, high winds/tornadoes, external floods, and transportation accidents. Analysis was based on two figures-of-merit, one based on core damage frequency and the other based on the frequency of large radioactive releases.

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

17. KEY WORDS AND DOCUMENT ANALYSIS
Reactor core damage risk
Environmental radioactive release risk
External hazards to nuclear power plants
fire hazards

17a. DESCRIPTORS
transportation hazards
floods
tornadoes

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
Unclassified

20. SECURITY CLASS (This page)
Unclassified

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (7-77)

☆U.S. GOVERNMENT PRINTING OFFICE:1987-202-292:60308