NUREG/CR-5580
SAND90-1507
Vol. 1

Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment

Main Report

Prepared by
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Sandia National Laboratories
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U.S. Nuclear Regulatory Commission

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Manuscript Completed: September 1992
Date Published: December 1992

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DOCUMENTS IN SERIES

This report is one of a series of reports documenting the technical findings associated with the resolution of Generic Issue 57: Effects of Fire Protection Systems on Safety-Related Equipment.

There are several reports published in association with the resolution of Generic Issue 57. These are:

NUREG/CR-5580, SAND90-1507, Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment, Main Report, December 1992.

NUREG/CR-5789, SAND91-1534, Risk Evaluation for a Westinghouse PWR, Effects of Fire Protection System Actuation on Safety-Related Equipment: Evaluation of Generic Issue 57, December 1992.

NUREG/CR-5791, SAND91-1536, Risk Evaluation for a General Electric BWR, Effects of Fire Protection System Actuation on Safety-Related Equipment: Evaluation of Generic Issue 57, December 1992.

NUREG/CR-5790, SAND91-1535, Risk Evaluation for a Babcock & Wilcox Pressurized Water Reactor, Effects of Fire Protection System Actuation on Safety-Related Equipment (Evaluation of Generic Issue 57), September 1992.

NUREG/CR-5906, SAND92-1547, Decision Making Under Uncertainty: An Investigation Into the Application of Formal Decision-Making Methods to Safety Issue Decisions, December 1992.

Letter Report, EGG-NTA-9081, Risk Evaluation of a Westinghouse 4-Loop PWR, Effects of Fire Protection System Actuation on Safety-Related Equipment (Evaluation of Generic Issue 57), Idaho National Engineering Laboratory, December 1991.

Letter Report, Seismic Risk Evaluation for a Pressurized Water Reactor, Effects of Fire Protection System Actuation on Safety-Related Equipment, Sandia National Laboratories, December 1991.

ABSTRACT

Nuclear power plants have experienced actuations of fire protection systems (FPSs) under conditions for which these systems were not intended to actuate and also have experienced advertent actuations with the presence of a fire. These actuations have often damaged safety-related equipment.

A review of the impact of past occurrences of both types of such events and their impact on plant safety systems, an analysis of the risk impacts of such events on nuclear power plant safety, and a cost-benefit analysis of potential corrective measures have been performed. Thirteen different scenarios leading to actuation of fire protection systems due to a variety of causes were identified. These scenarios ranged from inadvertent actuation caused by human error to hardware failure, and include seismic root causes and seismic/fire interactions. A quantification of these thirteen root causes, where applicable, was performed on generically applicable scenarios.

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ACKNOWLEDGMENTS

The authors would like to thank Mr. Demetrios Basdekas, the NRC Project Manager for GI-57, for sponsoring this work.

The authors wish to acknowledge the efforts of Sharon Daniel and Eric Klamerus of Sandia National Laboratories. Sharon Daniel performed the uncertainty analysis for the generically applicable scenarios analyzed. Eric Klamerus performed the generic plant seismic analysis.

The authors also wish to acknowledge the efforts of Steven Harris, Laurie Friedman, Omar Khemici, and James Johnson of EQE Engineering who were responsible for the analysis of fire protection system performance during the Loma Prieta earthquake. The results of their work are given in Appendix C.

The authors also wish to acknowledge the efforts of Harvey Goranson of Professional Loss Control, Inc. who was responsible for the survey of fire protection strategies at U.S. commercial nuclear power plants. The results of his work are given in Appendix D.

Additionally, the authors wish to acknowledge the efforts of Karen Jones and Dena Wood of Science & Engineering Associates, Inc. for their word processing and graphics support.

EXECUTIVE SUMMARY

Analysis of USNRC Generic Issue 57 involved development of a detailed understanding of the potential safety significance of U.S. commercial nuclear power plant fire protection system (FPS) advertent and inadvertent actuations. In this report, an extensive review of operational experiences involving such FPS actuations is presented. A methodology for the quantification of effects of fire protection system actuation on safety-related equipment has been developed. This methodology has been applied to specific plants: one boiling water reactor (BWR) and two pressurized water reactors (PWRs). In addition, analysis of a third PWR was independently conducted by the Idaho National Engineering Laboratory. For this third PWR, Sandia National Laboratories conducted an independent evaluation of the risk associated with the seismic root causes. In applying the methodology, extensive plant walkdowns were conducted in addition to detailed reviews of plant documentation. Building on the insights gained as a result of the analysis of these four plants, a risk assessment was made for a generic light water plant. For the generic plant, both core damage frequency and incremental risk are calculated. An uncertainty analysis is performed on both core damage frequency and risk. A cost/benefit assessment was then performed for candidate modifications for several plant as-found conditions that were demonstrated to be contributors to risk. Technical insights from all of these analyses are presented for the use of those involved with the design of the advanced light water reactor (ALWR). Finally, application of analytic techniques for decision making under uncertainty analysis was performed using the specific plant models and results. This decision-making analysis is documented in a separate report.

A review of fire protection system actuation events that occurred at U.S. commercial nuclear power plants during the time period of January 1, 1980 to December 31, 1989 is presented. Included in this section is a discussion of how the data review was conducted and a summary of both the inadvertent and advertent FPS actuations identified in the study. Data from two additional sources is reviewed. This additional data included information concerning FPS actuations at foreign nuclear power plants and actuations at U.S. naval shore facilities. A review is also conducted of FPS performance data collected after the Loma Prieta earthquake of 1989.

A methodology is developed for the evaluation of potential accident scenarios caused by FPS actuations. Thirteen root causes (both seismic and non-seismic) leading to increases in core damage frequency and risk have been identified. The methodology uses a plant internal event probabilistic risk assessment as a basis, with the addition of a vital area analysis for safety-related components and cables. While helpful, the existence of a fire PRA is not required for the application of the methodology. The methodology can be readily applied to a specific plant, as has been done to three PWRs and one BWR, the results of which are separately reported. In this report, a generic plant model is developed and analyzed using the methodology.

A generic plant analysis is conducted, based on insights gained from the individual plant analytic work as well as the survey of fire protection strategies at U.S. commercial nuclear power plants. For the generic plant, core damage frequency is calculated for a generic cable spreading room, diesel generator rooms, and emergency electrical switchgear rooms. For each space, fire protection systems in use in such spaces in U.S. commercial nuclear power plants were assessed. The FPSs analyzed were wetpipe water, preaction water, deluge water, Halon, and CO_2 systems. After core damage frequency is calculated, generic containment systems are modeled for a PWR and a BWR. Using these models, generic risk is assessed. For core damage frequency, both an uncertainty analysis and sensitivity studies were conducted. For generic plant risk, an uncertainty analysis is performed. In addition, core damage frequency and risk is assessed for "most vulnerable/typical/least vulnerable case".

The following core damage frequencies and 20-year risks associated with effects of fire protection system actuation on safety-related equipment were found for "typical" pressurized and boiling water reactors (PWRs and BWRs), to be:

	CDF	20 Year Risk
PWR	3.3E-5/ry	51 Person-REM
BWR	3.3E-5/ry	210 Person-REM

During the course of the analysis of individual plants, several design issues were identified as potential contributors to risk. These were identified for individual plants during documentation reviews, plant walkdowns, and application of the risk assessment methodology. Additional issues were identified while performing the analysis of the generic plant. For eleven of these issues, costing of risk-reduction modifications is computed, and then cost-benefit ratios (dollars/person-REM averted) for the potential modifications were calculated.

Technical insights are discussed for use in the Advanced Light Water Reactor (ALWR) program. These are provided as insights only, not as specific recommendations or design requirements. Presented for the use of the ALWR designer are summaries of findings of existing design features that result in contribution of risk due to the effects of fire protection system actuation on safety-related equipment. Additionally presented for consideration is the concept of applying the FPS risk assessment methodology to the ALWR in the design phase, in an effort to optimize the plant design against risk from the effects of FPS actuation on safety-related equipment.

1.0 INTRODUCTION

Experience in recent years has shown that fire protection systems in nuclear power plants have actuated at times and under conditions for which they were not intended to actuate as well as when intended in the presence of a fire. These actuations have often affected and even caused damage to plant equipment. On some occasions, the damage has been to safety-related equipment, that is, equipment required to ensure the capability to safely shutdown the plant. On other occasions, the damage has been to equipment required for the normal operation of the plant and the reactor was subsequently shutdown. As a consequence, the actuation of fire protection systems represents a potentially important safety issue requiring further study.

In the recently completed Fire Risk Scoping Study (Ref. 1.1), the inadvertent actuation of fire protection systems in commercial United States nuclear power plants was briefly reviewed. Seventy-one events resulting in submission of a Licensee Event Report (LER) were identified during the period from April 1, 1980 to July 14, 1987. The average frequency of occurrence of these inadvertent actuation events was found to be approximately ten per year.

The Fire Risk Scoping Study was a limited one and did not attempt to quantify the attendant contribution to core damage frequency (CDF) resulting from the inadvertent actuation of FPSs, primarily because the impact of inadvertent fire protection system actuations was found to be very plant specific. It was concluded that such events could significantly impact the risk at a specific plant only if multiple safety systems could be affected by the inadvertent fire protection system actuation event.

This study was begun in 1989. In this study, the potential safety significance of single and multiple FPS actuations is assessed including a more complete review of operational experiences involving such FPS actuations. This review is followed by the development of a methodology for the quantification of effects of fire protection system actuation on safety-related equipment. This methodology has been applied to three specific plants, one boiling water reactor (BWR) and two pressurized water reactors (PWRs), and the results of this quantification have been reported in References 1.2 through 1.4. In addition, analysis of a third PWR was independently conducted by the Idaho National Engineering Laboratory and is reported in Reference 1.5. Building on the insights gained as a result of the analysis of these four plants, this report conducts a core damage frequency and risk assessment of a generic light water plant. A cost/benefit assessment is performed with regard to several plant design issues that were found to be contributors to risk. Technical insights gained from the analysis of issues associated with Generic Issue 57 are presented for the use of those involved with the design of the advanced light water reactor (ALWR).

In the development of the quantification methodology, six main potential causes of inadvertent and advertent actuations of fire protection systems have been identified. These main root causes are presented in Table 1.1. For the general cases of random and seismically induced actuations, several potential root causes are also shown.

An objective of this study was to provide a probabilistic basis on which to evaluate the impact on core damage frequency and risk from fire protection system actuations. This objective was accomplished by first reviewing past events involving fire protection system actuations. The actuations were then categorized in order to draw some useful conclusions about the causes and effects of these actuations. A quantification of the impacts of such events including sensitivity and uncertainty studies, was performed in terms of increase in core damage frequency. An uncertainty analysis was also performed on the generic plant risk.

1.1 Organization of the Report

A review of fire protection system actuation events at U.S. commercial nuclear power plants during the time period of January 1, 1980 to December 31, 1989 is presented in Section 2.0. Included in this section is a discussion of how the data review was conducted and a summary of both the inadvertent and advertant FPS actuations identified in the study. Also, data from FPS actuations at naval shore facilities is discussed. Additionally, in Section 2.0, a summary of foreign nuclear power plant FPS actuation data is presented.

In Section 3.0 the methodology for the evaluation of potential accident scenarios caused by FPS actuation is presented. Included in this section are details on the methodology employed to calculate both core damage frequency and risk for each of the 13 root causes (both seismic and non-seismic). A discussion of the uncertainty analysis methodology is also presented.

Section 4.0 presents a generic plant analysis based on insights gained from the individual plant analytic work as well as the survey of fire protection strategies (Appendix D) at U.S. commercial nuclear power plants. Included are sensitivity analyses for the generic cases examined.

Section 5.0 presents generic cost/benefit information, beginning with the cost-estimating methodology. This methodology is then applied to several design issues that were identified during the documentation reviews, plant walkdowns, and detailed analysis of plant specific and generic scenarios.

Section 6.0 provides technical insights for the ALWR program. This material does not provide specific recommendations or design requirements. Presented are summaries of findings of existing design features that result in contribution of risk due to the effects of fire protection system actuation on safety-related equipment.

Finally, Section 7.0 provides a summary of the report.

Table 1.1

Causes of Potential FPS Actuation

A. Random causes of inadvertent actuation

Human error (Root Cause 4)

Hardware failure (Root Cause 6)

Unknown (Root Cause 13)

B. Actuation induced by fire or by steam pipe break in an adjacent area and smoke/steam spread

Fire in adjacent zone causing FPS actuation (Root Cause 1)

Fire-induced FPS actuation (due to fire in adjacent zone) preventing random failure recovery action (Root Cause 2)

Fire-induced FPS actuation (due to fire in adjacent zone) preventing access for manual fire suppression (Root Cause 3)

FPS actuation caused by steam release (Root Cause 5)

C. Seismic induced inadvertent actuation

Dust actuating smoke detectors (Root Cause 7)

Failure of FPS (e.g., failure of wet pipes, sprinkler heads, etc.) (Root Cause 9)

Actuation caused by FPS control system relay chatter (Root Cause 8)

D. Seismic induced failure of the FPS, diverting suppression agent from an area where a fire is present (Root Cause 12)

E. Fire external to plant (smoke via ventilation system) (Root Cause 10)

F. Fire present where the FPS is located (Root Cause 11)

1.2 References

1.1 J. A. Lambright, S. P. Nowlen, V. F. Nicolette, and M. P. Bohn, Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues, NUREG/CR-5088, SAND88-0177, Sandia National Laboratories, November 1988.

1.2 J.A. Lambright, et al., Risk Evaluation for a Westinghouse PWR, Effects of Fire Protection Systems Actuation on Safety-Related Equipment: Evaluation of Generic Issue 57, NUREG/CR-5789, SAND91-1534, Sandia National Laboratories, December 1992.

1.3 J.A. Lambright, et al., Risk Evaluation for a General Electric BWR, Effects of Fire Protection Systems Actuation on Safety-Related Equipment: Evaluation of Generic Issue 57, NUREG/CR-5791, SAND91-1536, Sandia National Laboratories, December 1992.

1.4 J.A. Lambright, et al., Risk Evaluation for a Babcock and Wilcox Pressurized Water Reactor, Effects of Fire Protection Systems Actuation on Safety-Related Equipment (Evaluation of Generic Issue 57), NUREG/CR-5790, SAND91-1535, Sandia National Laboratories, September 1992.

1.5 G. Simion, et al.,Risk Evaluation of a Westinghouse 4-Loop PWR, Effects of Fire Protection System Actuation on Safety-Related Equipment (Evaluation of Generic Issue 57), EGG-NTA-9081 Letter Report, Idaho National Engineering Laboratory, December 1991.

2.0 REVIEW OF FIRE PROTECTION SYSTEM ACTUATION EVENTS

A complete evaluation of past advertent and inadvertent fire protection system (FPS) actuations was performed. Those events reported in the USNRC License Event Report (LER) data base that fell between January 1, 1980 and December 31, 1989 were included. The objectives of this analysis were the following:

a. Update the data base of inadvertent events that was compiled in the Fire Risk Scoping Study (Ref. 2.1).

b. Review fire events over the entire period of the study which involved advertent actuations to determine their effects on plant safety systems.

c. A review of naval shore facility FPS actuation data (Ref. 2.2).

d. Collect and classify foreign data sources on FPS actuations.

Specific goals for this study included the following:

a. Broaden the scope of event searches to include events that may not have been included in References 2.1 and 2.2.

b. Ensure that all events that occurred during the period of data analysis (1980 through 1989) are included.

c. Summarize plant operating years, the distribution of events per year, and determine the number of events per reactor operating year.

d. Summarize the types of FPSs that were involved in the events, as well as those events that involved multiple FPS actuations.

e. Categorize common cause initiators resulting in FPS actuation.

f. Identify if and when redundant trains of safety equipment were affected.

g. Determine plant areas, systems, and specific equipment categories that have been impacted by FPS actuations.

h. Summarize failure modes of equipment that were adversely affected by the FPS suppression agent release. Identify specific equipment vulnerabilities where possible.

i. Identify events which resulted in or were caused by a plant transient or a fire in another location in the plant.

j. Identify those events involving personnel error and/or procedural deficiencies and identify any human interactions data showing an affect on operator performance resulting from FPS actuations.

k. Identify common cause failures resulting from FPS actuation.

2.1 Background

In early 1982, the Nuclear Regulatory Commission (NRC) identified Generic Issue 57 (GI-57), "Effects of Fire Protection System Actuation on Safety Related Equipment." In addition, several IE Information Notices (Refs. 2.3, 2.4, 2.5, and 2.6) have been issued which alerted nuclear power plant licensees of the potential affects of FPS actuations and gave examples of typical incidents.

Late in 1988, Sandia National Laboratories completed the Fire Risk Scoping Study (Ref. 2.1) for the NRC Office of Nuclear Regulatory Research. Among other issues, this study provided a preliminary review of inadvertent FPS actuations in U.S. commercial nuclear power plants. Seventy-one events resulting in submission of an LER during the period from April 1, 1980 to July 14, 1987 were identified. The average frequency of occurrence of these inadvertent actuation events was found to be approximately ten per year. The Fire Risk Scoping Study was limited in scope and did not attempt to quantify the attendant contribution to core damage frequency (CDF) resulting from the inadvertent actuation of FPSs, primarily because the impact of inadvertent suppression system releases was found to be very plant specific. Many incidents were identified in which safety system damage was reported. However, it was concluded that such events could significantly impact the risk at a specific plant only if multiple safety systems could be affected by the FPS actuation event.

The issue of the potential damaging effects of fire protection systems is comprised of two related topics: (1) the release of fire suppressant during actual fires (i.e. advertent events), and (2) the spurious operation of fire protection systems when there is no fire (i.e. inadvertent events). In many ways, fire-induced FPS actuations provide more severe exposure conditions than spurious actuations. In addition to the effects from water (flooding, spray, humidity) and the gaseous suppression agents (low temperatures, high thermal gradients, high differential pressures, and high static charge levels), fire introduces the effects of elevated thermal exposures, smoke deposition, and interactions of smoke, moisture, and corrosive compounds generated by the fire. Therefore, it is instructive to study the effects of fire suppressants on plant equipment under both conditions involving fires with suppressant, and with fire suppressant release only. Also, the studies of advertent and inadvertent actuations can provide insights into the potential damaging effects of the misapplication of suppressant agents during manual fire fighting efforts. As there is little available experimental data on the effects of suppressant generated environments on plant equipment, guidance must be garnered from the operational experience base.

2.2 Approach to Review and Update the LER Event Data Base

The primary source of information for the scoping study (Ref. 2.1) was the Licensee Event Report Data Base maintained by the Nuclear Operations Analysis Center at Oak Ridge National Laboratory. LERs are submitted to the NRC by individual nuclear power plants in the United States to report events that affected the safe operation of the plant. Currently, an LER is required only if safe plant operation is actually or potentially affected. Therefore, it is possible (even likely) that non-safety related inadvertent actuations were not reported. An initial limited search from the period of April 1980 through June 1988 resulted in 127 LER abstracts involving the actuation or operation of fire protection systems at nuclear power plants. From these 127 events, a set of 75 inadvertent actuations was derived.

Only inadvertent actuations were initially of interest. Additional information was available from a set of 108 LER abstracts involving actual fires. A review of these fire events verified that none involved the inadvertent actuation of additional fire protection systems beyond those required for suppressing the fire. The current study includes both advertent and inadvertent FPS actuation events. Therefore, the advertent actuations noted in the set of 108 abstracts are now included in the overall event data base.

One initial concern with the scoping studies was the completeness of the LER searches and the resulting data base. Also, the LER abstracts varied greatly in the amount of detail they provided. Many of the abstracts included sufficient detail to categorize the event without further research. However, many of the LER abstracts provided very limited information. For example, in many cases it was not possible to determine what type of FPS actuated, what the plant power level was at the time of the event, what the initiating causes were or what was the failure mode of the affected equipment.

An initial task of this study was to review the past searches. The intent was to provide an independent review for completeness and consistency in the assignment of information categories (data base fields). All of the LER summaries were reviewed with the goal of compiling a comprehensive set of keywords to use for conducting a much broader search of the LER data base. This information was discussed with Oak Ridge personnel who are responsible for the development and use of the data base. From this effort, 2795 LER abstracts were obtained from Oak Ridge for review covering the period from January 1, 1980 to December 31, 1989. A preliminary screening of this entire set of abstracts was then performed to eliminate those that were obviously unrelated to FPS actuations. Next, a detailed review of the remaining LERs of interest was performed. For the LERs for which insufficient information was provided in the abstract to determine whether FPS actuations occurred, an attempt was made to obtain the full text of the original LER for further review. Over 100 full LERs were obtained primarily from the Sandia Technical Library with the remainder provided by Oak Ridge.

In an attempt to reduce the uncertainty associated with the LER abstracts and the full text of the LERs, a survey was conducted of all nuclear utilities. The objectives of the survey were to make actual plant contacts to (a) verify questionable information in the LER abstracts or full LERs, (b) provide missing information left out of the LERs, (c) report additional fire protection system actuation events that may not have shown up in the LER searches, and (d) provide data on installed fire protection system configurations. Much new information was acquired, including 19 new events not reported in the LERs. Some of the information still remains unverified. Therefore, some uncertainty remains for certain events (such as power level, failure modes, etc.) although enough information was gathered to insure confidence in the basic facts associated with each event identified and included in the final data base.

To categorize the FPS actuations, the remaining screened list of actuation events was reviewed with certain questions in mind. The major question was what, if any, safety-related frontline or support systems were affected by the actuation. This question was further broken down into identifying the specific plant areas, plant systems, equipment affected, and the failure mode of that equipment. Another item of interest was whether the actuation was related to a plant transient, either immediately before or after the actuation. Several questions dealt with whether or not the actuation was associated with an actual fire and, if so, whether the fire was in the associated fire zone, in another fire zone internal to the plant, or external to the plant itself. Other items of interest were the cause of the actuation, the FPS component that initiated the actuation, and how many (and what type) fire protection systems actuated in each incident. In addition, the date of the incident, the type of nuclear plant involved, and the power level at the time of the incident were noted. To assist in the review and to ensure consistency, these questions were arranged into a checklist. The completed checklist sheets summarizing each event are included as Appendices A.1, A.2, and A.3 (seperately bound).

Finally, a summary of reactor operating years for all U.S. reactors was compiled for the period of this study. A primary source of information and data for this effort came from the Sandia Fire Data Base (Ref. 2.7) which provided station and individual unit operating years, both for the period prior to January 1, 1980, and for the period from January 1, 1980 to June 30, 1985. Operating years for the balance of the period (July 1, 1985 to December 31, 1989) were derived from the latest available issue of Nuclear Safety (Ref. 2.8). Between January 1, 1980 and December 31, 1989, the total number of reactor operating years from initial criticality (including shutdown periods) was approximately 878. (The shutdown times were included because several of the FPS actuations occurred during refueling or other shutdown periods). Total annual and cumulative operating years for each year of the study were further broken down into reactor type (BWR, PWR, HTGR). Figure 2.1 shows operating years by year (for the 1980's only), and Figure 2.2 depicts the cumulative total operating years during the 1980's. For the purposes of quantification, the operating years prior to 1980 were subtracted to yield the incremental total operating years for the period of the study only.

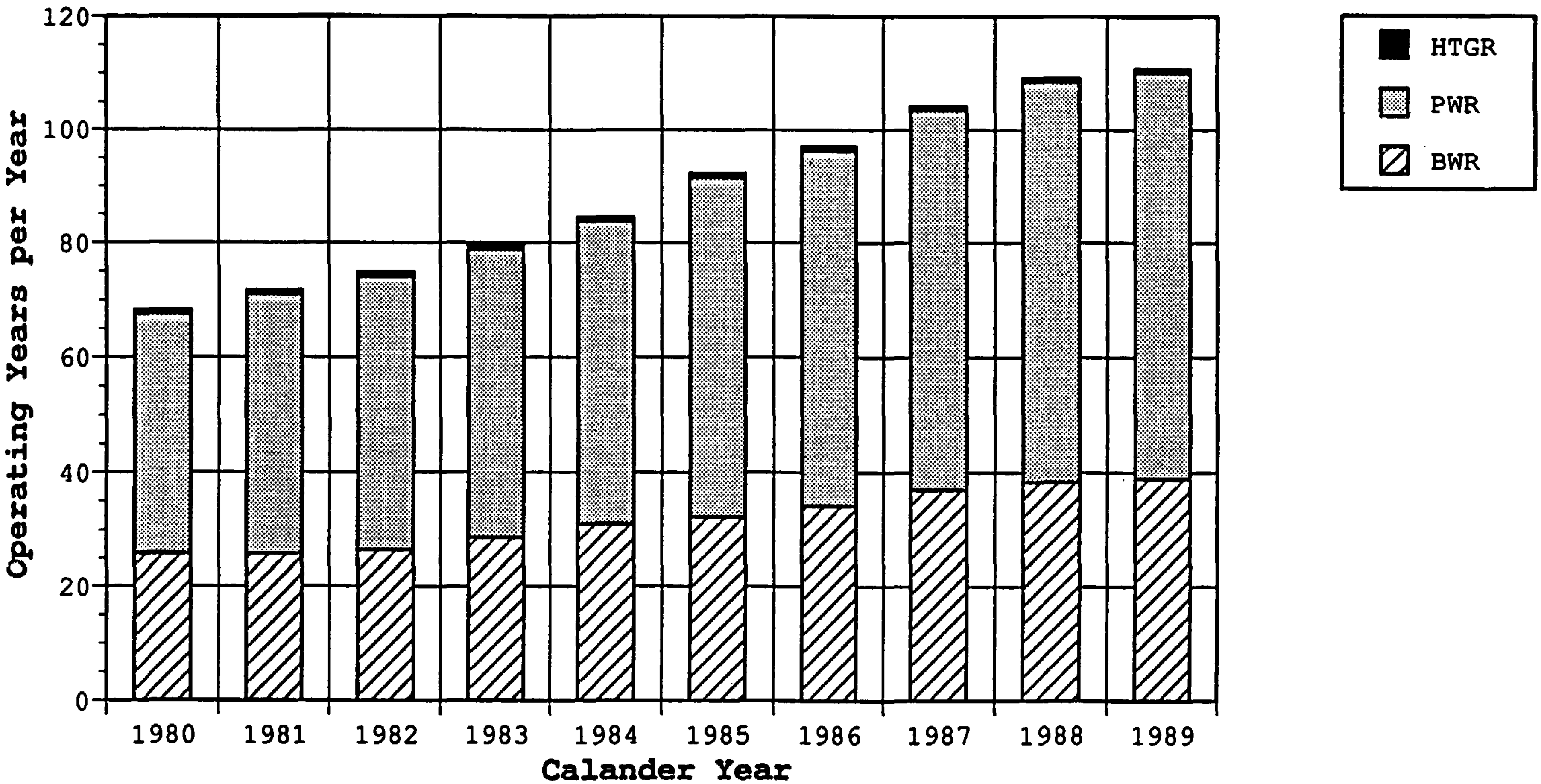

Figure 2.1 - Summary of Operating Years

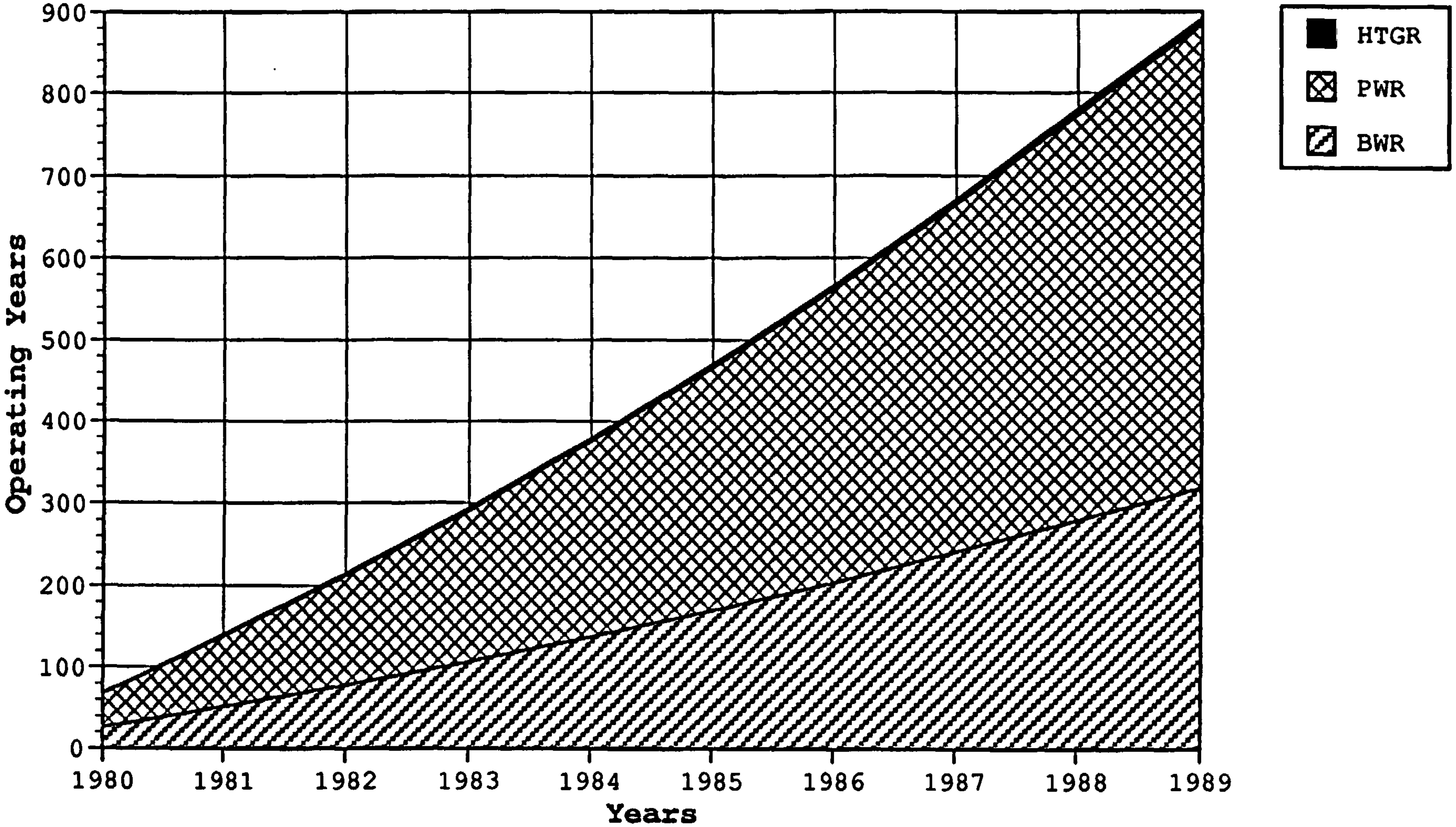

Figure 2.2 - Cumulative Operating Years

2.3 Results of LER Event Review

From the updated set of 2795 LER abstracts reviewed for this current study, five advertent events and 55 inadvertent FPS actuation events were identified and added to the data base. Thus, the total number of events in the data base as provided in the data sheets in Appendix A is 150 separate events, 121 of those inadvertent (Appendix A.1), 17 advertent (Appendix A.2), and 12 inadvertent but before initial criticality (Appendix A.3).

An event was considered advertent if the fire protection system either automatically actuated as designed or was manually initiated due to fire, smoke, or other environmental causes for which FPS operation would be required. An event was considered inadvertent if a fire protection system (or systems) actuated without the presence of a fire (in the associated fire zone) requiring the fire protection system. As an example of both, the transformer fire at Palo Verde 1 on July 6, 1988 involved an advertent manual actuation of the deluge system for the unit auxiliary transformer that caught fire, and three inadvertent manual actuations of the deluge systems for the adjacent transformers. Most of the events involved the application of the fire suppression agent in the designed manner, i.e., from the sprinkler head or nozzles, but at the wrong time. Some of the events involved leaking or ruptured system components or other similar breaches of the FPS resulting in untimely release of suppression agent. For example, water leaking from a deluge valve or pipe was relatively common. Many of the LERs reported failures of the fire protection systems or components which involved FPS operability considerations and lack of adequate fire protection for critical plant areas and systems. However, these events were eliminated from consideration if they did not result in the release of suppression agent into the protected area. Also, numerous events involved tests of the FPS or tests of a specific component which then failed the test.

These were also excluded from consideration. Maintenance activities which led to an unexpected discharge because of either inappropriate procedures or failure to follow proper procedures are included in the data base.

Not all of the events reported in the data base as shown in Appendix A are used for quantification. The inclusion of the events in the data base, whether used in risk quantification or not, is important for a number of reasons. Though many of the events had no impact on safety systems or occurred in plant areas of no concern from the standpoint of safe plant operation, they were nevertheless included in the data base for informational purposes. That the event occurred, regardless of location, is noteworthy. Many of the events could have occurred anywhere, at any time and at any plant power level. Many of the events involved plant equipment or occurred in plant areas that are not typically of concern for plant shutdown or for decay heat removal purposes following plant shutdown. For instance, 15 of the events involved wetting of charcoal filters in ventilation system filtration trains or the Standby Gas Treatment System. In many of these events, the plant was forced into a Technical Specification Limited Condition for Operation, though plant operation was not otherwise affected. Seven of the events involved the rupture of fire mains, fire hoses, or temporary (and under-designed) piping in outside areas (general yard areas, switchyards, etc.). Six of the events involved the deluge systems at the cooling towers, where the primary affect was excess drawdown of the fire water supply tanks. FPS operability was affected, but not other important plant safety systems. A number of events (12) actually occurred prior to initial criticality (Appendix A.3). Since the operating years were calculated from initial criticality, and events during initial construction and preoperational phases were excluded, these events were not included in risk quantification. However, they are included in Appendix A.3 for information. Having occurred just prior to initial criticality, they are considered important to note as events which again could happen at any time. It is the task of the risk analyst to provide for the screening of events in the quantification process.

The total number of events can be subdivided according to the year of occurrence. Figure 2.3 provides a simple bar chart showing this distribution and Figure 2.4 shows the breakdown by plant type for each year of the period of study. The maximum number of events in one year (24) occurred in 1983. The minimum number (five) occurred in 1988. The average number of events per year is 15, with six events per year for BWRs (60 events) and nine events per year for PWRs (90 events).

However, the total number of events and the breakdown by plant type can also be compared to the total number of reactor years to obtain a frequency of occurrence. For the full data base, the overall frequency of occurrence is approximately 0.17 FPS actuations per reactor year (see Figure 2.5). When subdivided by plant type, the frequency of occurrence for BWRs is 0.19 events per reactor year and 0.16 events per reactor year for PWRs.

2.3.1 Summary of Advertent Events

The 17 advertent events identified in Appendix A.2 are included in the above totals and were included in risk quantification along with the inadvertent events. However, they are also summarized here to provide an additional perspective on the characteristics unique to these type of events as opposed to events involving inadvertent FPS actuations. Overall, there were 0.02 advertent events per reactor year distributed over the ten years of this study as shown in Figure 2.6. Eleven events occurred in PWRs, six in BWRs. All of the actuations were the result of fire, smoke, electrical arcing, etc., in the associated fire zone. None were due to smoke spread from adjacent zones or from outside the plant. Six of the events involved manual actuations of the fixed systems by fire brigade personnel; the other eleven were automatic actuations. There were 12 water-based system actuations, including two actuations (multiple actuations in a single event) at Duane Arnold (November 4, 1984) two actuations at North Anna 2 (July 31, 1981), and four deluge actuations at

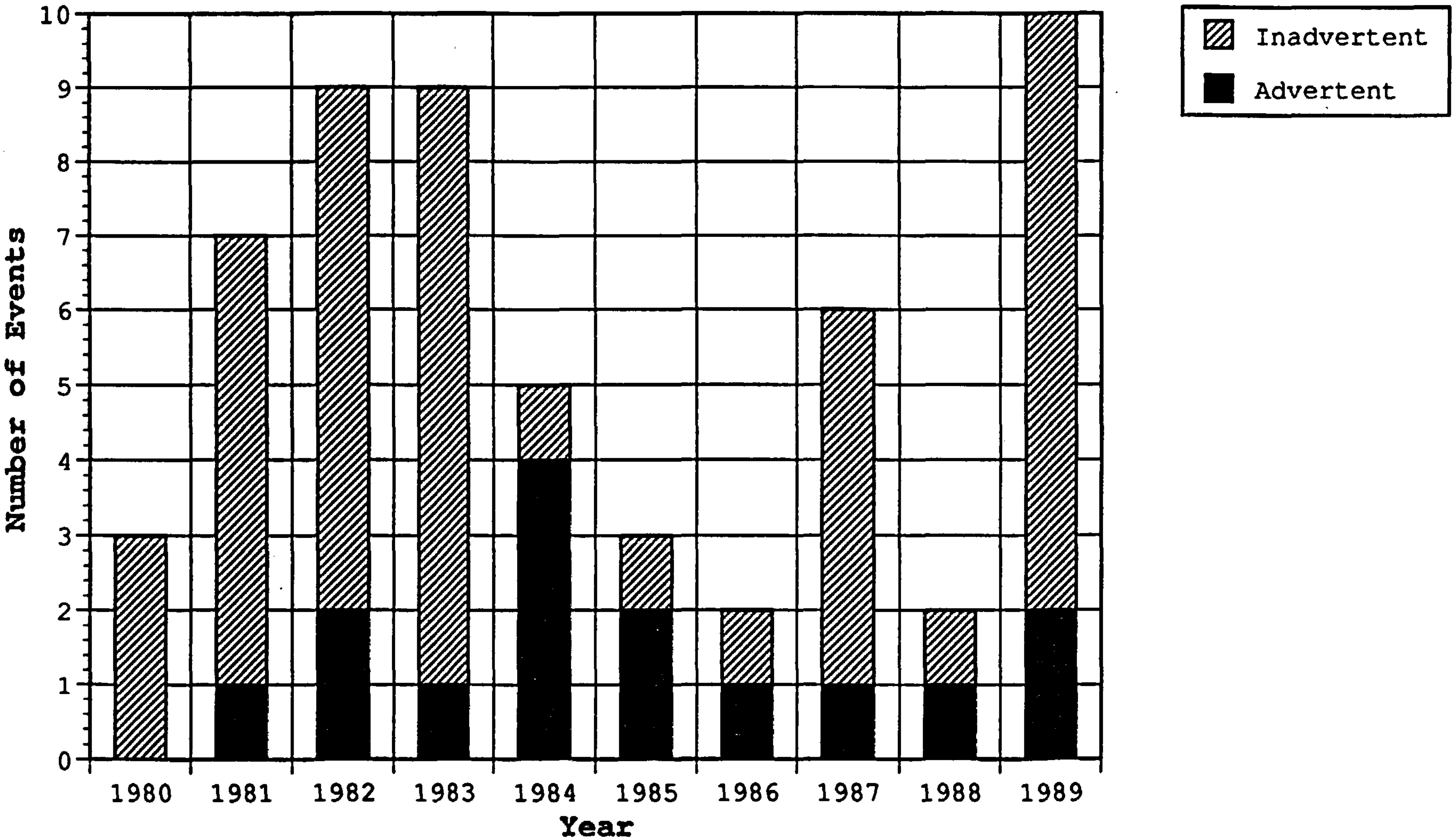

Figure 2.3 - New Events Added to LER Data Base

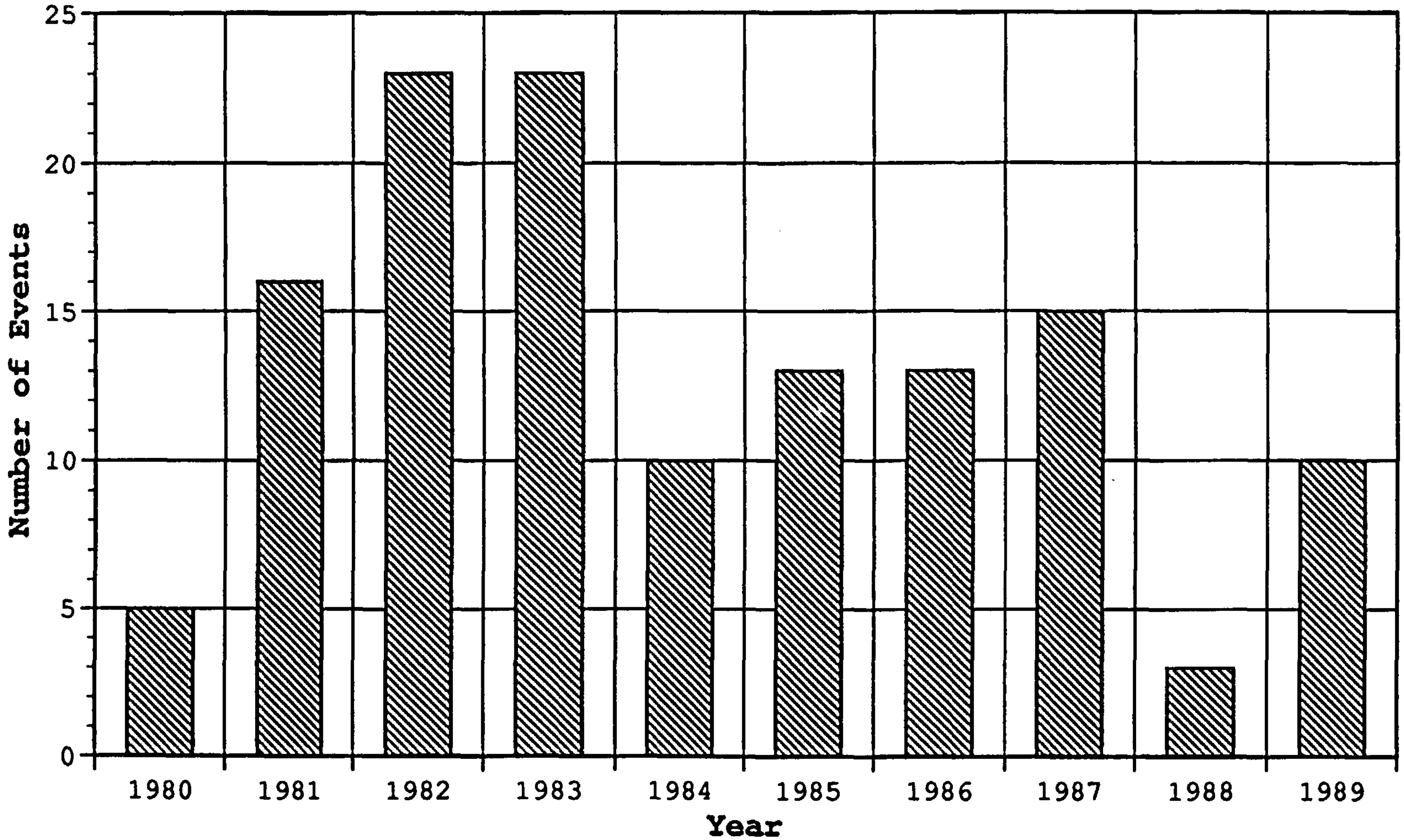

Figure 2.4 - Total Events per Year

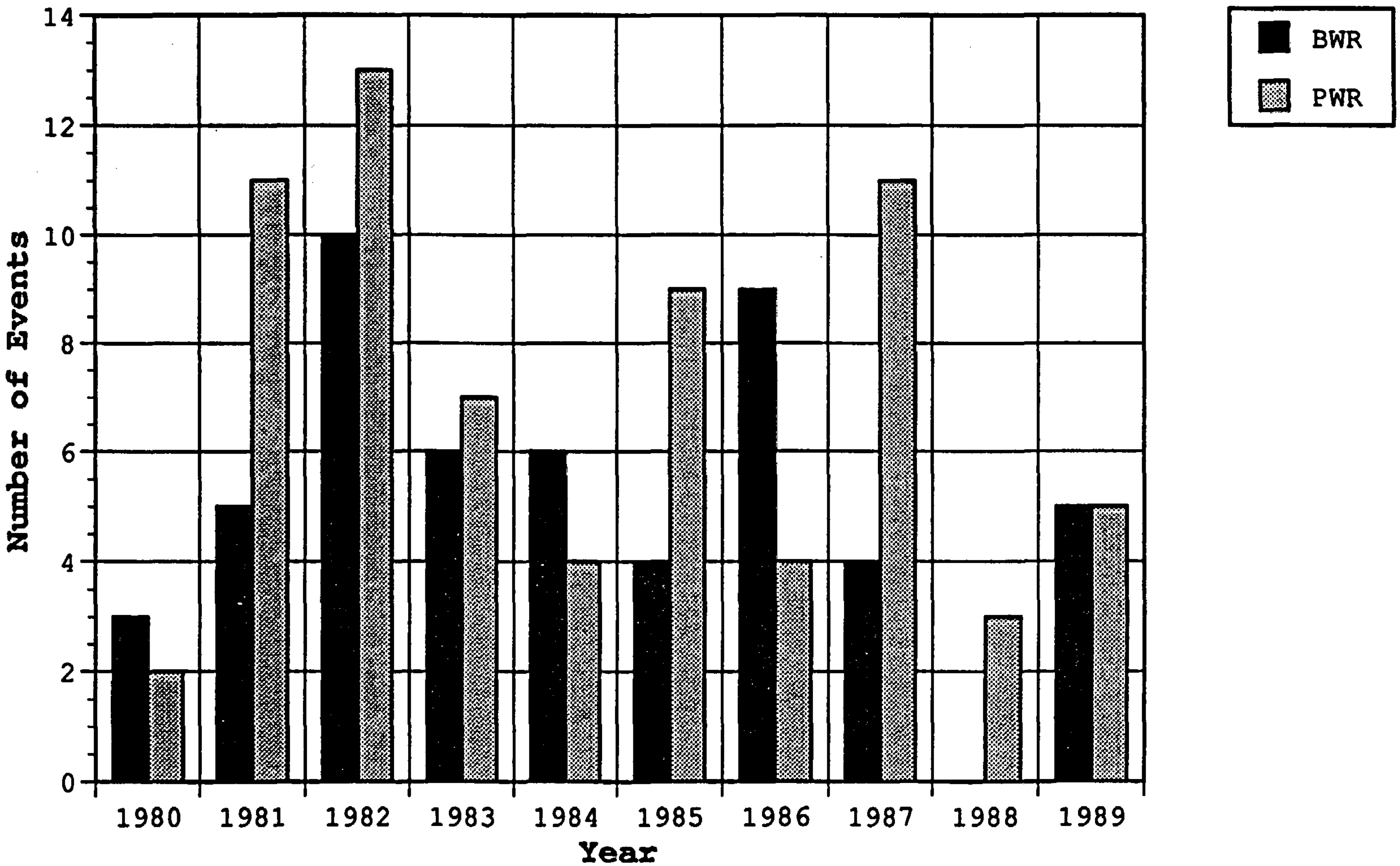

Figure 2.5 - Events per Year

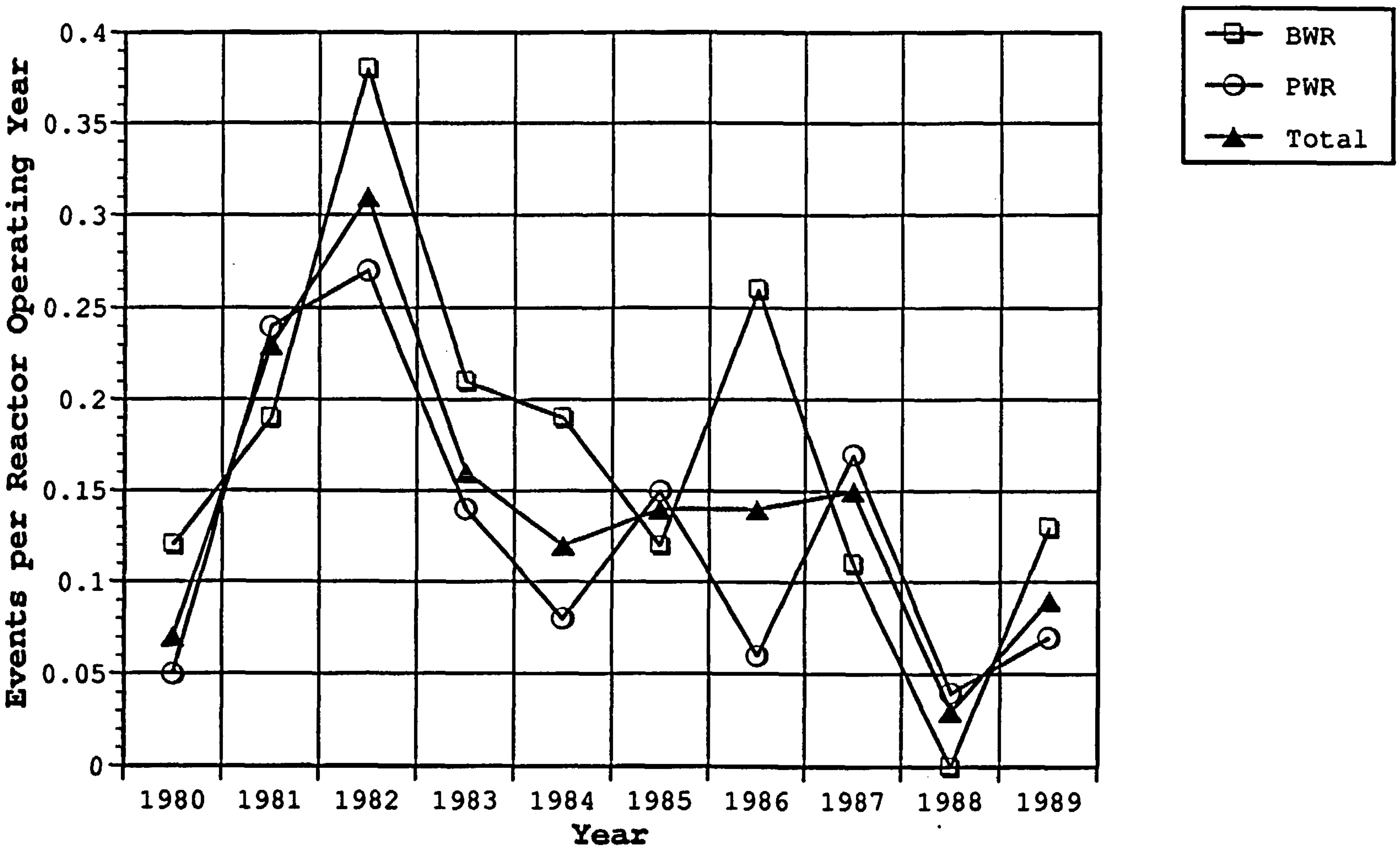

Figure 2.6 - Events per Operating Year

Palo Verde on July 6, 1988 (as indicated earlier, three of these were considered inadvertent). Six events involved CO_2 system actuations, three events involved Halon systems, and one event involved a fire protection system of unknown suppression agent. Eight of the events occurred at greater than 50 percent power level, one between 5 percent and 50 percent power, one at low power (<5 percent), four during unit shutdown or refueling, and three at an unknown power level. Eleven of the events led to plant shutdowns or a significant plant power transient, primarily due to the fire itself. In one event (Oyster Creek, February 18, 1982, fire water wetted switches, adversely impacting safety systems. In another event (Vermont Yankee, March 3, 1989) CO_2 leaked past fire doors, actuating the control room toxic gas monitoring system.

2.3.2 Summary of All Events

Regarding the full set of FPS actuations (both advertent and inadvertent), a significant number of the events occurred during normal power operations (see Figure 2.7). As noted previously, 12 of the events occurred prior to initial criticality. Over 35 percent of the events occurred at power levels greater than 50 percent, with another eight percent of the events at lower power levels, and over 11 percent at some unspecified power level. Approximately 23 percent of the events occurred during refueling outages or other periods when the unit was shutdown. Twenty-two of the events (just under 15 percent) occurred at unknown power levels. It is clear that FPS actuations can occur at any time, regardless of power level.

For all events which reported power level (excluding the 15 percent that occurred at unknown levels), 55 percent occurred at power. From Reference 2.9 the availability factor for all U.S. light water reactors (LWRs) from 1968 through 1986 was 68.4 percent. The availability factor is defined as the percentage of time plants were either operational or available to be operational. Events appear to occur at relatively the same frequency whether plants are operational or shutdown.

Events which took place during shutdown or refueling periods were only included for quantification purposes if they could have also occurred at power. Time periods when plants were shutdown or refueling were also included in the overall calculation of total operating years. Therefore, eliminating these events and correspondingly reducing the time period under consideration would have a negligible effect on initiating event frequencies for the identified root causes.

When the FPS actuations are classified according to the type of fire protection system that actuated, it is seen that the majority involved water-based systems. Table 2.1 provides a breakdown of the FPS actuations according to whether the event involved water, Halon, CO_2, several, or unknown system types; whether the FPS actuation was advertent, inadvertent, or unknown; and whether the totals included FPS actuations prior to initial criticality. Two of those events involving "several" water-based systems (Ginna, November 14, 1981 and Salem 1,

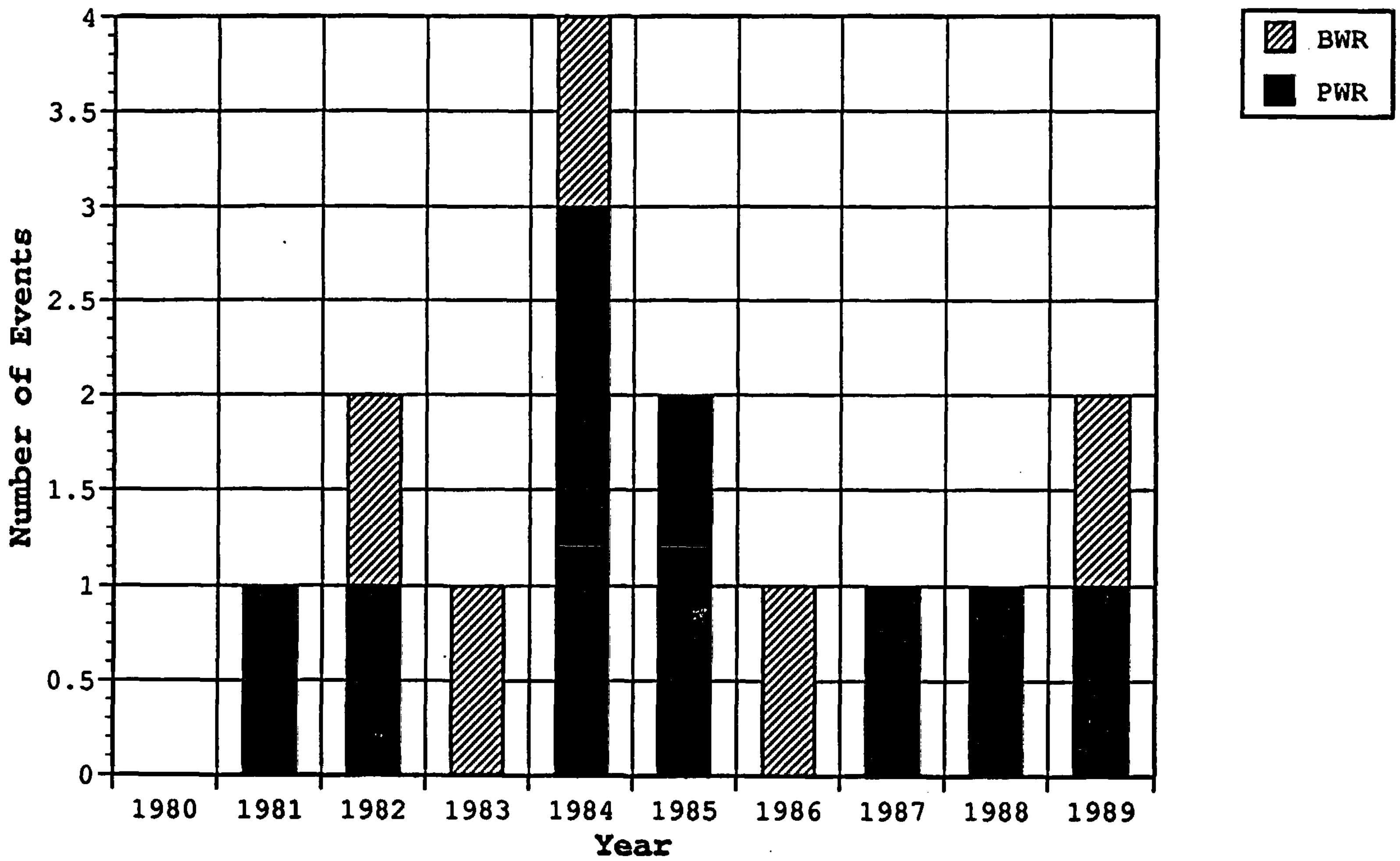

Figure 2.7 - Advertent Events by Year

November 9, 1982) did not specify in the LER abstracts either the exact number of system actuations or, in the case of Ginna, the exact plant locations that were affected. For the water-based and CO_2 actuations at Dresden 3 (May 4, 1983) and Cook 2 (October 27, 1982) it was unclear whether the actuation was advertent or inadvertent. The full LERs did not provide sufficient detail to categorize these. Plant contacts were made in an attempt to verify information lacking in the LERs; however, no new information was gained by this approach. Therefore, for the purposes of this study, they were considered inadvertent. Finally, the event at San Onofre 3 (April 8, 1985) involved a hydrogen ignition that was suppressed by an unspecified fire protection system (CO_2 is likely, but not assumed).

The numbers provided in Table 2.1 total to more than the 150 events as reported in the LERs as some of the events led to the actuation of more than one fire protection system. Table 2.2 summarizes the specific events wherein multiple FPS actuations occurred. As mentioned earlier, the events at Ginna and Salem 1 involved the actuations of "several" water based or deluge systems (caused by personnel error and an unknown cause, respectively). During the Surry 2 event in December, 1986 in which the feedwater pipe break occurred, all three types of systems were actuated: the 62 turbine building sprinklers first and then the CO_2 and Halon systems. A similar event affecting Surry 1 involved a pipe break, steam release, sprinkler actuation, and moisture intrusion into Halon system controller, and subsequently Halon system actuation. The events at Cooper (caused by personnel error) and WPPSS 2 (due to a steam leak) in 1984 involved the actuations of two and three water-based systems, respectively. Four of the events, all at Three Mile Island, involved the actuation of a deluge system and a Halon system in the Air Intake Tunnel at the same time. The causes were lightning (twice), welding, and an unknown cause. The Duane Arnold and North Anna 2 events both involved large transformer fires which resulted in the actuations of deluge systems for the damaged transformer and an adjacent transformer. The Palo Verde 1 transformer fire resulted in manual actuations of the deluge systems for the affected transformer, plus three adjacent transformers, due to operator uncertainty as to exactly which deluge system covered the burning transformer.

When the entire set of FPS actuations are analyzed for an initiating cause, the most common initiator is found to be human error (see Figure 2.8). In 36 cases, the cause was very simply personnel error: not following proper procedures, miscommunication, inadequate design, bumping or accidentally actuating switches, or other mistakes (one event even involved suspected tampering or sabotage). In this report, this type of error is distinguished from the more subtle error of a mistake in or omission from a documented test or maintenance procedure. This latter type of error resulted in 12 actuations during the performance of a test or maintenance procedure. Personnel and procedural related events thus totaled almost 31 percent of the events. In 30 of the events (over 19 percent), breaches of the fire protection systems resulted in leakage of fire suppressant which then caused damage or adversely impacted plant

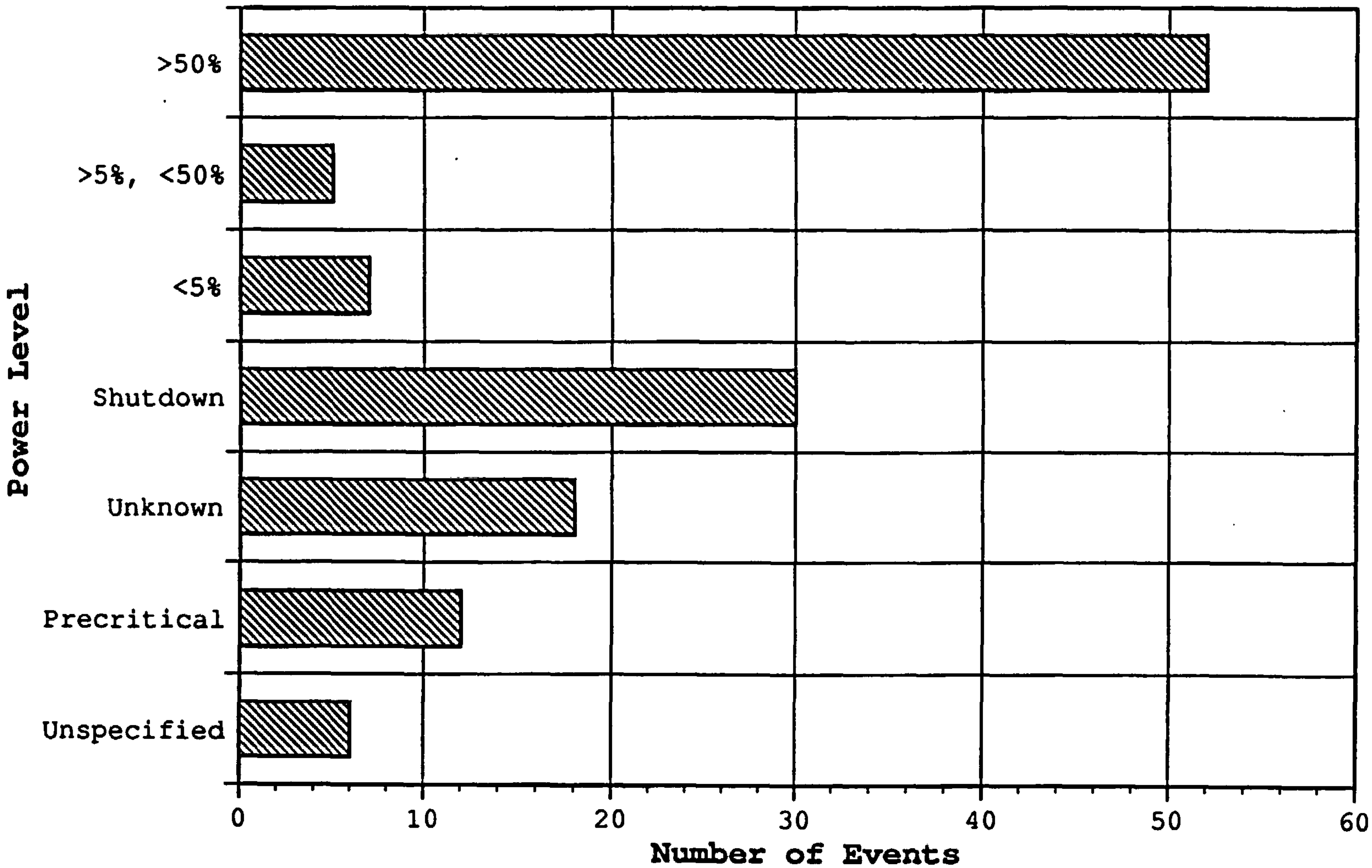

Figure 2.8 - Events by Power Level

Table 2.1

Types of Fire Protection System Actuated

	Water	"Several"	CO_2	Halon	Unknown
Advertent:	12	-	6	3	1
Inadvertent:	114	2	12	13	-
Unknown:	1	-	1	-	-
TOTAL:	127	2	19	16	1
Less Pre-Crit:	9	-	2	1	-
To Quantify:	118	2	17	15	1

equipment. Eighteen of these events involved valve leakage or piping system leakage or failure due to actual breaks or erosion/corrosion effects. Seven of the events were caused by fire main or valve ruptures, or hose ruptures, in outside areas. Another five events involved suppressant leakage due to mechanical failures of the fire protection system boundary (gaskets, valves, manifolds). The initiating cause was unknown or not reported for 31 of the 150 events. Advertent actuations due to fires, explosions, etc., occurred in 17 cases (11 percent). Over 11 percent of the events were due to environmental initiators: steam, dust, or high humidity caused actuations in 12 cases; welding, smoke without a fire, or hot equipment set off systems in four events; and lightning was blamed for two of the actuations. (Detector sensitivity probably contributed to these actuations due to environmental causes.) Pump starts and other pressure spikes in the FPS piping and air lines led to actuations in five percent (or seven) of the events. Threeactuations were due to water from other sources in fire detectors, and three of the events were caused by various electrical failures in the fire protection systems. In addition to the initiating cause, each actuation was also characterized by the fire protection system component that initiated the actual release of FPS agent. For example, if plant personnel inadvertently shorted the FPS control circuitry during a maintenance activity, the initiating cause would be personnel error and the initiating component would be the fire protection system control circuits. The breakdown by initiating component is shown in the

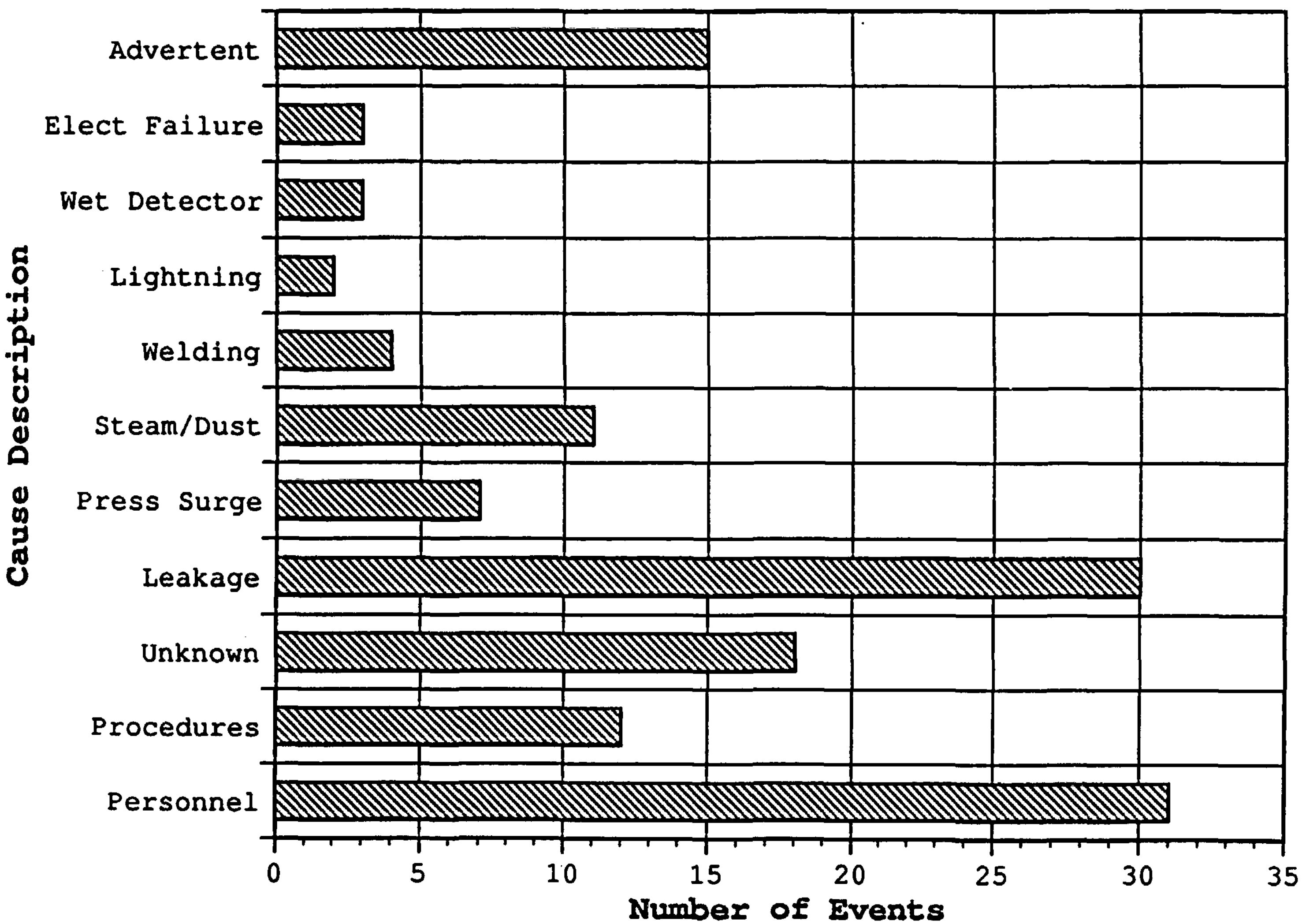

Figure 2.9 - Initiating Causes

Table 2.2

Multiple FPS Actuations

LER Number	Plant Name	Type	Date	Notes
244/81-019	Ginna	PWR	11/14/81	"Several" Inadvertent Water
272/82-087	Salem 1	PWR	11/9/82	"Several" Inadvertent Water
280	Surry 1	PWR	3/23/89	Inadvertent Water (1), Halon (1)
281/86-020	Surry 2	PWR	12/9/86	Inadvertent Water (1), CO_2 (1), Halon (1)
298/84-007	Cooper	BWR	4/19/84	Inadvertent Water (2)
320/82-018	TMI-2	PWR	6/1/82	Inadvertent Water (1), Halon (1)
320/82-023	TMI-2	PWR	6/29/82	Inadvertent Water (1), Halon (1)
320/83-009	TMI-2	PWR	3/3/83	Inadvertent Water (1), Halon (1)
320/83-014	TMI-2	PWR	5/6/83	Inadvertent Water (1), Halon (1)
331/84-040	Duane Arnold	BWR	11/4/84	Advertent Water (2)
339/81-055	North Anna 2	PWR	7/3/81	Advertent Water (2)
397/84-096	WPPSS 2	BWR	9/1/84	Inadvertent Water (3)
528/88-010	Palo Verde 1	PWR	7/6/88	Advertent and Inadvertent Water (4)

bar graph in Figure 2.9. About 27 percent of the events were initiated by FPS valves, which opened, leaked, failed, or were manually opened due to various root causes. In 23 percent of the incidents, the actual FPS component that failed or actuated to release suppression agent was unknown or not reported. Piping, fittings, or hoses were the components that failed in 20 events (13 percent). Detectors or detector systems actuated the fire systems in 28 cases, eleven of those due to advertent actuations required by fires. Six of the events were due to manual advertent actuations wherein no component failed, and the actual component that was used for the actuation was not specified. Failures of relays, switches, electrical control panels/components and circuits accounted for another eight percent of the events. Failures or actuations at the sprinkler heads themselves (melted links, physical damage) occurred in five of the events (all of these were inadvertent events). Finally, four events were caused by mechanical failures of pressure regulators or compressors.

Eight of the inadvertent actuations occurred as a result of or during reactor and turbine trips in progress or after a loss of offsite power (refer to Table 2.3). Four of the events involved BWRs, four occurred in PWRs. Six of the eight events occurred during power operation and involved water system actuations due to pipe breaks or scram discharge volume leaks where steam then actuated the deluge systems (at the sprinkler heads directly, by shorting control circuits, or by actuating detectors). One event at Vermont Yankee involved a loss of offsite power during refueling that led to pump starts, a pressure surge, and the rupture of under-designed temporary FPS piping, spilling over 2000 gallons of water on the reactor building refueling floor. The other event (River Bend) involved main transformer arcing due to animal intrusion that led to a main generator trip and reactor scram. Three of the BWR events occurred in the reactor building and specifically the RCIC room at Hatch 2. The other BWR event (River Bend) occurred in the main transformer area. Actuations occurred for PWRs in the turbine building at Trojan and Arkansas Nuclear One, and both the turbine building and service building during the Surry events where water, CO_2, and Halon systems all actuated. The Surry, Hatch, and Arkansas Nuclear One events all adversely impacted important plant systems and equipment. In fact, the actuations at Surry and Arkansas Nuclear One actually caused additional perturbations to plant operations.

It is of interest to note the plant locations where these FPS actuations occurred. It must be recognized that, due to the provisions of 10CFR50 Appendix R fire regulations, those fire zones within a nuclear power plant which are protected by fixed fire protection systems are generally those housing safety-related equipment. Thus, FPS releases will typically involve the exposure of safety related equipment. Table 2.4 provides a summary of plant areas as they are identified in the LER abstracts. The designations are both general (such as the reactor building, or outside areas) and specific (such as the RCIC room) due to the variation in the level of detail provided in the LERs. Also, the affected plant areas have been broken out by advertent and inadvertent events. The LERs that report fires provide information regarding the effects of the fire, which is of primary interest. Very little detail is provided regarding any additional effects of the FPS actuation. Therefore, those areas affected during advertent incidents were listed separately from the areas affected by inadvertent fire protection system actuations. Almost all of the major areas in nuclear power plants have been impacted by FPS actuations, including those areas containing safety systems and equipment important to stable plant operations.

Table 2.3

Actuations Resulting From Plant Transients

LER Number	Plant Name	Type	Date	Notes
219/85-012	Oyster Creek	BWR	6/12/85	Steam, 1-Water, Reactor Bldg
271/87-008	Vermont Yankee	BWR	8/17/87	Surge, 1-Water, Reactor Bldg
280	Surry 1	PWR	3/23/89	Steam, Water, Halon, ESGR
281/86-020	Surry 2	PWR	12/9/86	Steam, Water, CO_2, Halon
344/85-002	Trojan	PWR	3/9/85	Steam, 1-Water, Turbine Bldg
366/82-100	Hatch 2	BWR	8/25/82	Steam, 1-Water, RCIC Room
368/89-006	Arkansas N-2	PWR	4/18/89	Steam, 1-Water, Turbine Bldg
458	River Bend 1	BWR	9/6/88	Arcing, Main Transformer Trip

A significant result of this review is the finding that 56 of the 150 FPS actuation events damaged or adversely impacted other systems in the plant. Of these, 32 events were categorized as having affected safety-related or other frontline/support systems that are risk significant, and that often severely affected plant operations or system and equipment operability. Table 2.5 provides a summary of those systems that have been impacted by FPS actuations. FPS actuations have been responsible for degradation or spurious operation of transmission/distribution systems, vital ventilation systems, ECCS and ESF systems, systems vital for reactor operation and protection, the turbine generator subsystems and controls, diesel generators, and other fire protection systems. Regarding the 24 remaining events of the 56 mentioned above (excluding the 32 events just discussed), 15 involved systems in which the failure mode was wetting of charcoal filters, and nine events involved other plant systems and equipment not needed for safe, reliable plant operation.

Another nine events (not included in the 56 mentioned in the previous paragraph) involved the drawdown of the fire system water supply tanks below minimum required levels, thereby adversely impacting fire protection for those areas served by the tanks. Other events also

Table 2.4

Affected Plant Areas

Area Description	Inadvertent	Advertent	Total
Transformer Areas	22	4	26
Reactor Building, Containment	21	1	22
Turbine Building	15	4	19
Miscellaneous Service Areas, Radwaste	16	1	17
Cable Rooms, Vaults, Tunnels, Shafts	16		16
Auxiliary Building	14	1	15
Switchgear, Relay, MCC, Battery Rooms	10	2	12
Control Room, Building, Computer Room Office Areas	10	1	11
Unknown	10		10
Diesel Generator Rooms	6	3	9
Outside, Switchyards	7		7
Cooling Tower Area	6		6
HPCI Room	4		4
RCIC Room	1		1
Fuel Handling Building	5		5
Pump Rooms	3		3
Recombiner Building	2		2

impacted the availability, operability, or fire protection capabilities of the affected fire systems and detection systems. For instance, when full Halon or CO_2 discharges occurred, fire watches were required until the tanks could be recharged. In a number of events, the actuations resulted in degraded fire detection operability and masking of fire detection in other adjacent areas. Many events not included as having affected plant systems and equipment did, however, result in evacuations, or isolations/actuations of equipment by design. For instance, five of the events involved evacuations following CO_2

Table 2.5

Affected Plant Systems
(Risk Important Systems Marked With Asterisk)

System Description	Inadvertent	Advertent	Total
Other Fire Protection	19		19
Transmission/Distribution (Transformers, Switchgear, Busses, DC, etc.)*	13	4	17
Ventilation, HVAC, Filtration	14		14
HPCI, RCIC*	8	1	9
Core Spray, RHR, ESFs*	3	1	4
Standby Gas Treatment	7		7
Air Cleanup, Monitoring, Toxic Gases	1	1	2
Reactor, RCS, RCPs, Pressurizer*	3		3
Reactor Protection System*	2	1	3
Control Rod Drive System*	1		1
Turbine/Generator, Hydrogen Seal Oil*	3	4	7
Secondary (Feedwater, Main Steam, Pumps)	6		6
Circulating Water	6		6
Diesel Generators*	2	3	5
Drain Systems	2		2
Instrumentation*	1		1
Containment Isolation*	0	1	1
Instrument Air*	1		1
Plant Computer	1	1	2
Hydrogen Recombiner	1		1
Access Control	1		1
Communications	1		1
None Affected	40	2	42

discharges (prior to initial criticality, the Grand Gulf 1 incident on July 13, 1982 resulted in total evacuation of the Auxiliary Building when CO_2 overpressure blew open a locked fire door). Both CO_2 and Halon actuations have caused control room ventilation isolations. A Halon actuation at Millstone 2 (October 9, 1982) tripped the plant computer. In summary, even during events in which FPS actuations may not severely impact safety systems, the impacts on other plant systems and plant operations can be significant (e.g. Halon actuation coupled with ventilation isolation requiring donning of breathing apparatus that degrades operator communications).

Table 2.6 provides a similar summary of specific equipment that was damaged, degraded, or adversely impacted as a result of the FPS actuations. In some cases, only the particular system that was affected is specified in the LERs. In other cases, no system was specified, but the particular affected equipment or component was identified. The piece of equipment that was damaged most often was charcoal filters and filter units due to wetting by deluge systems, resulting in system inoperability and often plant operational limitations. Damage to control panels, instrument racks, junction boxes, load centers, motor control centers, switchgear, electrical busses, and miscellaneous switches, indicators, monitors, and other plant instrumentation significantly impacted system operability and often led to plant transients. Numerous events involved FPS actuations that shorted transformers or tripped transformers off-line due to system interlocks.

In conjunction with the analysis of the affected plant equipment, the particular failure mode of that equipment was also determined when possible (Table 2.7). The investigation found that the most common failure mode was electrical shorting caused by water from the fire protection system. As can be seen in Table 2.7, this occurred in 26 of the events. In another six events, the FPS actuation directly caused equipment (and plant) trips due to interlocks with the fire protection system. Wetting of charcoal filters was the failure mode in 15 events, and degradation of other fire protection systems occurred in 13 events. Other common failure modes include wetting of equipment, water damage, water contamination, corrosion, and adverse impacts from CO_2 and Halon (evacuations, system isolations, contamination, and icing).

The seriousness of the equipment either damaged or affected is further indicated by the number of actuations which resulted in a plant transient. Indeed, 30 of the 150 FPS actuations led directly to a reactor trip or plant shutdown. Table 2.8 provides a breakdown of plant and equipment transients by advertent and inadvertent actuations, and according to the type of transient that occurred. For those advertent actuations due to fires, the transient is usually the direct result of the fire and not the FPS release (except in the case of the Oyster Creek event). It is interesting to note that all of the trips were caused by actuations of water-based fire protection systems. For example, several sprinkler systems were actuated through personnel error at Ginna (November 14, 1981). Water entered the control rod drive switchgear

Table 2.6

Affected Equipment
(Risk Important Equipment Marked With Asterisk)

Description	Inadvertent	Advertent	Total
Charcoal Filters, Filter Units	15		15
Transformers (Shorting)*	10	4	14
Transformers (Trip Signal)*	4		4
Control Panels, Switches, Instruments*	11	2	13
Electrical Junction Boxes*	2		2
Load Centers, Switchgear, MCCs, Busses*	10	1	11
Fire Tanks	9		9
Sensors/Detectors	3		3
HPCI Equipment, Oil*	3		3
Diesel Generator Actuators, Dampers*	3		3
ESF Pumps, Valves*	2		2
Misc. Instrumentation, Indicators, Monitors	2		2
Instrument Air Lines*	2		2
Pump Motor*	1		1
Plant Computer	1	1	2
Motor Generator Set*	1		1
Access Card Reader	1		1
Radio Repeater	1		1
Fire Door	1		1
Cable Tray Barrier Material	1		1
None Affected	51	11	62

Table 2.7

Failure Modes

Description	Inadvertent	Advertent	Total
Electrical Shorting	26		26
Wetting of Filter	15		15
Degraded FPS Operability	13		13
FPS-Generated Trip of Plant, Equipment	6		6
Water Contamination, Corrosion	5		5
Wetting of Equipment (wiring, motors, junction boxes)	5	1	6
Unknown	4		4
Water Damage	2		2
CO_2, Halon Contamination	1	1	2
CO_2, Icing	1		1
None or Not Applicable	43	13	56

cabinet causing misalignment of two control rods to the fully withdrawn position. Water also tripped a Reactor Protection System motor generator set, and operators manually tripped the reactor. At Palisades on July 14, 1987, an inadequate maintenance procedure resulted in deluge actuation over the startup transformer. Grounds due to the fire water actuated relays which tripped all three startup transformers, causing a loss of offsite power and forced a manual reactor scram. At Brown's Ferry on December 28, 1989, a deluge valve failed to reset during flushing operations causing full flow discharge onto the 500KV shunt reactors. The resulting fault caused transients on the reactor protection system, leading to a half scram and ESF actuations. Personnel error at Point Beach 2 on March 29, 1989 caused a deluge system actuation over the main transformer resulting in a main transformer lockout (spray-induced flashover), turbine/generator trip, and automatic reactor trip. The emergency diesels also started on bus undervoltage. These and other similar events have often involved complex system interactions and cascading chains of events that have seriously impacted plant operation. Whenever plant transients of this

Table 2.8

Actuations Resulting in Plant Transients

Description of Transient	Total	Fire	Water
Controlled Shutdown, Precautionary	5	2	3
Controlled, due to FPS, fire	9	3	6
Automatic Plant Scram, Trip	16	5	11
Power Transient, ESF actuations	5	1	4
TOTALS:	35	11	24

type happen, challenges to plant operability occur that will always introduce a finite probability of core damage. The details of the LER reports for these events are summarized in Volume 2 of this report.

2.4 Naval Shore Facilities Data for Halon and CO_2 Systems

An additional source of data utilized in this analysis was incidences of actuation at naval shore facilities (Ref. 2.2). Records of the Naval Safety Center were reviewed to find data relevant to inadvertent operation of FPSs in commercial nuclear power plants. Reasons for inadvertant operation at naval shore facilities resembled those of commercial nuclear power plants. Of 112 Halon System actuations and 30 CO_2 system actuations none caused any collateral damage at naval shore facilities.

2.5 Foreign Data

One of the objectives of this analysis was to collect and classify data on FPS actuations from foreign sources. Unfortunately, the data that was available during the preparation of this report was rather incomplete and of inconsistent content. Many of the events that have occurred have not been abstracted. Therefore, little useful information could be obtained for those events. Also, events that do not involve fires are not always documented in distributed reports.

The Oak Ridge Nuclear Operations Analysis Center was the primary source for what information is available. No data was obtained from Japan, and numerous European countries with operating nuclear plants (such as England) are not represented in the data. An article from the Nuclear Energy Agency of the OECD (Ref. 2.10) summarized a sampling of fire-related, FPS-related, and other events that have occurred in nuclear plants for the member countries. These event summaries, presented in

Appendices of Reference 2.10, were reviewed and an attempt was made to identify the source (plant and country) for those involving FPS actuations. All but six of the U.S. commercial reactor events reported in this paper were identifiable and are included in Appendix A of this report.

Two separate submittals from Oak Ridge provided data on 53 events in ten countries. No information on reactor operating years was provided. These events are summarized in Appendix B.1 and are summarized as follows:

a. Of the 53 events, 16 did not involve FPS actuations. Of the 16 events involving fires, five resulted in advertent FPS actuations. There were five events for which FPS actuation is indeterminate from the event descriptions. Finally, 27 events involved inadvertent FPS actuations.

b. Single actuations of water-based systems occurred in 19 of the incidents. There were four CO_2, one Halon, and five foam system actuations. One event involved actuations of five water-based systems, another involved several foam systems, and another event resulted in the actuation of one water system and one Halon system. There were five events for which the type of system that actuated is unknown.

c. Occurrences in BWRs totaled nine events while the number of incidents was 28 in PWRs. In 14 of the events, the power level was greater than 50 percent. The power level is unknown for 18 of the incidents. The plant was shutdown during three events and one event occurred in an unspecified offsite area common to the local power pool (therefore, no power level was applicable).

d. The causes of the actuations were as follows: personnel/procedural (13); unknown (10); advertent (5); mechanical ruptures/leaks of valves, piping, components (6); sprinkler head failure (1); and other (2).

e. The actuating or failed FPS components included: unknown (14); valves/hydrants (8); manual advertent actuation (3); actuation pin/switch/pushbutton (3); controller/circuits (2); detectors (2); sprinkler heads (2); and piping (2).

f. Affected locations included the diesel generator and fuel oil tank rooms (9); the turbine building (6); cable rooms/decks (4); reactor building or containment (3); control room (3); outside/transformer areas (3); and miscellaneous areas (2). The specific area affected was unknown for 12 of the events.

g. No systems or equipment were affected in seven of the events involving FPS actuations and the specific affected equipment was unknown for another five of the events. A number of important systems and/or equipment were affected in a number of the events as follows: diesel generator system components (6); pump motors (5); instrument/control panels (5); turbine generator components (2); steam generator components (2); and reactor recirculation pump or control rod drive (CRD) components (2).

h. Eight of the events resulted in a plant transient and one event was the result of a plant transient. In five of the events, it was not possible to determine whether a plant transient resulted from the occurrence. One event was both the result of, and in turn resulted in a plant transient.

Canadian data was provided in a separate submittal, summarized in Appendix B.2. There was a total of 16 inadvertent actuation events reported for a period involving 166 reactor operating years (from December, 1980 to April, 1988). Though this might imply a frequency of 0.10 events per reactor year (similar to the U.S. frequency of 0.15 events per year), it was made clear in the submittals that the data was incomplete, and likely did not include many potential unreported FPS events. These 16 events can be briefly summarized as follows:

a. Eleven of these inadvertent events involved water-based systems, of which one event resulted in three actuations, and two events involved two actuations. Two events involved CO_2 systems, Halon actuation occurred in one event, two foam systems actuated in one event, and one event involved an unknown FPS type.

b. Causes of the FPS actuations included personnel error (4); pipe rupture, water hammer, aged hose (4); welding, diesel smoke (3); unknown (2); defective sprinkler head (1); defective detector (1); condensation (1); and controller problems (1).

c. Areas affected included the diesel generator/fuel oil tank rooms (5); outside areas and transformers (4); cable rooms (2); turbine building (2); telecom room (1); and unknown areas (2).

d. In eight of the events, no systems or equipment were adversely impacted. In three of the events, the diesels or associated fuel oil tanks were affected. In two events, the operability of fire protection systems themselves were affected. In one event, the turbine vacuum pumps were flooded. In one event, the affected equipment was unknown. Finally, one event resulted in a plant trip.

Appendix B.3 provides a listing of 47 title abstracts of foreign events for which very little information is known at this time. These data represent events reported from five countries, with 36 of the events from one country. The type of plant involved in the events was also disproportionate, with 32 events occurring at BWRs and 16 at PWRs. Eight of the events involved some type of equipment damage or impact, such as wetting of pumps. Eight of the events actually resulted in plant/equipment transients or reactor scrams. Other than this brief summary, little more can be said regarding the applicability of these events to the overall foreign data base. Until further research is conducted, no quantitative conclusions can be made regarding FPS actuations based upon this data.

2.6 Summary

In summary, the operating experience with nuclear power plants in the United States has shown that, for the U.S. nuclear industry as a whole, approximately 15 fire protection system actuations occur each year. Of these, approximately one in ten are advertent. This average is based on the finding that 150 actuations, both advertent and inadvertent, occurred between January 1, 1980 and December 31, 1989. When this number is compared to the number of operating reactors in the United States, the frequency of FPS actuations is calculated to be about 0.17 events per reactor-year. The incremental frequency for BWRs is 0.19 events per reactor year and for PWRs 0.16 events per reactor year.

Of all FPS actuations, about 89 percent were inadvertent and 11 percent were fire-induced advertent actuations. About 79 percent of all individual system actuations were water-based systems, 11 percent were CO_2 systems, and the remainder (ten percent) were Halon actuations. Thirteen of the events involved multiple actuations of fire systems. There did not appear to be any recurring cause for these multiple actuations events.

Almost one-third of the events were caused by human error. These errors included misunderstandings by personnel, accidental manipulation of FPS actuating devices, and deficient procedures. Breaches or failures of fire protection system piping, fittings, valves, and other pressure boundary components were the root cause in another 19 percent of the events. Over 11 percent of the events were caused by some environmental stimulus such as steam, dust, welding, lightning or high humidity. Eight of the events occurred during or as a result of conditions from a plant transient or trip in progress.

Actuations of fire protection systems have occurred in most of the major areas in nuclear power plants, and have impacted or caused direct damage to many of the safety-related and other important plant support systems. In 30 (20 percent) of the events, the FPS actuations led to plant transients or reactor trips due to the nature of the specific failure modes and affected equipment. Significant damage and adverse affects occurred to numerous types of electrical equipment, including control panels, motor control centers, switchgear, load centers, electrical busses, junction boxes, and other electrical components. The primary failure mode was electrical shorting due to water intrusion or direct water spray.

Equipment damage and equipment reconfiguration occurred due to FPS interlocks, wetting of equipment, water contamination (of instrument air lines), and corrosion due to prior FPS actuations.

Few conclusions or insights can be made from the analysis of the foreign data. The data base is incomplete and inconsistent in content. Also, little is known regarding actual historical operating reactor experience. Much more data needs to be acquired before meaningful assessments can be made. A thorough, detailed comparison with experience in the United States alone cannot be made at this time.

2.7 References

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3.0 METHODOLOGY FOR EVALUATION OF POTENTIAL ACCIDENT SCENARIOS CAUSED BY FPS ACTUATIONS

The safety significance of actuations of FPSs is highly plant-specific, depending on such factors as the plant layout and number and types of fire protection systems. Furthermore, the significance of the actuation of any particular FPS is highly dependent on system inter-dependencies as determined by the logic models (event trees and fault trees) as well as potentially important random or test/maintenance unavailabilities. Thus, to rigorously analyze the impact of such events, the models and logic under study must be used. A methodology for accomplishing this was developed as part of this project. This methodology is based on use of a "vital area analysis" which is an important part of the fire probabilistic risk assessment methodology developed by Sandia National Laboratories for the USNRC (Ref. 3.1). The methodology can be applied to any plant for which a detailed probabilistic risk assessment (PRA) is available.

3.1 Vital Area Analysis

The basic tools of any PRA are the event trees and fault trees which describe a plant's response to any off-normal condition (initiating event) which requires a plant to be shutdown. The event tree enumerates the possible end states which result (i.e., successful shutdown, core damage, or core vulnerable) depending on the success or failure of the safety systems required to mitigate the off-normal condition.

The occurrence of a random FPS actuation or an actuation in the presence of a fire in a nuclear power plant can result in a plant transient caused either by the operator manually tripping the plant or the plant automatically tripping as a result of the actuation itself. Thus, for those actuations caused by a fire or by failures in the FPS, a general transient event tree is used to quantify the effect of FPS actuations.

By contrast, when a seismic event occurs, a loss of offsite power (LOSP) is highly likely due to failure of ceramic insulators in the switchyard. Thus, for seismically induced FPS actuations, the LOSP transient event tree is used.

As examples, Figures 3.1 and 3.2 present a general transient event tree and the LOSP transient event tree for a typical PWR nuclear power plant. Each of the (non-success) branches on this tree represent a potential accident scenario. The success or failure of the required safety systems (shown across the top) is determined by fault trees, which are downward branching trees which logically identify all possible combinations of component failures (due to any cause) which lead to the failure of the safety system in question.

These logic models are combined using Boolean algebra (as embodied, for example, in the SETS computer code, Ref. 3.2) to give an expression for each accident scenario (accident sequence) in terms of combinations of

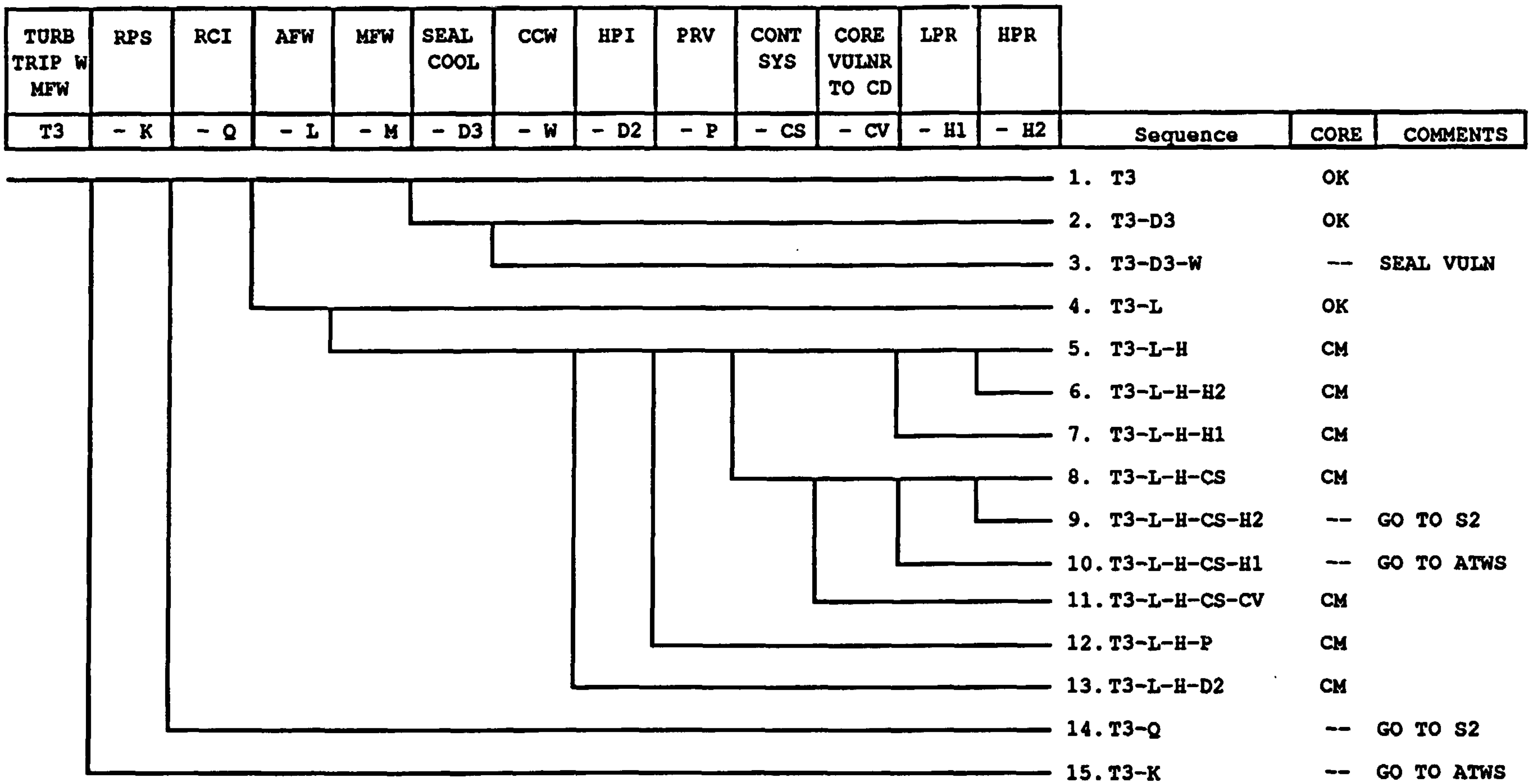

Figure 3.1. Event Tree for T_3 - Turbine Trip with MFW Initially Available

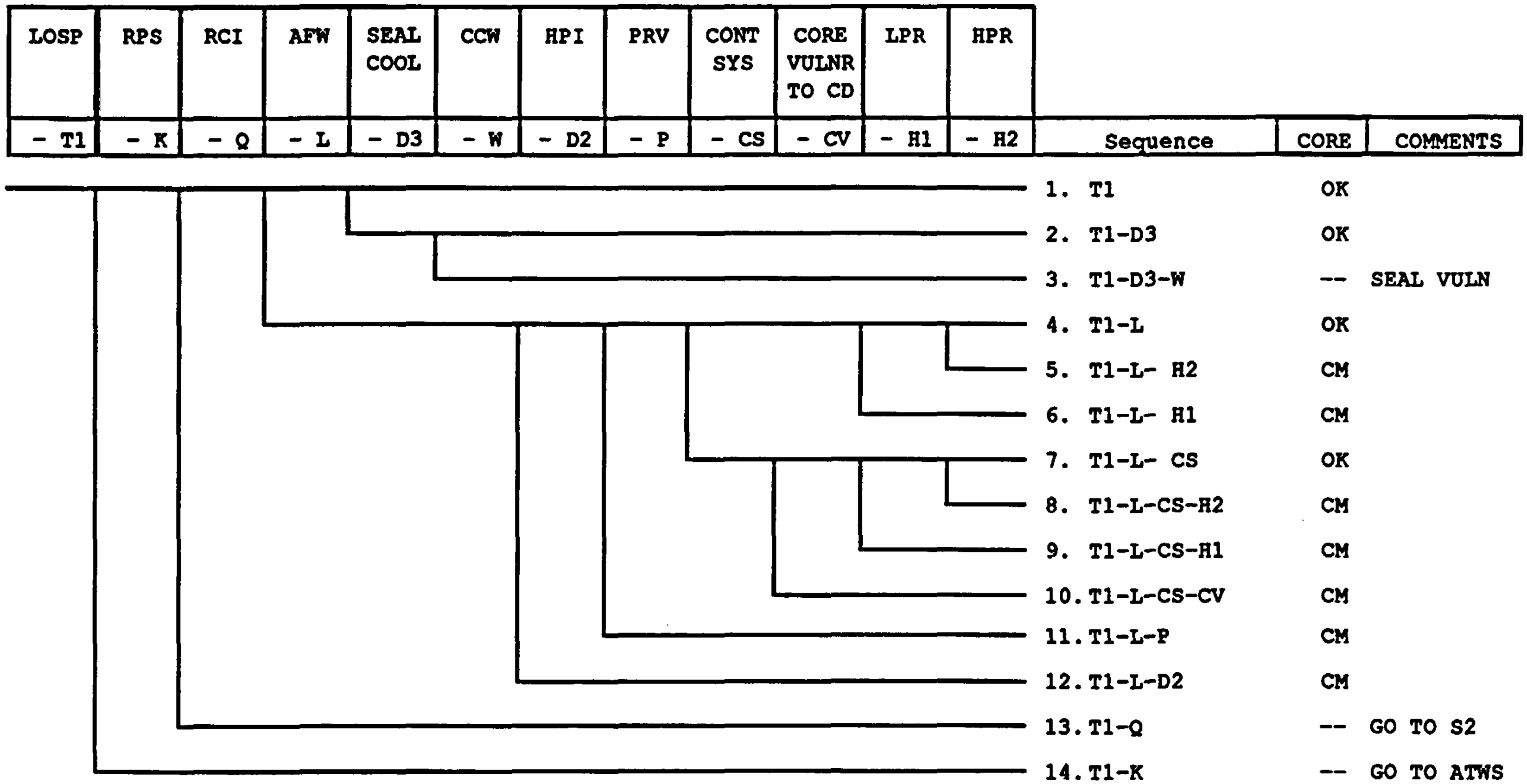

Figure 3.2. T_1 (Loss of Offsite Power) Seismic Event Tree

component failures. These combinations of basic component failures are called "cut sets." A typical accident sequence has the general form

$$Acc = C_1 + C_2*C_3 + C_4*C_5*C_6 + ...$$

where the C_i are component failures. Note that this is a Boolean logic equation in which the "+" denotes the logical union operator and the "*" denotes the logical intersection operator. It is not until later that the equation is converted into an algebraic equation for quantification. The occurrence of the failure events in any one cut set results in the occurrence of the accident sequence. Once numerical values for the basic component failure events are determined (from some data base), the frequency of the accident sequence can be found using the laws of probability to evaluate the cut sets and the union of the cut sets. The existence of correlation between the basic component failure events must also be considered in this process. The final result is a numerical value for the frequency of each accident sequence, and the frequency of core damage is determined from the sum of the accident sequence frequencies.

In a vital area analysis, the same process is used, except that the goal is to determine which areas in the plant are vital in the sense that, if some or all components in a single area or combination of areas were failed due to some cause (e.g., a fire, FPS actuation, etc.), then core damage would result. This is accomplished by mapping each susceptible component (and its associated cables) occurring on the fault trees to the area in which they reside and using the laws of Boolean logic to obtain logic expressions for the accident sequences in terms of the locations (hereafter referred to as fire zones) and random failure events. The general form for each accident sequence is

Acc = Zone 1 + Zone 2*Zone 3 + Zone 4*Random 1 + ...

where Zone 1, Zone 2, etc., are the locations and Random 1, Random 2, etc., denote random failures or test/maintenance unavailabilities. This form of the accident sequences is obtained using the Boolean mapping option in the SETS code in conjunction with tables relating each component to its location. As part of the solution process, numerical screening is performed so that only probabilistically significant cut sets are retained. The value of the numerical cutoff is specified by the analyst, and is chosen to be consistent with the remainder of the PRA for the plant in question.

This form of the accident sequence can be used to perform a quantitative assessment of the impact of an FPS actuation in any particular fire zone, including the concurrent unavailability of equipment located in other zones due to random causes. In addition, this form of the equation yields directly useful qualitative insights, since those fire zones which are single point vulnerabilities are identified directly. This can be used as a basis for reviewing the critical equipment in those zones for vulnerability to any hazards which might conceivably be postulated in that zone.

In the following sections, criteria are given for utilizing these very general accident sequence vital area equations to evaluate the potential risk arising from actuation of a plant's fire protection systems under a quite general set of root cause scenarios.

3.2 Generic FPS Actuation Scenarios

Based on the review of past experiences and walkdowns of a number of plants, thirteen generic root-cause scenarios, as shown in Table 3.1, were identified. Three root causes are due to inadvertent FPS actuations caused by a fire in another zone. Four are due to actuations resulting from purely random causes. Four are due to seismic causes, and one is due to the occurrence of a fire outside the plant. Also included is advertent FPS actuation with the presence of a fire. The various root causes of FPS actuation are described below and the specific tasks and information required to evaluate them are briefly discussed:

a. Fire-induced FPS actuation - FPS agent-induced damage. Based on the vital area analysis and plant specific data (for example data submitted in accordance with 10CFR50 Appendix R), fire zones are identified where smoke or heat spread could cause inadvertent actuation in other plant areas which are either physically adjacent or connected through ventilation paths. Estimates are made of the impact of the FPS agent on equipment in these plant areas and are applied to the appropriate cut sets and accident sequences.

b. Fire-induced FPS actuation - recovery prevention. A plant's fire PRA is reviewed for risk-significant recovery actions on equipment not damaged by fire and it is also determined in which plant areas these actions must occur. Then, other fire zones which are either physically adjacent or connected through ventilation are examined to determine if either heat or smoke spread could actuate the FPS and prevent the recovery action hypothesized. If any such combinations are found, the applicable accident sequences are requantified.

c. Fire-induced FPS actuation - access prevention. For each critical fire zone identified in the fire PRA where manual fire suppression was identified as the means of mitigating the fire, access to the fire zone is identified via plant specific data and a plant walkdown. As was the case for smoke or heat spread actuating a FPS and preventing recovery actions, a similar analysis is conducted for the delay in manual fire suppression caused by FPS actuation and the applicable accident sequences are requantified.

d. FPS actuation - human error. The vital area analysis and plant data are reviewed to determine which fire zones have an FPS that can be manually actuated. Erroneous manual actuation in any single fire zone may occur due to false detector signals or human errors of commission. For those cut sets requiring failures in more than one fire zone, the most likely scenario is that of a fire in one of the

Table 3.1

Potential Root Cause Scenarios Resulting From FPS Actuation

1. Fire in an adjacent zone causing FPS actuation
2. Fire-induced FPS actuation (due to fire in an adjacent zone) preventing random failure recovery actions
3. Fire-induced FPS actuation (due to fire in an adjacent zone) preventing access for manual fire suppression
4. FPS actuation caused by human error
5. FPS actuation caused by steam pipe break
6. FPS actuation caused by hardware failures of FPS components
7. Seismic FPS actuations resulting from dust-triggered smoke detector activation
8. FPS actuations caused by seismic relay chatter
9. FPS actuations resulting from seismic-caused mechanical failures of FPS
10. Fires external to plant causing FPS actuation
11. Fire-induced FPS actuation due to a fire in the same zone
12. Seismic/fire interaction leading to FPS diversion
13. FPS actuation due to unknown causes

areas. Frequencies for such events are obtained from the historical data base for different types of FPSs. Using these frequencies, the accident sequences are requantified.

e. FPS actuation - steam pipe break. This root cause quantifies the core damage frequency contribution from inadvertent FPS actuation caused by a high-temperature steam environment. This actuation can occur due to moisture intrusion into a FPS controller, activation of smoke/heat detector(s), or melting of fusible link heads. An estimate of steam release frequency is made and the applicable accident sequences are requantified.

f. FPS actuation - hardware failures of FPS. In this scenario, inadvertent actuation of the FPS is caused by hardware failures of the FPS itself, such as a pipe break in a wet pipe system, or a failure in an FPS control circuit. Frequencies for such events are obtained from the historical data base for different types of FPSs. Using these frequencies, the accident sequences are requantified.

g. Seismic FPS actuation - dust. Those fire zones where automatic FPS are actuated solely by smoke/particulate detectors are identified. Then dust is assumed to cause FPS actuation in the fire zone given a seismic occurrence. The additional FPS damaged equipment failures are added to the seismic sequences and these sequences are then requantified.

h. Seismic FPS actuation - relay chatter. The potential for seismically induced relay chatter is quantified based on a detailed evaluation of each FPS actuation circuit within a given plant. The additional FPS damaged equipment failures are added to the seismic sequences and these sequences are then requantified.

i. Seismic FPS actuation - mechanical failures. The potential for seismically induced mechanical failure is quantified based on a detailed evaluation and a plant walkdown of each FPS. The vital area equations can again be used directly to assess the impact of such events.

j. External fire-caused FPS actuation. Frequency of smoke intake from external fires is estimated from a combination of generic and plant-specific data. Fire zones potentially affected by smoke spread from outside ventilation are identified.

k. FPS actuation with the presence of a fire. Quantification of this scenario requires either an existing fire PRA or identification of fire sources in critical fire zones. Each fire zone with a FPS is identified to judge the effect of a fire with the simultaneous release of FPS agent on otherwise undamaged vital equipment. These failures are added to the fire sequences and requantified.

l. Seismic/fire interaction. In this scenario one or more seismically induced fires are evaluated for the probability of occurrence based on a plant walkdown and seismic fragility analysis of fire sources within the zone(s). The probability of diversion of FPS agents into zones not containing the fire(s) is made. These failures are added to the seismic sequences and requantified.

m. FPS actuation - unknown causes. In this scenario inadvertent actuation of the FPS is due to unknown causes. Frequencies for such events are obtained from the historical data base for different types of FPS. Using these frequencies, the accident sequences are requantified.

To identify the critical plant fire zones, criteria were developed for each root cause scenario which enable the analyst to determine which zones are potentially subject to each root cause of FPS actuation, given the general vital area analysis accident sequence equations. These criteria are shown in Table 3.2. This step is performed manually, and requires a review of plant systems, plant layouts, and plant specific data for the plant. This review must consider such factors as the following:

a. the presence of automatic or manual fixed FPSs,
b. physical and electrical separation of redundant trains,
c. susceptibility to seismic events,
d. propagation of combustion products (generated either inside or outside the plant) through the ventilation system,
e. possible water and steam ingress into vulnerable equipment,
f. single random actuations of FPSs,
g. multiple actuations of FPSs, and
h. type of fire detectors.

3.3 Quantification

Quantification of the frequency of these root cause scenarios requires determination of the following:

a. Frequency of fires in fire zones (Root Causes 1,2,3, and 11)
b. Frequency of human error of commission (Root Cause 4)
c. Probability of barrier failure (for smoke, heat, or steam spread) (Root Causes 1, 2, 3, 5, and 10)
d. Probability of FPS actuation damaging equipment (all root causes except Root Cause 12)
e. Probability of additional random failures, if required (all root causes)
f. Probability of non-recovery of random failures, if required (all root causes)
g. Frequency of FPS hardware failure (Root Cause 6)
h. Frequency of steam release (Root Cause 5)
i. Frequency of smoke from fires external to the plant reaching intake to plant ventilation system (Root Cause 10)
j. Frequency of FPS actuation due to unknown causes (Root Cause 13)
k. Probability of fire given a seismic occurrence (Root Cause 12)
l. Probability of LOSP given a seismic occurrence (Root Causes 7,8,9, and 12)
m. Frequency of a seismic event (Root Causes 7, 8, 9, and 12)
n. Probability of FPS agent diversion given a seismic occurrence (Root Causes 7, 8, 9, and 12)

The specific equations used to quantify each root cause scenario are given in Table 3.3. Parameter values used to perform the quantifications are described.

Table 3.2

Fire Protection System
Actuation Root Cause Scenarios

Root Cause 1: Fire-Induced FPS Actuation Due to Smoke Spread	
Event Sequence:	Fire in Room A; smoke travels and actuates fire protection system in Room B; protection system damages critical equipment in Room B.
Cut Set Criteria:	At least one fire zone having a fire protection system (manual or automatic) and smoke detectors; no more than one fire zone without FPS and smoke detectors; reasonable access for smoke to enter Room B from Room A.
Root Cause 2: Fire-Induced FPS Actuation Preventing Recovery	
Event Sequence:	Fire in Room A; smoke travels and actuates protection system in Room B; protection system prevents risk-significant recovery action from being performed in Room B.
Cut Set Criteria:	This is a cut set involving a fire zone in conjunction with one or more random failures. A recovery action (for a random failure) is in a fire zone with a fire protection system and potential connectivity to the fire zone postulated to experience a fire.
Root Cause 3: Fire-Induced FPS Actuation Preventing Fire-Fighting Access	
Event Sequence:	Fire in Room A; smoke travels and actuates FPS in Room B; protection system prevents access to Room A for manual fire fighting.
Cut Set Criteria:	A fire zone accessible through only one other fire zone having a fire protection system (manual or automatic) and smoke detectors; only one fire zone without FPS; manual fire fighting in Room A must be significant in reducing core damage frequency (CDF).
Root Cause 4: FPS Actuation Caused by Human Error	
Event Sequence:	Operator (in Control Room or locally) erroneously actuates FPS in room or rooms without fire (possibly because of a detector alarm); protection system damages critical equipment in affected fire zones.

Table 3.2 (Continued)

Fire Protection System
Actuation Root Cause Scenarios

Cut Set Criteria: All fire zones in cut set have protection systems capable of being actuated from one control panel (in the Control Room or locally).

<u>Root Cause 5: FPS Actuation Caused by Pipe Break</u>

Event Sequence: Steam release which actuates (automatically or manually) nearby fire protection system; FPS then damages nearby critical equipment.

Cut Set Criteria: Two types of cut sets are significant here. First, a cut set involving a fire zone which contains both a FPS or its controller and a steam source. Second, a zone with a FPS which is adjacent to another fire zone containing a steam source, with a potential for steam spread between the zones.

<u>Root Cause 6: FPS Actuation Caused by Random Failures in FPS</u>

Event Sequence: Failure of fire protection system component causes actuation of FPS in fire zone; FPS then damages nearby equipment.

Cut Set Criteria: All fire zones in cut set have FPS (manual or automatic); no fire zones without FPS; if more than one fire zone in cut set, possible common cause failure for multiple actuations.

<u>Root Cause 7: Dust-Triggered FPS Actuations in Seismic Events</u>

Event Sequence: Seismic event stirs up dust which actuates automatic FPS using dust-sensitive detectors; FPS then damages nearby equipment.

Cut Set Criteria: All fire zones in cut set have an automatic FPS triggered by dust-sensitive (photoelectric or ionization) detectors; no fire zones without FPS.

<u>Root Cause 8: Relay Chatter FPS Actuations in Seismic Events</u>

Event Sequence: Seismic event causes relay chatter in vulnerable FPS control circuits; FPS actuates and damages nearby equipment.

Table 3.2 (Continued)

Fire Protection System
Actuation Root Cause Scenarios

Cut Set Criteria:	All fire zones in cut set have a FPS (manual or automatic) that has relays in control circuitry; no fire zones without FPS.

Root Cause 9: FPS Actuations Due to Seismic Failures of FPS

Event Sequence:	Seismic event causes failure of FPS components; protection agent is released and damages nearby equipment.
Cut Set Criteria:	All fire zones in cut set have a FPS (manual or automatic); no fire zones without FPS.

Root Cause 10: External Plant Fires Causing FPS Actuations

Event Sequence:	Fire outside the plant generates smoke; smoke is drawn into plant ventilation system; smoke actuates detectors and FPS in rooms serviced by outside ventilation; FPS damages plant equipment.
Cut Set Criteria:	All fire zones have a FPS (manual or automatic) and smoke detectors; all fire zones receive unfiltered outside ventilation; no fire zones without FPS.

Root Cause 11: FPS Actuation with the Presence of a Fire

Event Sequence:	Fire in a zone causing actuation of a FPS; Damage to vital equipment caused either by FPS alone or the FPS and fire.
Cut Set Criteria:	All fire zones have a FPS (manual or automatic).

Root Cause 12: Seismic/fire Interaction

Event Sequence:	Seismic event causes one or more fires; protection agent is released in another plant area without fire; fire(s) and FPS agent damage nearby equipment.
Cut Set Criteria:	At least one fire zone in cut set has safety-related equipment. Another fire zone must have FPS with capability of releasing all the FPS agent from a central supply.

Table 3.2 (Concluded)

Fire Protection System
Actuation Root Cause Scenarios

Root Cause 13: Random FPS Actuation-Unknown causes	
Event Sequence:	Failure due to unknown reasons causes actuation of FPS in fire zone; FPS then damages nearby equipment.
Cut Set Criteria:	All fire zones in cut set have FPS; no fire zones without FPS; if more than one fire zone in cut set, possibly common cause failure for multiple actuations.

Table 3.3

Equations Used in Quantification of
FPS Actuation Core Damage Sequences

Definitions:

ϕ_{cd} = frequency of core damage
P(dam) = probability of FPS damaging critical equipment
P(BF) = probability of fire barrier failure (smoke, heat, or steam spread)
P(rand) = probability of other random failures that are required to lead to core damage
P(nr) = probability of non-recovery, if applicable
P(LOSP) = probability of seismically induced loss of offsite power

Root Cause 1:

ϕ_{cd} = frequency(fire in adjacent areas) * P(BF) * P(dam) * P(area ratio) * P(rand) * P(nr)

or, if the cut set contains two fire zones,

ϕ_{cd} = frequency(fire in area without FPS) * P(BF) * P(dam) * P(rand) * P(nr)

Root Cause 2:

ϕ_{cd} = frequency(fire in area in cut set) * P(BF) * P(rand) * P(nr)

Root Cause 3:

ϕ_{cd} = frequency(fire in area in cut set) * P(BF) * P(non-suppression of fire damage) * P(rand) * P(nr)

Root Cause 4:

ϕ_{cd} = frequency(human error FPS actuation) * P(dam) * P(rand) * P(nr)

or, if the cut set contains two fire zones,

ϕ_{cd} = [frequency(fire in first area) * P(human error actuation of FPS in second area) * P(damage in second area) + frequency(fire in second area) * P(human error actuation of FPS in first area) * P(damage in first area)] * P(rand) * P(nr)

Table 3.3 (Continued)

Equations Used in Quantification of
FPS Actuation Core Damage Sequences

Root Cause 5:

ϕ_{cd} = frequency(steam environment) * probability(actuation due to suppression agent intrusion into controller or steam actuation of smoke detector) * P(dam) * P(rand) * P(nr)

Root Cause 6:

ϕ_{cd} = frequency(actuation failure of any FPS) * P(dam) * P(rand) * P(nr)

Root Cause 7:

ϕ_{cd} = frequency (seismic event) * P(LOSP given seismic event) * P(FPS actuates in zone due to dust) * P(dam) * P(rand) * P(nr)

Root Cause 8:

ϕ_{cd} = frequency (seismic event) * P(LOSP given seismic event) * P(relay chatter FPS actuation) * P(dam) * P(rand) * P(nr)

Root Cause 9:

ϕ_{cd} = frequency (seismic event) * P(LOSP given seismic event) * P(FPS failures in zone) * P(dam) * P(rand) * P(nr)

Root Cause 10:

For fire areas served by unfiltered, outside air:

ϕ_{cd} =frequency (smoke intake from below)
+ frequency (smoke intake from adjacent area)
+ frequency (smoke intake from building fire)
+ frequency (smoke intake form wildland fire)
* P(dam) * P(rand) * P(nr)

Root Cause 11:

ϕ_{cd} = frequency(fire) * P(rand) * P(nr) * P(dam) * P(area/severity ratios) * P(manual non-extinguishes)

Table 3.3 (Concluded)

Equations Used in Quantification of
FPS Actuation Core Damage Sequences

Root Cause 12:

ϕ_{cd} = frequency (seismic event) * P(LOSP given seismic event) * P(seismically induced fire/fires) * P(suppression agents exhausted in non-related area) * P(dam) * P(rand) * P(nr)

Root Cause 13:

ϕ_{cd} = frequency (unknown cause for FPS actuation) * P(dam) * P(rand) * P(nr)

3.4 Generic Quantification Data

The purpose of this study is to quantify the impact on risk of inadvertent and advertent actuations of fire protection systems. In general, for such a study, values chosen for the various parameters involved should be best-estimate values based on data. In most cases, historical data were used to estimate numerical values. When little data existed best estimate probability assignment were made based on plant walkdowns and engineering judgement.

3.4.1 Fire Occurrence Frequencies

A data base of fire occurrence frequencies was developed in Reference 3.3 for a variety of typical nuclear power plant buildings (e.g., auxiliary building, turbine building, emergency switchgear rooms, diesel generator rooms, etc.) based on the history of significant fires in commercial nuclear power plants as contained in the USNRC Licensee Event Reports. These generic frequencies (per reactor-year) were used in this study. Specific point estimate frequencies are:

Auxiliary Building	6.4 E-2	per Rx-year
Turbine Building	3.2 E-2	per Rx-year
Diesel Generator Building	2.3 E-2	per Rx-year
Emergency Switchgear Room	3.0 E-3	per Rx-year
Cable Spreading Room	2.7 E-3	per Rx-year
Battery Room	3.0 E-3	per Rx-year
Control Room	4.4 E-3	per Rx-year
Reactor Building	1.8 E-2	per Rx-year

Note that for the generic buildings, it is often necessary to ratio the overall building fire occurrence frequency down to reflect the fact that fires in only a small subset of the building can cause the postulated damage. This is called "partitioning" and is based on both analyst judgement and sensitivity calculations using a fire growth computer code, if necessary.

3.4.2 Effect of FPS Actuation on Safety-Related Equipment

Very little data exist on the effects of the FPS agents on various types of equipment. The LER review described in Chapter 2 yielded the following insights:

a. 0 of 17 CO_2 reported actuations caused some safety equipment damage

b. 0 of 15 Halon reported actuations caused some safety equipment damage

c. 32 of 118 water system reported actuations caused some safety equipment damage

The naval shore facility data review (CO_2 and Halon systems) yielded the following insights:

a. 0 of 112 reported Halon system actuations caused some safety equipment damage

b. 0 of 30 reported CO_2 system actuations caused some safety equipment damage

Based on the LER data, the probability of damage to active safety-related electromechanical equipment was taken as 0.27 per exposure for water system actuations.

The median probability of damage to active electromechanical equipment from CO_2 systems based on 47 actuations was taken to be 1.5E-2 (based on both LER data and the naval shore facility data). The derivation of this probability is described in Section 3.6. Even though no CO_2 events were counted in the data as damaging safety-related components, a pre-operational CO_2 actuation, which was not included, is known to have damaged safety-related equipment (freezing and icing of relays). Also, a radio repeater was damaged while one plant was operational but not counted as safety equipment.

The median probability of damage to active electromechanical equipment from Halon system actuation, calculated from 127 actuations, was taken to be 5.4E-3 (using both LER data and the naval shore facility data). However, a recently completed EPRI report (Ref. 3.4) questions whether there are any short term damage mechanisms from Halon. In light of this report, a sensitivity study assuming no potential damage from Halon suppressant agent has been analyzed for each of the plant specific studies (Refs. 3.5, 3.6, 3.7) as well as for generic scenarios.

The median probability of damage to cables (and their associated electrical penetrations, terminal blocks, etc.) was taken to be 3.0E-3 as there was no reported damage to cabling in any of the 280 FPS actuations. As was the case for CO_2 damage probability, the derivation of this probability is described in Section 3.6.

Note that in estimating the conditional probability of failure of equipment exposed to an FPS actuation, one must take into account plant specific ventilation configurations. For example, actuation of a CO_2 system in a diesel generator room may require the room to be sealed off automatically, so that the necessary concentrations of fire suppression agent can be obtained. Without room ventilation, diesel failures due to room temperature increase (which result from diesel operation) are likely. In this case, the conditional probability of damage should be taken as 1.0, instead of using the values described above.

3.4.3 Probability of Barrier Failure

A generic probability of failure of a fire barrier (smoke, heat, or steam spread) between two fire zones was taken from the screening values for fire spread used in the NUREG-1150 fire PRAs (Ref. 3.8 and 3.9). This value was assumed to be 0.1 (believed to be a conservative value) for all fire zones. Based on plant visits for the plants examined in this study, no modifications were made to this generic probability of barrier failure.

3.4.4 Inadvertent FPS Actuation Due to Human Error of Commission

The number of operational years for each FPS type was assigned based on total plant operating years for the decade of the 1980s and information developed by Professional Loss Control, Inc. (PLC) specifically for this project on fire protection strategies at commercial nuclear power plants (refer to Appendix D). For safety-related areas only it was found that a typical plant has an average of 2.9 deluge, 3.2 wet pipe, 2.2 preaction, 2.5 CO_2, and 1.3 Halon systems. Total plant operating years of 878.5 were then multiplied by these system averages to obtain total operating years by system type. LER data exists on this root cause. Based on the PLC survey described above and LER operational data, frequencies of 1.4E-3 for CO_2, and 3.5E-3 for Halon per system year were assigned. Also, water based FPS actuation frequencies due to human error of 9.4E-3 for deluge systems, 2.8E-3 for wet pipe systems and 5.2E-4 for pre-action systems were assigned.

3.4.5 Random Failure and Human Error Values

The random failure rates and the human error probabilities have been taken from plant-specific PRA data. Random failure rates have been calculated from the operating history of component failures at the plant in question when sufficient data existed. Human error probabilities have been taken from the plant specific PRA.

Modifications to the human error probabilities should be made in the case of recovery actions which must be performed in a fire zone in which a fire is present, in which a significant amount of FPS agent is present or during a seismic event. Such recovery actions are usually denoted as "high stress actions" and procedures are available to develop modifying factors to reflect high stress situations as, for example, in Reference 3.10.

For the plants analyzed, all recovery actions were taken directly from the original internal events PRA with the addition of credit (where applicable) for (a) recovery from the remote shutdown panel given control room abandonment, and (b) cross-connection of the adjacent unit's systems given that a fire or FPS actuation has occurred in an area where a local recovery action must be performed.

3.4.6 Inadvertent FPS Actuation Due to Hardware Failure

LER data combined with the PLC survey described earlier was used to calculate a frequency for hardware failure. Since it was found that the FPS failure rate is dependent on the type of fire protection system, different frequencies were assigned for each of the different FPS types. System operating years were calculated as described in Section 3.4.4.

On a per system-year basis, the frequencies assigned were 5.5E-3 for deluge, 5.0E-3 for wet pipe, 5.2E-4 for preaction, 2.3E-3 for CO_2, and 5.3E-3 for Halon.

3.4.7 Frequency of a FPS Actuation Due to a Steam Environment

Root Cause 5 quantifies the core damage frequency contribution from inadvertent FPS actuation caused by a high-temperature steam environment. This actuation can occur due to moisture intrusion into a FPS controller, automatic FPS actuation due to activation of smoke or heat detector(s), or melting of fusible link heads.

To provide an estimate of the frequency of FPS actuation due to a high-temperature steam environment, a review was conducted of events at nuclear power plants during the time period January 1, 1980 - December 31, 1989. The source of this review was the Licensee Event Report Data Base maintained by the Nuclear Operations Analysis Center at Oak Ridge National Laboratory. A total of 972 LER abstracts were received and reviewed.

Of the 972 LERs reviewed, 67 events were choosen for further review. The categorization of the 67 events is as follows:

Type of Event	# of Events
Leaking/Failed Valve	25
Failed/Cracked Weld	14
Pipe Leak	13
Large Steam Release	10
Steam Condensation	5

Of the 67 events, 4 events led to the actuation of a FPS. One of the four events led to multiple FPS actuations. FPS actuation in three of the events was due to smoke detector actuation from the presence of steam in the same fire zone as the steam release. For one of the events, FPS actuation occurred due to steam melting the fusible link heads in the same fire zone. For this same event moisture intrusion into a FPS controller led to FPS actuation in an adjacent area.

Based on plant walkdown experience, approximately 50% of the time FPSs themselves or their actuation controllers have been found in plant areas where there is a potential for exposure to a high-temperature steam environment. It is therefore assumed based on these plant walkdowns and the PLC plant fire protection system survey that a "typical" plant will have an average of six fire protection systems protecting safety-related areas that have the potential to be actuated by steam releases. Hence, on a per system-year basis the frequency of actuation due to steam release in the same fire zone as the fire protection system is taken to be 7.6E-4, and due to moisture intrusion into an FPS controller 1.9E-4.

3.4.8 Frequency of External Fires

The dominant external mechanism for actuation of internal smoke detectors is residual smoke. Residual smoke is defined as the smoke produced from smoldering combustion and not contained in a convection column. When flaming ceases, the convection column dissipates and all subsequent smoke produced from smoldering combustion remains near the ground as residual smoke. Only a HEPA filter can remove significant fractions of residual smoke from air flow.

Free-standing fires adjacent to the power plant could originate in stored material, refuse, or vehicles. Frequencies of such fires could be estimated as the product of a generic frequency per source times the average number of sources available. Based on site visits performed for this study, walkways along the intake side of critical buildings were found to be free of potential fuels. Therefore, inadvertent FPS actuation due to a free-standing fire adjacent to a critical building was not considered for the generic plant. Individual plants must evaluate this potentiality based on the specific site layout.

Based on data in Reference 3.11, large building fires were estimated to occur at a frequency of 2.0E-4 fires/employee-year. This value was obtained from the product of 300 annual ex-urban basic industry occupancy fires per million population, an average population of seven persons for each industrial employee, and an estimate that 10% of such fires are large enough to provide a significant amount of external smoke. For a study focusing on full-power risk, the number of employees is set at 500 permanent on-site employees (including all shifts) for the generic plant. An additional factor of 0.1 is applied representing care in the removal of fire hazards by nuclear industry personnel and reflecting the presence of an onsite fire brigade. On this basis, and estimate of 1.0E-2 large building fires per reactor site per year is used in the generic plant analysis.

Large forest fires tend to occur on windy days. As well as influencing fire spread, the wind affects the drying rate of fuels and maintains the fire in level terrain. On a windy day, wind-driven, rolling leaves can quickly scatter fire over a wide area. In addition, higher wind speeds may make it unsafe for aircraft to fly surveillance flights, and aircraft suppression activities are also affected. Also, as the number of fires increases, there are fewer people available to fight each one. A ten-year study reported in Reference 3.12 found that the ratio of large fires to all fires increased dramatically as more fires started, such that 61 percent of the acreage burned occurred on only 2 percent of the total days. From the statistical point of view, the probability of a windy day, given the presence of a large wildland fire, is very nearly one.

Although surface wind determines the direction in which smoke travels, it is not a dominant factor in affecting the total land area exposed to smoke. Therefore, from the statistical point of view, the direction of the wind may not be important. If the site is surrounded by wildland, the nuclear power plant is at risk from upwind wildfires, regardless of which direction is upwind. A generic estimate for the fraction of wildland acres burned annually was based on the assumption that there is a strong influence from the extent of resources committed to prevention and fire fighting. Review of available data (Refs. 3.11 and 3.12) indicates that for wildland fires, one in 100 acres is burned annually.

The area reached by smoke can be estimated for specific conditions. For the generic case, it was estimated that, under unfavorable conditions, the smoke from a wildland fire will be sufficiently thick over an area ten times the size of the burn.

However, some of the area that could lead to wildland smoke at a nuclear power plant has been cleared for the site. A generic estimate of 30 percent was used for the fraction of the relevant area that remains as wildland, leading to an estimate of 3.0E-2 wildland fires/year that could potentially produce sufficient smoke at an intake.

A large building fire or a wildland fire requires unfavorable meteorological conditions to keep the smoke near the ground. Reference 3.13 indicates that in most parts of the country, fewer than six days per year are expected to have high meteorological potential for air pollution. Thus a factor was applied to take into account unfavorable conditions (2.0E-2, i.e. 6 days/365 days). With consideration of this factor for both the adjacent building fire and the wildland fire cases, the following frequencies are presented:

- Frequency of smoke intake from adjacent building fire = 2.0E-4/reactor-year.

- Frequency of smoke intake from wildland fire = 6.0E-4/reactor-year.

It is emphasized that the above frequencies are those used for the generic plant, and are based on the assumptions and considerations that precede them. Each plant must evaluate its own external fire source environment, layout, and meteorological environment, and then determine appropriate values for frequency of external smoke intake. For example, a plant located in a Southwestern desert area might be expected to significantly reduce the frequency of smoke intake due to wildland fire, due both to reduced combustible density on the wildland, and less frequent adverse unfavorable meterological conditions.

3.4.9 Inadvertent FPS Actuation Due to Unknown Causes

LER data combined with the PLC survey described earlier was used to calculate a frequency due to unknown causes. The number of system operating years was calculated as described in Section 3.4.4. On a per-system-year basis, the frequencies assigned were 1.3E-2 for deluge systems, 1.8E-3 for wet pipe systems, 5.2E-4 for preaction systems, 1.8E-3 for CO_2, and 8.8E-4 for Halon.

3.5 Quantification of Seismic FPS Actuations

As was discussed in Section 3.1, the seismic sequences which must be considered are those where offsite power is assumed to be lost. Once the vital area analysis has been performed for the LOSP sequences, one can quantify them in a similar fashion as was done for the random and fire-induced FPS actuation scenarios. The one significant difference is that the accident sequences so evaluated are conditional on the plant site seismic hazard curve, so that the accident sequence conditional frequencies must be integrated over the hazard curve (Ref. 3.14) using the equation

$$F(Acc) = \int P(Acc) * F_h(pga)\,dpga$$

where,

$F(Acc)$ = total frequency of accident sequence due to the weighted probability of all earthquake levels,

$P(Acc)$ = conditional accident sequence frequency, and

$F_h(pga)$ = density function for the seismic hazard curve as a function of peak ground acceleration (pga).

In computing the conditional frequencies of accident sequences resulting from FPS actuations in the fire zones in the vital area analysis equations, the root cause being considered must be taken into account.

Figure 3.3 is a simplified seismic event tree for any given fire zone found in the vital area analysis. This event tree illustrates the relationship between the four seismic root causes. The first question that is asked (given a seismically induced LOSP) is whether or not a seismically induced fire occurs in the zone under consideration. If so, then the only applicable root causes are seismic/fire interaction (Root Cause 12) or advertent actuation (Root Cause 11). Since the advertent

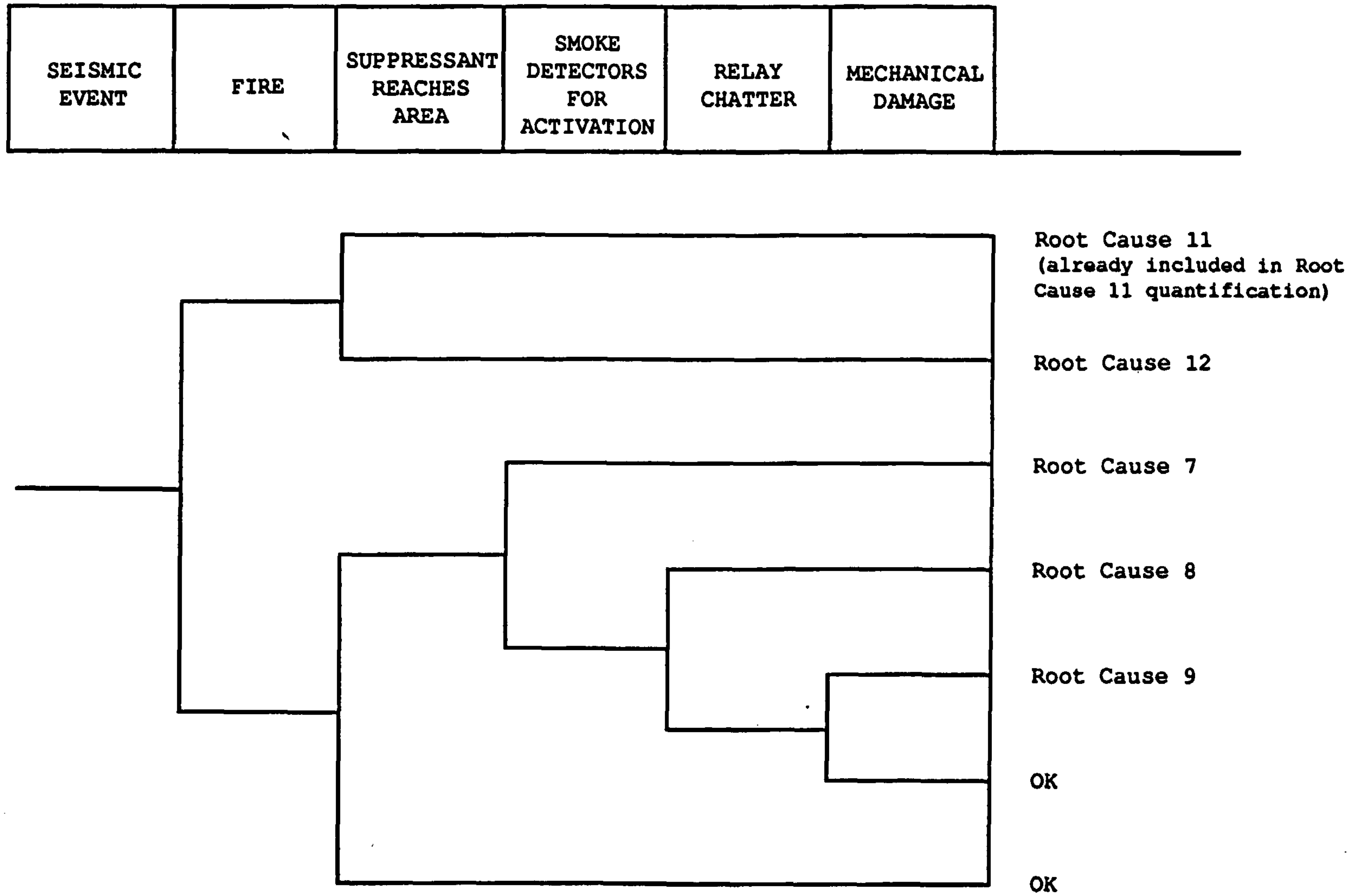

Figure 3.3. Seismic Root Cause Event Tree

actuation scenarios take into account fires from any source, this event tree branch has been quantified already in these scenarios if suppressant reaches the applicable zone. If suppressant is diverted from the applicable zone, then a seismic/fire interaction occurs.

When no seismically induced fire occurs then the only applicable root causes are actuations due to dust (Root Cause 7), relay chatter (Root Cause 8), and mechanical failures (Root Cause 9) or combinations of these root causes. One additional stipulation is that suppressant diversion cannot occur.

3.5.1 FPS Actuation by Dust Raised in a Seismic Event

This root cause is only considered applicable for those fire zones in which the FPS is automatically actuated by smoke detectors alone, and hence would be actuated by dust raised during any significant seismic event. If included, the conditional probability of inadvertent FPS actuation given a seismic event should be taken as 1.0. If more than one fire zone is being considered for this root cause, then the inadvertent actuations are considered as fully correlated, i.e., all FPSs are assumed to be actuated simultaneously in each affected fire zone.

3.5.2 FPS Actuation Due to Seismically Induced Mechanical Failures

For inadvertent FPS actuations brought about by seismic failures of the various components of the FPS systems, it is, in general, necessary to "walkdown" each FPS during a visit to the plant and identify the various components in the system. Then a system fragility function can be developed from the failure levels for the individual components themselves. Procedures for developing such seismic fragility functions are described in Reference 3.15. Since the typical components in a FPS are pipes, valves, nozzles, solenoids, electric pump motors and electrical control cabinets, it may sometimes be possible to use generic fragility functions for these components as given in Reference 3.16. This must be done with caution, however, for components in the generic data base referenced are typically seismically qualified, while seismic qualification has not been required for FPS components in the past (with the exception of water standpipes in rooms containing safety equipment). In particular, socket welded pipe joints and threaded pipe joints are not used in safety grade piping, while such joints are typically used in fire protection pipes.

For this study, plant walkdowns were performed for all three plants to develop system-specific fragilities. The results of these walkdowns are described in References 3.5, 3.6 and 3.7.

A study of fire protection system performance during the October 17, 1989 Loma Prieta earthquake was performed by EQE, Inc. and is described in Appendix C. The data collected represents approximately 100 Halon and CO_2 systems and over 1000 water sprinkler systems. This study found no instances of damage or failures of Halon and CO_2 systems. Thirteen failures of water systems were reported. The fragilities of water sprinkler systems corresponding to both inertial shaking and impact interaction failure modes was found to be 0.85 g median with a composite uncertainty of β=0.55. This study was used to modify (on a plant specific basis) the generic fragility functions from Reference 3.16.

The differences between types of FPS systems and suppression agents involved as shown by the Loma Prieta earthquake must be considered. For example, in Halon or CO_2 systems, a failure of piping which delivers the FPS agent to the fire zone would not cause inadvertent actuation inasmuch as the agent has not been released. By contrast, for certain types of sprinkler systems, the pipes are full of water at all times, and hence, failure would result in release of water on equipment below.

3.5.3 FPS Actuation Due to Seismically-Induced Relay Chatter

For relay chatter-induced FPS actuations, the fragility function for generic relay chatter from Reference 3.17 can be used if relay types are unknown. This fragility curve (as a function of local floor acceleration) has a median acceleration at failure of 4.0g (spectral acceleration in 5-10Hz range) with uncertainty parameters Beta-r = 0.48 and Beta-u = 0.75. From a plant walkdown for all three plants under consideration plant-specific values for relay chatter were developed. One type of system actuation relays found at two of the plants selected were manufactured by Potter-Brumfield. A recent EPRI report (Ref. 3.17) has developed fragilities for relay chatter for a wide range of relay types. For one type of Potter-Brumfield relay the median fragility was found to be 4.0 g with a corresponding random uncertainty of 0.48. In another type of Potter-Brumfield relays, a median fragility of 0.8g was found with a corresponding random uncertainty of 0.5. Inasmuch as relay chatter is expected to occur at relatively low seismic shaking levels and in all control circuits at the same time, it should be assumed for accident sequence evaluation that all FPS actuations are fully correlated, and occur simultaneously.

This study assumes that the circuitry is such that momentary chatter will actuate the FPS system (i.e., there are seal-in circuits involved). This assumption is reasonable from the viewpoint that earthquakes are known to cause many types of relays to chatter due to the common-cause nature of the ground shaking associated with earthquakes. This type of circuitry is typical based on the detailed analysis of four plants during this study. However, other designs are possible, and would require specific analysis.

3.5.4 Probability of Seismically Induced Fires

The following methodology has been developed for assignment of probability estimates for seismically induced fires for any fire zone under consideration. A plant walkdown is conducted which lists the potential fire sources in any given area and the seismic failure modes.

Electrical components and cabinets remaining energized on LOSP are identified. The mountings of these items are analyzed for ability to withstand seismic forces.

A cable spreading room area with vital motor control centers provides an example. If an energized motor control center fails by either tipping or sliding, there will be some probability of initiating a fire. For this study for motor control centers which have seismically failed, the probability of a subsequent fire was taken to be 0.5.

By using this methodology, plant-specific differences can be taken into account. If plant-specific vulnerabilities are noted during a walkdown, such as unanchored cabinets, appropriate modifications to fire probability can be made.

A recently completed EPRI report (Ref 3.18) surveyed earthquake-induced fires in electric power and industrial facilities. Four sites out of the 108 investigated had fire ignition following an earthquake. Therefore, on a plant-wide basis the probability of fire given a seismic event was found to be 3.7E-2. However, in the general industrial case, few electrical components remain energized after a significant seismic event, so few fires of this type would be expected. To develop zone-specific probabilities, some partitioning would be required.

3.6 Offsite Dose and Risk Calculations

This section provides the derivation of generic offsite dose calculations for a typical BWR and PWR. The MARCH (Meltdown Accident Response Characteristics) code approach (Ref. 3.19) to core damage consequence analysis is used in this study. This analytic approach includes the following considerations:

- Analysis of in-vessel thermal-hydraulic processes during the accident sequence.
- Vessel water boil-off and core support failure.
- Core debris generation including drop off and vessel melt through.
- Core debris-concrete interaction.
- Containment volume steam-condensing and heat-removal system performance.
- Containment failure modes including considerations for:
    - In-vessel steam explosion.
    - Containment leakage.
    - Hydrogen burn overpressure.
    - Ex-vessel steam pressure spiking.

- Steam and non-condensible gas overpressure.

- Base mat melt through.

Fission product release analysis includes consideration of several categories of releases (i.e. cladding rupture release, fuel melting, fission product vaporization, steam explosion/fuel oxidation, etc.) each evaluated for eight groups of radionuclides (noble gasses, molecular iodine, organic iodine, cesium-rubidium, tellurium, barium-strontium, ruthemiun, and lanthanum). Radionuclides are removed from the release by settling, deposition, spray condensing, scrubbing, filtering, etc.

There are two BWR containment functions that are important during accidents: containment overpressure protection (COP) and post accident radioactivity removal (PARR). Successful COP is defined as successful blowdown of steam from the reactor vessel to the suppression pool (or in some cases, the main condenser). Successful long-term COP requires that heat then be removed from the suppression pool via the Residual Heat Removal system. PARR also involves the suppression pool and is dependent on successful COP. If the suppression pool water inventory is maintained and cooled during a core meltdown then a large fraction of the fission products released from the core should be retained in the pool. Knowing the status of COP and PARR during a severe accident is the starting point for estimating containment failure modes and accident releases.

There are two PWR containment functions that are important during accidents: containment overpressure protection (COP) and post accident radioactivity removal (PARR). Successful COP is defined as successful blowdown of steam from the reactor vessel. Successful long-term COP requires that heat then be removed from the reactor building sump via the Low Pressure Recirculation system. PARR also involves the reactor building sump and is dependent on successful COP. If the reactor building sump water inventory is maintained and cooled during a core meltdown then a large fraction of the fission products released from the core should be retained in the pool. Knowing the status of COP and PARR during a severe accident is the starting point for estimating containment failure modes and accident releases.

Once outside the containment, the behavior of the released fission product radionuclides is analyzed with a Gaussian dispersion calculation, (using the CRAC2 (Calculations of Reactor Accident Consequences) code, Ref. 3.20) taking into account the thermodynamic properties of the source (plume buoyancy), meteorological conditions, surface roughness, resuspension, etc. The primary CRAC code result is the radiation dose in person-rem received by the population around the plant after an accident integrated out to a distance of fifty miles. While in the most catastrophic accidents this radius limitation is insufficient, for credible accidents a 50 mile radius is sufficient. Three different sets of CRAC results were calculated. The first calculation is called the "upper bound" calculation. For the second calculation, all of the release fractions except for the noble gases were uniformly reduced by a factor of seventy percent (0.3 times the upper bound). This is called the "central" calculation. For the final "lower bound" calculation, all of the upper bound release fractions except noble gases were uniformly reduced by ninety percent (0.1 times the upper bound values). These additional calculations were performed to illustrate the potential sensitivity of the results to variations in the source term are consistent with the methodology used in the WASH-1400 and USI-45 studies. This selection of source terms should not, however, be interpreted as an endorsement of any particular set. The "real" source term may be larger or smaller.

Finally, offsite human radiation dosage results are calculated considering such factors as population distribution, respiration rates, food-chain ingestion paths, sheltering and evacuation plans, radiation dose/organ conversion factors, etc.

Actual calculations have been simplified by development of plant specific factors that can be used in conjunction with core damage frequency and the accident sequence type to obtain summed offsite human dosage. Here it is important to remember that the type of accident sequence determines the containment failure mode. Thus it is possible that two accident sequences with the same core damage frequency might result in a markedly different offsite dose, because of the difference in containment performance during the different accident types (sequences).

To calculate the risk due to a given sequence the following approach is utilized.

$R_{x,s}$ = Risk resulting from Root Cause x accident sequence s in person-REM for the population within 50 mile radius of generic plant, summed over 20 years of remaining plant life.

$\phi_{cd,x,s}$ = Frequency of core damage resulting from Root Cause x from accident sequence s per reactor-year.

$CF_{m,s}$ = Containment failure probability for failure mode m applicable to sequence s.

$FPR_{c,s}$ = Fission product release category c applicable to sequence s.

For this analysis, values for $CF_{m,s}$ and $FPR_{c,s}$ have been taken from the specific plants studied and are considered typical for a PWR and BWR. In that there are literally hundreds of variables involved in their development, they are highly plant specific. However, for this analysis generic risk is based of these values, and is calculated by the following relationship:

$$R_{x,s} = \Phi_{cd,x,s} \bullet \Sigma \{CF_{m,s} \bullet FPR_{c,s}\} \bullet 2.0E+1$$

Values for $CF_{m,s}$ and $FPR_{c,s}$ for the PWR and BWR generic case can be found in Section 4 Generic Plant Analysis.

3.7 Uncertainty Analysis

Distributions on fire frequencies; random failure probabilities; barrier failure probability; operator recovery action failure probability; FPS actuation frequencies; probability of damage estimates for the various FPS agents; seismic hazard curve frequencies; and seismically induced relay chatter, mechanical failure, and fires generate uncertainties on core damage frequencies for all root causes.

The uncertainty in these parameters is propagated through the accident sequence models using two computer codes. A Latin Hypercube Sampling (LHS) algorithm is used to generate the samples for all of the parameter values (Ref. 3.21) while the Top Event Matrix Analysis Code (TEMAC) is used to quantify the uncertainty of the accident sequence equation using the parameter value samples generated by the LHS code (Ref. 3.22).

LHS is a constrained Monte Carlo technique which forces all parts of the distribution to be sampled. The LHS code is also flexible in that it can sample a variety of random variable distributions. Furthermore, parameter distributions for similar events can be correlated. For example, if two similar components (e.g., MOV-XX-FTO and MOV-YY-FTO) are modeled from the same probability distribution, then the sampling of these two distributions is perfectly correlated, meaning the same value is used for both events in a given sample member. For basic events which are modeled with very similar but slightly different distributions (e.g., MOV-XX fails to remain closed for 100 hrs and MOV-YY fails to remain closed for 200 hrs), the LHS code permits an induced correlation between the samples. However, LHS does not allow the correlation coefficient for this case to be equal to 1.0. LHS does permit sampling with a coefficient of 0.99 in these cases.

Uncertainty on fire frequency was developed consistently with NUREG-1150 internal events initiators of similar frequency, lognormal distributions were used.

Random failure probability and operator recovery action uncertainties were assigned consistently with their respective internal events values.

Uncertainties in the seismic hazard curves were taken from the LLNL and EPRI source documents. Seismically induced failures were assigned consistently with the respective values of the seismic PRAs.

Fire protection system actuation frequencies were taken to be maximum entropy distributed variables since little was known about their respective distributions. Upper and lower bounds were assigned a factor of 10 above and below their best estimate frequencies.

Probability of damage estimates for any type of suppressant for cabling and for CO_2 and Halon for any other active electromechanical component other than cabling and their respective distributions were developed as described below. In each case the data gave x = 0 out of N actuation, where x=number of times damage occurred, and n=number of actuations.

Normally, the estimated probability of damage, $\hat{p}$, is taken to be X/N. In cases of zero failures an upper confidence bound for the true probability of damage, p, can be calculated and used as a point estimate of p. γ, the confidence limit, is often taken to be 0.50 or sometimes a larger value.

For the case of x = 0, the upper bound for p has been obtained from the following equation:

$$(1-\hat{p})^N = (1-\gamma)$$

or

$$\hat{p} = 1 - \exp\,[N^{-1}\ln(1-\gamma)]$$

3.8 Summary

A methodology has been developed which can be applied to any plant for which a vital area analysis is available. If such a vital area analysis is not available, it can be developed in a fairly straightforward manner provided a systems analysis of the plant (as embodied in event trees and fault trees) has been completed and generic values for the various failure rates have been estimated. A continuing need exists for actual data on the effect of FPS agents on different types of equipment (both with and without the presence of a fire), for barrier failure rates in the presence of smoke, heat, and steam, and for identification of typical seismic vulnerabilities of fire protection system components.

3.9 References

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4.0 GENERIC PLANT ANALYSIS

4.1 Introduction

While the safety significance of fire protection system (FPS) actuation with subsequent safety-related component damage is highly plant specific, there are sufficient similarities among plants that some generic conclusions can be drawn. This chapter discusses the results of a generic plant analysis for typical FPS configurations. For this generic plant analysis, three types of fire zones will be assessed: the cable spreading room(s), the emergency electrical switchgear room(s), and the diesel generator room(s). These zones were selected for several reasons. First, they are all represented in each of the individual plants studied. (Refs. 4.1, 4.2, and 4.3). Second, in each study, these zones are all contributors to overall risk. Third, these zones are representative of equivalent zones in all the U.S. commercial power plants. More directly, all power plants have spaces that are functional equivalents to cable spreading rooms, diesel generator rooms, and emergency electrical switchgear rooms. However, it must be emphasized that there may well be, in specific plants, other fire zones that dominate risk associated with fire suppressant damage to safety-related systems. Individual plant analysis must be done to identify such zones.

Fire suppressants studied include water, Halon and CO_2. Sensors used in the actuation control systems include:

- Smoke detectors, generally the ionization type. These sensors are sensitive to dust generated in a seismic event as well as fire-generated smoke.

- Flame detectors, used primarily in diesel generator rooms and in other areas where oil fires are likely.

- IR heat detectors.

- Cable tray heat detectors, that have long wire sensors in individual cable trays.

Suppression piping systems are of two general types, pressurized and non-pressurized headers. In the generic cases presented only the wet pipe water system is of the pressurized type. Water preaction, water deluge, Halon and CO_2 are the non-pressurized type, relying on sensors and the control system to detect the need for suppressant and subsequently actuate pressurization valves. In the piping system, nozzles that dispense the suppressant are of the open or closed type. Open nozzles dispense suppressant as soon as the piping system is pressurized. Closed nozzles release suppressant, if the system is pressurized, when the desired temperature is reached. Table 4.1 summarizes the elements of the various types of systems analyzed in the generic cases.

Table 4.1

Fire Protection System Summary

System Type	Normally Pressurized	Sensors Utilized	Nozzle Type
H_2O Preaction	No	Yes	Fusible
H_2O Wetpipe	Yes	No	Fusible
H_2O Deluge	No	Yes	Open
Halon	No	Yes	Open
CO_2	No	Yes	Open

Within each zone that is examined, use of the various types of fire protection, systems and agents will be evaluated. The selection of the suppressants to be analyzed is based on the data from plant surveys. These data are found in Appendix D to this report. As each case is presented, the number of FPS type and fire zone installations that are found in U.S. commercial nuclear power plants will be given. In some cases the sum of the installations stated is greater than the number of existing nuclear power plants. This is because in many plants there may be more than one zone to be counted. For example, a plant with two diesel generator rooms protected by a preaction water FPS systems would be counted as two installations of this type.

4.2 Scope of Analysis

Different strategies have been employed in U.S. commercial nuclear power plants for critical fire zones as to the type of suppressant agent and fire protection system actuation scheme utilized. Table 4.2 lists the cases that will be analyzed in this chapter. These combinations represent the majority of the combinations found in the plant survey data of Appendix D.

In the following sections, a standard notation will be used. This notation is:

$\Phi_{cd,x}$ = Frequency of core damage resulting from Root Cause x per reactor-year.

$\kappa_{a,b}$ = FPS actuation frequency for system a due to cause b.

λ_y = Fire frequency in fire zone y.

Table 4.2

Fire Protection Cases for Analysis

Suppression Agent	System Type	Cable Spreading Room	Diesel Generator Rooms	Emergency Switchgear Rooms
Water	Preaction	X	X	-
Water	Wetpipe	X	X	X
Water	Deluge	X	X	X
Halon		X	-	X
CO_2		X	X	X

X indicates analysis will be preformed

λ_{ext} = Fire frequency for fires external to the plant resulting in ingestion of smoke in the plant ventilation system.

δ_y = Steam leak frequency in fire zone y.

$P_{d,z,cb}$ = Probability of damage to cables due to agent z.

$P_{d,z,cp}$ = Probability of damage to active electromechanical components due to agent z.

$P_{d\ control,s}$ = Probability of FPS controller actuation due to steam leak.

Q = Non-recovery probability.

$P_{rand,cc}$ = Probability of random failure of safety-related component(s) or cables cc.

$P_{quake}(g)$ = Probability of occurrence of earthquake with peak ground acceleration g.

LOSP(g) = Probability of loss of offsite power in seismic event of g peak ground acceleration.

$P_{rand,DG}(g)$ = Sum of random diesel failure probability plus diesel seismic failure probability.

$R_{chat}(g)$ = Probability of seismically induced FPS control relay chatter.

$P_{man\ non\text{-}ext}$ = Probability fire is not extinguished manually prior to FPS actuation.

$P_{sys\ breach}(g)$ = Probability of seismically induced FPS breach inside fire zone of interest.

$P_{fire}(g)$ = Probability of seismically induced fire inside fire zone of interest.

$P_{FPS\ div}(g)$ = Probability of seismically induced FPS agent diversion outside fire zone of interest.

A_r = Area ratio of FPS area where suppressant agent if released could contact vital components/total area of FPS within the zone.

Γ = Factor accounting for need for multiple releases or a very large release of suppressant in a large space to damage sufficient number of cables and components.

M = Probability of unavailability the of Power Conversion System given a general transient.

The conditions and assumptions used in the generic scenario analysis are:

- The earthquake hazard was chosen to be that of the median United States East Coast nuclear power plant based on the earthquake analysis conducted by the Lawrence Livermore National Laboratory (LLNL). Sensitivity studies follow showing impact of using EPRI seismic hazard curves.

- Seismic fragility of plant fire protection system control relays [R_{chat} (g)] have a median capacity A_m = 1.5g spectral acceleration (Sa) and Beta = 0.75 based on data collected in the specific plant analyses.

- Mean recovery time for diesel generators following seismically induced CO_2 FPS initiation is set at 8 hours; for seismically induced water FPS initiation, the diesels are not recovered, based on engineering judgement of the expected level of damage imposed on wetted engine.

- Diesel generator rooms having wet pipe or preaction water fire protection systems are partitioned such that for a single seismically induced break, the likelihood of wetting the critical components which can fail the diesel generators is set at 25 percent.

- The fragility of wet pipe and preaction water fire protection system piping [$P_{sys\ breech}$ (g)] built to NFPA-13 standards has a median capability of A_m = 1.1g peak ground acceleration (pga) and Beta = 0.63 based on data taken from fire protection system failures in the Loma Prieta earthquake (extrapolated to nuclear power plant structures) as discussed in Appendix C. Note that in an earthquake, damage of sprinkler heads due to impact with building structural and architectural features is one of the principle damage mechanisms. Current NFPA-13 requirements do not require branch restraint or field review to preclude this kind of damage. It is recognized that there is not a requirement for water agent fire protection system piping to be seismically qualified. However in the plants examined it was found that for some critical safety areas, seismic pipe motion restraints were installed. For the generic case, only the requirements of NFPA-13 are considered.

- A data base of fire occurrences was developed in Reference 4.4 for a number of typical nuclear power plant fire zones. Those applicable to the generic studies in this Chapter are presented in Table 4.3.

- There are several combinations of root causes where the FPS is actuated by seismically raised dust (Root Cause 7), seismically actuated control relays (chatter)(Root Cause 8), and seismically induced breaches in the FPS system (Root Cause 9). The combinations result from the differences in FPS systems, and how they react in a seismic event. As some examples:

  a. Root Cause 7 alone is considered in the generic analysis. There are cases installations where CO_2 systems were actuated by smoke detectors alone in the fire zones of interest.

  b. Root Cause 8 alone can occur in a water deluge, Halon, or CO_2 system, where relay chatter actuates the FPS and suppressant is discharged.

  c. Root Cause 9 alone is never considered in systems that are not normally pressurized, thus Root Cause 9 can occur only in wet pipe water systems. All other systems studied are not normally pressurized.

  d. Root Causes 7 and 8 could act in conjunction in Halon, CO_2, or water deluge systems that were cross zoned so that dust actuates the smoke detector and relay chatter actuates the other sensor.

  e. Root Causes 7 and 9 can act in conjunction in a case where smoke detectors initiate the pressurization of a preaction water system, and a seismic breach occurs in the system as well.

  f. Root Causes 8 and 9 can act in conjunction where relay chatter results in a preaction system actuation pressurizing the piping, and a seismic breach occurs in the system as well.

Table 4.3

Fire Occurrence Frequencies

Fire Zone	Frequency/Reactor-Year
Cable Spreading Room	2.7E-3
Diesel Generator Building	2.3E-2
Emergency Switchgear Room	3.0E-3

g. Root Causes 7, 8, and 9 could all occur in conjunction where a cross-zoned preaction system was actuated by a combination of dust triggering the smoke detectors, relay chatter actuating the other sensor, and finally a seismic breach occurs in the system as well.

- Frequency of fires external to the plant is discussed in detail in Section 3.4.8. In the generic case, it is assumed that the fire source is an adjacent building, and the frequency is set at 2.0E-4/reactor-year

- As discussed in detail in Section 3.4.7, analysis of Root Cause 5 scenarios data provides the basis for setting values associated with FPS actuation as a result of steam or hot water leaks. The values are set at steam release in the effected zone causing actuation of the FPS of 7.6E-4/system-year, and steam release in an adjacent zone resulting in actuation of the FPS of 1.9E-4/system-year (based on LER data given in Appendix A).

- Non-recovery probability for cable spreading room scenarios is set at 4.6E-2/occurrence for non-seismic sequences and at 4.6E-1/occurrence for seismic sequences. Seismic sequences are not a function of 'g' due to a lack of data collected in this area. The increase for seismic sequences is based on the expected degradation of operator and equipment performance following a major seismic event. These values are standard based on plant specific evaluations and on internal events analyses (Ref. 4.5).

- Based on LER data, the probability of water suppressant agent damage to active electromechanical equipment is set at 2.7E-1 per exposure.

- Based on LER and naval shore facility data (Ref. 4.6), the probability of CO_2 suppressant agent damage to active electromechanical equipment is set at 1.5E-2 per exposure.

- It is recognized that short-term Halon-induced damage to safety related cables and components is arguable, and little data was found to serve as a basis for Halon damage probabilities. Recognizing this issue, a sensitivity study will follow that examines the case where short term Halon conditional probability of damage to cables and active electromechanical components is set to zero. However, for the generic analysis, our best estimate based on available data, is that the mean conditional probability of Halon fire suppressant damage to electromechanical equipment is 5.4E-3 per exposure.

- Mean probability for fire suppressant damage (all suppressants) to electrical cables, connectors, or terminal boxes is 5.0E-3/exposure based on LER and Navy data.

- In seismic/fire interaction scenarios, a fire is started in the effected zone when an electrical or electronics cabinet that remains energized after a LOSP, tips over or slides during a seismic event and the energized cables spark and start a fire [P_{fire} (g)]. Analysis of such cabinets in the plants examined showed that a median capacity of lightly anchored cabinets was A_m = 2.0g Sa and Beta = 0.06. This is recognized as a value that is highly plant specific. In some cases seismic/fire sources may be too distant from safety-related components and cables to pose a threat to them. Accordingly, use of a fire and heat spreading computer code such as COMPBRN (Refs. 4.7 and 4.8) is required to analyze this root cause for specific plants and fire zones. Given a tipping or sliding event, a fire is assumed to occur 50 percent of the time. This conditional probability may be conservative. A sensitivity study will be performed that assigns a conditional fire occurrence probability at 10 percent of the time.

- Fragility data for seismic suppressant diversion is based on individual plant specific analysis [$P_{FPS\ div}$ (g)]. It must be noted that while diversion is a concern in the assessment of Root Cause 12, it acts as an advantage in the analysis of Root Causes 7/8/9, because if suppressant is diverted away from a zone of interest, it cannot cause damage to safety-related components and cables. Probability of diversion has been considered as follows:

  a. For diversion of water, seismically caused diversion at the source was identified. The fragility of the diesel driven fire pump starting battery system was found to be 0.3 pga and Beta = 0.6. In this case, a specific plant installation was modelled. Loss of the diesel pump means loss of pressure on the fire main. The electric pump (powered off a non-vital bus) is also lost due to a LOSP.

  b. For diversion of Halon, it was found that Halon bottles are restrained only by single strap. The fragility for the failure of these straps which would result in the diversion of Halon has a median capacity of A_m = 0.8g pga and Beta = 0.6.

c. For diversion of CO_2, it was found that actuating power (for the tank outlet valve) from dedicated DC power supplies and batteries was the same as in subparagraph "a." above, and the slipping of the unanchored CO_2 tanks and severing the outlet lines had a median capability of A_m = 0.41g pga and Beta = 0.5.

- A LOSP is highly likely due to the low capacity or ceramic insulators found in the electrical switchyard at most nuclear power plants. For the generic plant analysis, the probability of LOSP given a seismic event is 0.0 up to the Operating Basis Earthquake (OBE) and 1.0E+0 above the OBE, which in the generic analysis was chosen to be 0.05g. Earthquakes below the OBE are not considered seismic initiators and therefore do not result in LOSP. Earthquakes levels between the OBE and SSE are assumed to result in LOSP in this study, which is conservative, but since LOSP is given, all seismically induced accident sequences not involving LOSP are ignored.

- In a seismic event, dust is stirred up and distributed through the building. Smoke detectors are sensitive to dust, and actuate when dust is present in the air (Ref. 4.9). The probability of actuation of smoke detectors in a seismic event is set at 1.0/event.

- For the cable spreading room, the Γ area factor is set at 1.0E-1 based on engineering judgement, plant walkdowns and typical equipment layout configurations.

- Non-seismic fire protection system actuation frequencies (per system-year) are based on the LER data in Appendix A, and are provided in Table 4.4.

Thirteen generic cases will be examined on the basis of these assumptions and conditions. They represent those cases presented in Table 4.2. Details of calculations are presented in Appendix E. Calculations for core damage frequency, risk, and sensitivity studies are accomplished with the use of the Top Event Matrix Analysis Code (TEMAC) (Ref. 4.10) and the Latin Hypercube Sampling Code (Ref. 4.11).

4.3 Cable Spreading Room

In the following analysis of the cable spreading room, it is assumed that there are enough emergency components or component cables in the zone such that if these cables or components are damaged, the damage is sufficient to cause an accident sequence, i.e. in the vital area analysis, the cable spreading room fire zone is a single cut set. While there are some plant configurations that use dual or multiple cable spreading rooms, this analysis will be done on a plant configuration using a single cable spreading room. This approach is considered to be more realistic for plants with a single room, since the initiation of an accident sequence through cable or component damage from FPS suppressant agent simultaneously in multiple cable spreading rooms is intuitively

Table 4.4

FPS Actuation Frequency per System-Year

	Human Error	Hardware Failure	Other	Total
Water Preaction	5.2E-4	5.2E-4	5.2E-4	1.6E-3
Water Wet Pipe	2.8E-3	5.0E-3	1.8E-3	9.6E-3
Water Deluge	9.4E-3	5.5E-3	1.3E-2	2.8E-2
Halon	3.5E-3	5.3E-3	8.8E-4	9.7E-4
CO_2	1.4E-3	2.3E-3	1.8E-3	5.5E-3

less likely than doing so in a single cable spreading room. The Generic Issue 57 core damage frequency associated with the cable spreading room(s) in a plant with multiple cable spreading rooms will be less than or equal to that in a plant with a single cable spreading room.

In the case of the generic cable spreading room, some assumptions are made, based on the findings of the studies of the individual plants. It is assumed that the cable spreading room is not impacted by events in adjacent zones. This means that: fires in an adjacent zone do not cause cable spreading room FPS actuation, adjacent zone fire induced FPS actuation does not prevent random failure recovery action, and adjacent zone fire induced FPS actuation does not prevent access for manual fire suppression. Accordingly, Root Causes 1, 2, and 3 are eliminated from consideration in the generic cable spreading room. While these root causes were eliminated from consideration in the generic case, examples of installations where these root causes must be considered include: (a) A zone protected by a deluge/Halon/CO_2 system triggered by smoke detectors, where ventilation flow is such that smoke from another space can migrate and initiate the effected zones detector's, with cable or component damage as the result (Root Cause 1); (b) a zone protected by a deluge/Halon/CO_2 system triggered by smoke detectors, where ventilation flow can migrate and initiate the affected zone's detectors, with the suppressant interfering with recovery action in the affected zone (Root Cause 2); (c) and finally, a zone protected by a deluge/Halon/CO_2 system triggered by smoke detectors, where ventilation flow can migrate and initiate the effected zone's detectors, with the suppressant interfering with fire fighting team access to the zone where the smoke originated.

4.3.1 Cable Spreading Room with a Preaction Water FPS Case

4.3.1.1 Non-Seismic Initiating Events

Non-seismic root causes of interest in the generic cable spreading room are limited to 4, 6, and 13. Water in a preaction sprinkler system is used as a cable spreading room and other cable area suppression agent in

35 commercial nuclear power plant applications. The following relationship represents the core damage frequency relationship for releases of water in preaction water systems:

$$\Phi_{cd,x} = \kappa_{Wp,b} \bullet P_{d,w,cb} \bullet Q \bullet \Gamma$$

In the case of a preaction sprinkler system, a single fire protection system piping failure will cause wetting of only a small portion of the cables or components in the generic cable spreading room. To wet and subsequently damage a sufficient number of cables to initiate an accident sequence, multiple fire protection system suppressant releases, or a very large suppressant release may well be required. To reflect this condition, in calculations of core damage frequency the Γ factor of 1.0E-1 is included. For Root Causes 4,6, and 13:

$$\Phi_{cd,4} = <1.0E-8$$

$$\Phi_{cd,6} = <1.0E-8$$

$$\Phi_{cd,13} = <1.0E-8$$

4.3.1.2 Seismic Initiating Events

Seismic root causes of interest are combinations of Root Causes 8/9 and Root Cause 12. Root Cause 7 is not considered for the generic case since while seismically generated dust is sufficient to actuate the FPS system, suppressant is not released.

The Root Causes 8/9 scenario occurs when seismically induced relay chatter actuates the preaction FPS, and seismic pipe breach occurs, releasing suppressant to cause damage. In seismic events, manual non-recovery probabilities are increased to their upper bound because of the disruptive conditions in the plant following the initiating earthquake. Breach is a consideration in preaction systems, the Loma Prieta earthquake data is used in this calculation. For the generic cable spreading room the seismic Root Causes 8/9 core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet P_{d,w,cb} \bullet \int \{P_{quake}(g) \bullet LOSP(g) \bullet R_{chat}(g) \bullet P_{sys\ breach}(g) \bullet P_{FPS\ non\text{-}div}(g)\}\, dg$$

and in the generic case:

$$\Phi F_{cd,8/9} = <1.0E-8$$

For the seismic Root Cause 12 scenario, a seismic event causes a diversion of FPS agent from a central supply, a seismically induced fire occurs and is subsequently not suppressed. An example is the case where the emergency diesel driven fire pump starting batteries are not seismically mounted, so that in an earthquake the diesel pump fails to start. Also, the electric fire pump is failed due to LOSP (its power source is a non-vital bus). Additionally for this sequence to occur, a fire source such as an electrical cabinet in close proximity to vital cables, and that remains energized on LOSP must initiate the fire damage. Three of the examined plant cable spreading rooms contained seismic/fire sources, one did not. Individual plants can inspect to confirm Root Cause 12 applicability. Core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet \int P_{quake}(g)) \bullet LOSP(g) \bullet P_{fire}(g) \bullet P_{FPS\ div}(g)\ dg$$

and in the generic case:

$$\Phi_{cd,12} = 1.2E-5$$

4.3.2 Cable Spreading Room with a Wetpipe Water FPS Case

4.3.2.1 Non-Seismic Initiating Events

Water in a wet pipe sprinkler system is used as a cable spreading room and other cable area suppression agent in 69 commercial nuclear power plant applications. The following relationship represents the core damage frequency for releases of water in wet pipe water systems:

$$\Phi_{cd,x} = \kappa_{Ww,b} \bullet P_{d,w,cb} \bullet Q \bullet \Gamma$$

In the case of a wet pipe sprinkler system, a single fire protection system piping failure will cause wetting of only a small portion of the cables or safety-related components in the generic cable spreading room. To wet and subsequently damage a sufficient number of cables or components to initiate an accident sequence, multiple fire protection system releases may well be required. To reflect this condition in calculations of core damage frequency, a Γ factor of 1.0E-1 in included. For Root Causes 4,6, and 13:

$$\Phi_{cd,4} = 4.7E-8$$

$$\Phi_{cd,6} = 8.4E-8$$

$$\Phi_{cd,13} = 2.9E-8$$

4.3.2.2 Seismic Initiating Events

Seismic root causes of interest are combinations of Root Causes 8/9, and Root Cause 12. For the seismic Root Causes 8/9, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet P_{d,w,cb} \int P_{quake}(g) \bullet LOSP(g) \bullet R_{chat}(g) \bullet P_{sys\ breach}(g) \bullet$$
$$P_{FPS\ non\text{-}div}(g)\ dg$$

and in the generic case:

$$\Phi_{cd,9} = 5.3E-8$$

For seismic Root Cause 12, core damage frequency is governed by the following relationship:

$$\Phi_{cd,12} = Q \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet P_{fire}(g) \bullet P_{FPS\ div}(g)\ dg$$

and in the generic case:

$$\Phi_{cd,12} = 1.2E-5$$

4.3.3 Cable Spreading Room with a Deluge Water FPS Case

4.3.3.1 Non-Seismic Initiating Events

Water in a deluge sprinkler system is used as a cable spreading room and other cable area suppression agent in 44 commercial nuclear power plant applications. The following relationship represents the core damage frequency relationship for releases of water in deluge water systems:

$$\Phi_{cd,x} = \kappa_{Wd,b} \bullet P_{d,w,cb} \bullet Q$$

and for Root Causes 4,6, and 13:

$$\Phi_{cd,4} = 1.6E-6$$

$$\Phi_{cd,6} = 9.1E-7$$

$$\Phi_{cd,13} = 2.3E-6$$

In a cable spreading room configuration where the actuation controls for the deluge water FPS are located in the turbine building, or in another plant area where steam/feed/condensate piping is located, there is the potential for a steam or hot water leak to damage the controller in such a way as to cause actuation. There is an example of this scenario in the LER data in Appendix A. For the generic cable spreading room, Root Cause 5 core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = \delta_y \bullet P_{d\ control,s} \bullet P_{d,w,cb} \bullet Q$$

For Root Cause 5:

$$\Phi_{cd,5} = <1.0E-8$$

For generic cable spreading rooms equipped with deluge systems, if a fire in the zone occurs and initiates the FPS, water suppressant will be sprayed throughout the zone. In preaction or wet pipe systems, the suppressant will spray only in the area of the fire. Accordingly, in the case of deluge systems consideration must be given to damage as a result of Root Cause 11. The Root Cause 11 scenario is that a fire occurs in the zone, and the FPS is actuated prior to manual fire suppression. The FPS suppressant then causes damage to vital cables or safety-related components. The following relationship represents the core damage frequency relationship for release of water in deluge systems:

$$\Phi_{cd,x} = \lambda_{csr} \bullet P_{d,w,cb} \bullet Q \bullet P_{man\ non\text{-}ext}$$

and for Root Cause 11:

$$\Phi_{cd,11} = 4.4E\text{-}7$$

4.3.3.2 Seismic Initiating Events

Seismic root causes of interest are Root Cause 7, Root Cause 8, and Root Cause 12. Root Cause 7 is included for this case since some deluge systems may be initiated by signals from smoke detectors alone. For these root causes, manual non-recovery probabilities are increased to their upper bound because of the disruptive conditions in the plant following the initiating earthquake.

For the seismic Root Cause 7, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet P_{d,w,cb}\ \kappa_{Wd,dust} \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet P_{FPS\ non\text{-}div}(g)\,dg$$

and in the generic case:

$$\Phi_{cd,7} = 2.6E\text{-}5$$

For the seismic Root Cause 8, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet P_{d,w,cb} \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet R_{chat}(g) \bullet P_{FPS\ non\text{-}div}(g)\,dg$$

and in the generic case:

$$\Phi_{cd,8} = 1.1E\text{-}6$$

For seismic Root Cause 12, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet P_{fire}(g) \bullet P_{FPS\ div}(g)\ dg$$

and in the generic case:

$$\Phi_{cd,12} = 1.2E\text{-}5$$

4.3.4 Cable Spreading Room with a Halon FPS Case

In this analysis it must again be pointed out that the short term effects of Halon in causing damage to safety-related cables and components is arguable. There was little data found to serve as a basis for quantification of damage estimates.

4.3.4.1 Non-Seismic Initiating Events

Halon systems are used as a cable spreading room suppression agent in 24 commercial nuclear power plant applications. The following relationship represents the core damage frequency relationship for releases of suppressant in Halon systems:

$$\Phi_{cd,x} = \kappa_{H,b} \bullet P_{d,H,cb} \bullet Q$$

and for Root Causes 4,6, and 13:

$$\Phi_{cd,4} = 6.0E-7$$

$$\phi_{cd,6} = 8.8E-7$$

$$\Phi_{cd,13} = 1.4E-7$$

In a cable spreading room configuration where the actuation controls for the Halon FPS are located in the turbine building, or other plant areas containing steam, feed, or condensate piping, there is the potential for a steam or hot water leak to damage the controller in such a way as to cause actuation. For the generic cable spreading room, Root Cause 5 core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = \delta_y \bullet P_{d\ control,s} \bullet P_{d,H,cb} \bullet Q$$

and in the generic case:

$$\Phi_{cd,5} = 1.7E-8$$

4.3.4.2 Seismic Initiating Events

Seismic root causes of interest are combinations of Root Causes 7/8 and Root Cause 12. In seismic event analysis, manual non-recovery probabilities are increased to their upper bound because of the disruptive conditions in the plant following the initiating earthquake. For the seismic Root Causes 7/8, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet P_{d,H,cb} \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet R_{chat}(g) \bullet P_{FPS\ non\text{-}div}(g)dg$$

and in the generic case:

$$\Phi_{cd,7/8} = 1.3E-6$$

For seismic Root Cause 12, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet P_{fire}(g) \bullet P_{FPS\ div}(g)\ dg$$

and in the generic case:

$$\Phi_{cd,12} = 3.2E-6$$

4.3.5 Cable Spreading Room with a CO_2 FPS Case

4.3.5.1 Non-Seismic Initiating Events

CO_2 systems are used as a cable spreading room suppression agent in 39 commercial nuclear power plant applications. The following relationship represents the core damage frequency for releases of suppressant in CO_2 systems:

$$\Phi_{cd,x} = \kappa CO_{2,b} \bullet P_{d,CO_{2},cb} \bullet Q$$

and for Root Causes 4,6, and 13:

$$\Phi_{cd,4} = 2.3E-7$$

$$\Phi_{cd,6} = 3.9E-7$$

$$\Phi_{cd,13} = 3.0E-7$$

In a cable spreading room configuration where the actuation controls for the CO_2 FPS are located in the turbine building, or other plant areas containing steam, feed or condensate piping, there is the potential for a steam or hot water leak to damage the controller in such a way as to cause actuation. For the generic cable spreading room, Root Cause 5 core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = \delta_y \bullet P_{d\ control,s} \bullet P_{d,CO_{2},cb} \bullet Q$$

and in the generic case:

$$\Phi_{cd,5} = 1.8E-8$$

In one of the specific plants analyzed as a part of this study (Ref. 4.3), there was some probability of core damage that resulted from Root Cause 10. In this case, the scenario was a fire occurs in an adjacent building, smoke was drawn into the cable spreading room via the ventilation system, and the smoke initiated the CO_2 FPS system. Since there was only a single case for this scenario identified, the configuration was not considered generic. There are very few FPSs installed where smoke alone is sufficient to actuate the system.

4.3.5.2 Seismic Initiating Events

Seismic root causes of interest are combinations of Root Causes 7/8 and Root Cause 12. For these root causes, manual non-recovery probabilities are increased to their upper bound because of the disruptive conditions in the plant following the initiating earthquake. For the seismic Root Causes 7/8, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet P_{d,CO_{2},cb} \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet R_{chat}(g) \bullet P_{FPS\ non\text{-}div}(g)\,dg$$

and in the generic case:

$$\Phi_{cd,7/8} = 1.2E-6$$

For seismic Root Cause 12, suppressant diversion is an issue of concern. An example of diversion is the case where the CO_2 discharge valve actuation power supply has a low seismic fragility, and in a seismic event fails to pressurize the suppression system even on the receipt of actuation signals. Core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet P_{fire}(g) \bullet P_{FPS\ div}(g)\ dg$$

and in the generic case:

$$\Phi_{cd,12} = 9.1E-6$$

4.4 <u>Diesel Generator Rooms</u>

In the analysis of the generic diesel generator room fire zone(s), only Root Causes 7,8,9 and 12 are of concern. In the event of the LOSP, the loss of the diesel generators is a station blackout consideration. For the diesel generators even to be demanded, LOSP must in all likelihood be seismically induced because of the following reasoning. Random LOSP events have a frequency of about 8.0E-2/year. When this frequency is ratioed to a 24-hour period or less when random FPS actuations also occur, then the likelihood of the combined events is below the truncation point for consideration. Therefore, risk due to FPS induced damage to safety-related components in the diesel generator rooms starts with LOSP as an initiator. The seismic event provides this initiation. Note that damage to cables is not a consideration in the diesel generator room, of concern are components such as the engine control panel and the excitation cabinet, whose failure will result in loss of the generator.

The model for the generic diesel generator room configuration is an arrangement of two independent diesel generator rooms for generation of emergency power. The rooms have identical arrangements and fire protection systems, however the systems operate independently. Recent analysis has been conducted on the effects of fire fighting water impingement on the operation of emergency diesel generators (Ref 4.12). This analysis assessed the following damage mechanisms and developed levels of concern for each mechanism:

- Localized cooling on the engine block or on the exhaust manifold could result in warping or cracking - not considered likely.

- Ingestion of very large quantities of water could result in engine shutdown. For this to occur the engine would have to take its air from the space being sprayed by the FPS. This is not considered likely.

- The generator could become grounded with a subsequent electrical failure. This is not considered likely.

- The excitation system, particularly rotating exciter designs, often has armature housings and brushes open to the room atmosphere, and grounding could be likely.

- Switchgear and control panels have high potential for outage due to grounding if not secured against spray. This event is considered likely.

In diesel generator rooms protected by a CO_2 FPS, the following observations and considerations apply:

- In some configurations, actuation of the FPS system results in a control signal that shuts down the diesel generator. This, of course, compounds the problem in an inadvertent FPS actuation.

- In some configurations, actuation of the FPS system results in a control signal that isolates ventilation to and from the diesel generator room. In this configuration, there are two concerns. If the ventilation system is isolated and the engine continues to run, it is likely to overheat and fail. Additionally, if the engine draws its air supply from the room, engine shutdown due to starvation (high CO_2/low O_2 concentration preventing combustion) will result.

As was the case with the generic cable spreading room, in the generic diesel generator rooms, some assumptions are made based on the findings of the studies of the individual plants. It is assumed that the diesel generator rooms are not impacted by events in adjacent zones. This means that: (a) fires in an adjacent zone do not cause diesel generator room FPS actuation, (b) adjacent zone fire induced FPS actuation does not prevent random failure recovery action, and (c) adjacent zone fire induced FPS actuation does not prevent access for manual fire suppression. Accordingly, Root Causes 1, 2, and 3 are eliminated from consideration in the generic diesel generator rooms. It is additionally assumed that steam piping is not routed in such a way that steam pipe rupture will actuate the generic diesel generator room FPS. This assumption eliminates Root Cause 5 from consideration. Finally, Root Cause 10 was eliminated since in the evaluation of individual plants no cases were found where diesel generator room FPS were actuated by smoke alone.

4.4.1 Diesel Generator Rooms with a Preaction Water FPS Case

Seismic root causes of interest are Root Causes 7/8/9. For this analysis, manual non-recovery probabilities are set to 1.0 in the case of water suppressants, due to the long time required to clean and dry electrical equipment that has been wetted. Preaction water FPSs are installed in 36 diesel generator rooms in U.S. commercial nuclear power plant applications. For this space A_r is set at 0.25. For the seismic

Root Causes 7/8/9, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = (P_{d,w,cp})^2 \bullet (A_r)^2 \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet (P_{sysbreach}(g))^2 \bullet R_{chat}(g) \bullet P_{FPS\ non\text{-}div}(g)dg$$

and in the generic case:

$$\Phi_{cd,7/8/9} = <1.0E-8$$

Root Cause 12 was considered, but this root cause scenario would require a fire to occur in a diesel generator room component, that would not in itself lead to failure of the diesel generator. Diesel generator rooms inspected as a part of this study were found to be free of such equipment. Therefore, Root Cause 12 is not included in the generic diesel generator room case.

4.4.2 Diesel Generator Rooms with a Wetpipe Water FPS Case

The seismic root cause of interest is Root Cause 9. Wetpipe water FPS systems are installed in diesel generator rooms in 12 U.S. commercial nuclear power plant applications. For this analysis, manual non-recovery probabilities are set to 1.0 in the case of water suppressants due to the long time required to clean and dry electrical equipment that has been wetted. For the seismic Root Causes 9, core damage frequency is governed by the following relationship:

$$\phi_{cd,x} = (P_{d,w,cp})^2 \bullet (A_r)^2 \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet (P_{sys\ breach}(g))^2 \bullet P_{FPS\ non\text{-}div}(g)dg$$

and in the generic case:

$$\Phi_{cd,9} = 1.6E-8$$

4.4.3 Diesel Generator Rooms with a Deluge Water FPS Case

Seismic root causes of interest are Root Causes 7/8. Root Cause 9, piping breach, does not apply with open nozzle deluge systems. Additionally, area ratioing is not required, because the deluge system, when actuated, wets the entire zone. However, it must be pointed out that in the generic case, the deluge system is assumed to wet all safety-related components (exciter cabinets, engine control panels,, engine air intakes, etc.) in the zone. This was the case as observed in plant walkdowns conducted during the individual plant analysis. For other plants, this may not be the case, and should be verified on an individual plant basis. In this analysis for water FPS systems in diesel generator rooms, it is assumed that if the diesel generator is failed due to wetting, the recovery will require so much time that the accident sequence will be over prior to restoration of the diesel generator. For the seismic Root Causes 7/8, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = (P_{d,w.cp})^2 \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet R_{chat}(g) \bullet P_{FPS\ non\text{-}div}(g)dg$$

and in the generic case:

$$\Phi_{cd,7/8} = 8.8E\text{-}5$$

4.4.4 Diesel Generator Rooms with a CO_2 FPS Case

The seismic root cause of interest is Root Cause 8. CO_2 FPSs are installed in diesel generator rooms in 49 U.S. nuclear power plant applications. In this scenario, a seismic event results in relay chatter. It is assumed for the generic CO_2 FPS system for diesel generator rooms, that the control signal that initiates release of the CO_2 suppressant also shuts down and locks out the diesel generator. Therefore, suppressant damage is not a consideration in this case. However, unlike the cases of water suppressants, in the CO_2 FPS, credit is given for operator recovery following the earthquake. Based on individual plant analyses, recovery is a complex process, and is exacerbated by the post-earthquake conditions of both the operators and the systems. To restore the diesel generators, the operators must first recognize why they were lost. Then, the CO_2 in the diesel generator spaces must be vented. Breathing appliances (scuba-like equipment) must be donned in order to enter the spaces. Controllers must then be opened in order to reset the diesel generator lockout. For seismic Root Cause 8, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = Q \bullet \int P_{quake}(g) \bullet LOSP(g) \bullet R_{chat}(g) \bullet P_{FPS\ non\text{-}div}(g)dg$$

and in the generic case:

$$\phi_{cd,8} = 1.8E\text{-}5$$

4.5 Emergency Electrical Switchgear Rooms

The model for the generic emergency electrical switchgear rooms is similar to that selected for the diesel generator rooms. The generic emergency electrical switchgear is divided into two essentially identical rooms, each with similar but independently operated FPSs. The generic electrical switchgear rooms split the trains of safety-related systems. While at least ten types of emergency electrical switchgear room designs have been identified (Ref 4.13), the majority of arrangements are like the generic model. Figure 4.1, illustrates the various types of emergency electrical switchgear room arrangements in existing plants. The model EP-6 is selected for generic analysis as the model represents the majority of commercial plant, and there are not known specific vulnerabilities in a smaller group of plants that warrant individual or sensitivity analyses. The rooms contain sufficient components and cables such that initiation of accident sequences is possible without the addition of random failures in or outside of the generic emergency switchgear rooms. However, in the generic case, the impact of random failure of safety-related cables and

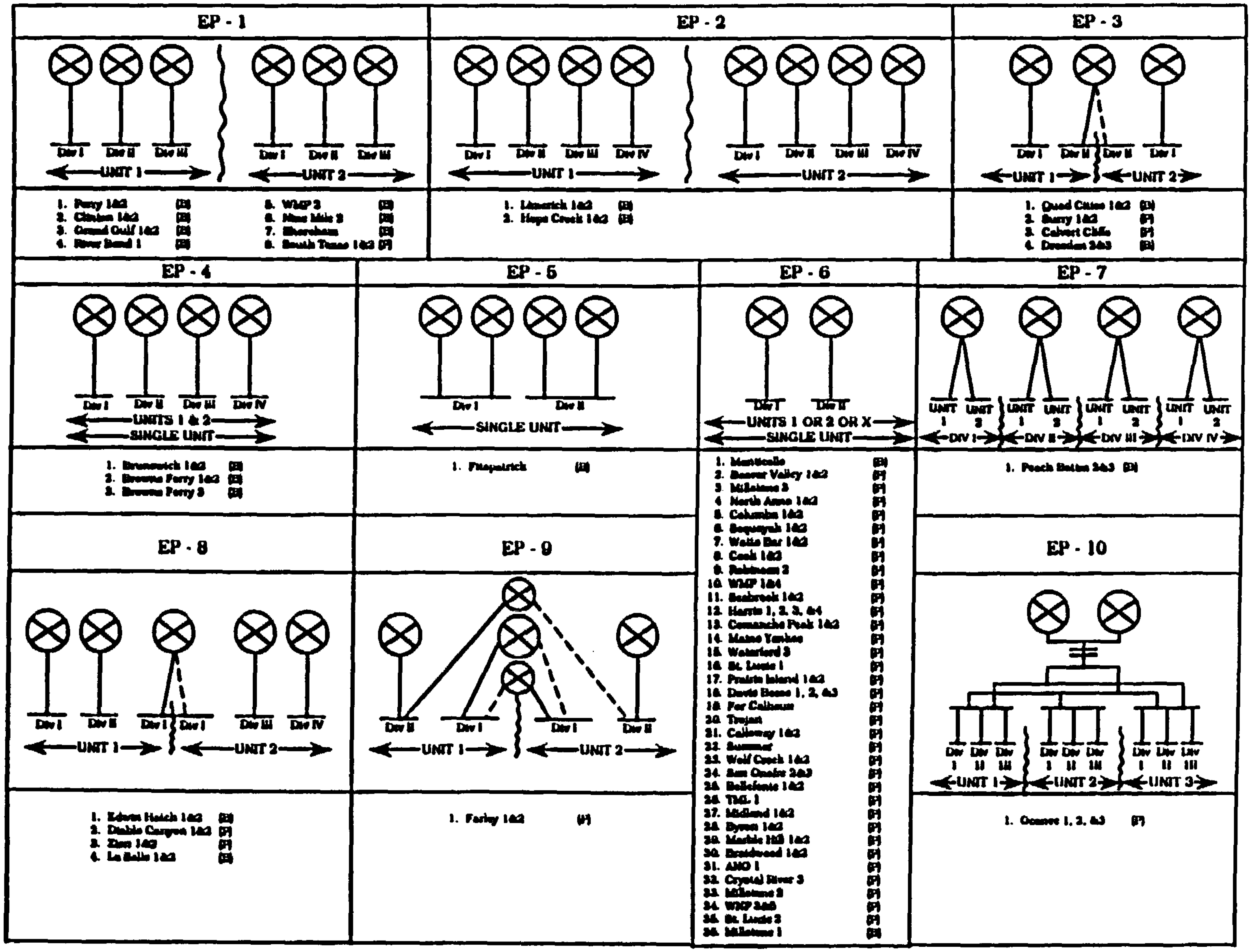

Figure 4.1. Emergency Electrical Switchgear Arrangements

components is examined. In the generic electrical switchgear rooms, the fire suppressant damage mechanism is associated with electromechanical equipment (the switchgear itself) as opposed to the less vulnerable cables. As in the other generic rooms studied, it is assumed Root Causes 1,2, and 3 do not apply. Based on the findings from the individual plants studied in detail, non-recovery probability is not an issue with this fire zone, as emergency operating procedures for bypassing the rooms were not found to be in place. Bypassing these rooms is too complex, based on operational engineering judgement, to be done "Ad Hoc", or an a "reasonable person response" basis.

4.5.1 Emergency Electrical Switchgear Rooms with a Wetpipe Water FPS Case

4.5.1.1 Non-Seismic Initiating Events

Wetpipe water systems are used in emergency switchgear rooms as a suppression agent in two commercial nuclear power plant applications. The following relationship represents the core damage frequency for releases of water in wetpipe water systems:

$$\Phi_{cd,x} = \kappa_{Ww,b} \bullet P_{d,w,cp} \bullet M \bullet A_r \bullet P_{rand,cc}$$

In the case of a wetpipe sprinkler system, a single fire protection system piping failure will cause wetting of only a small portion of the components in the generic emergency electrical switchgear room. Therefore it is assumed that to wet and subsequently damage a sufficient number of components to initiate an accident sequence, multiple fire protection system suppressant releases, or a very large suppressant release may well be required. To reflect this condition, in calculations of core damage frequency the A_r factor of 1.0E-1 is included. In these non-seismic sequences, the power conversion system must fail in conjunction with a failure of other safety-related components not located in the affected zone if a damage sequence is to be initiated. A single area failure alone cannot lead to core damage. Therefore for these root causes, it is assumed that failure of one of the switchgear rooms in conjunction with a random failure of one train (of emergency feedwater or high pressure coolant injection) is required. For failure of one train, $P_{rand,cc}$ is set at 3.0E-2/ry. For Root Causes 4,6, and 13:

$$\Phi_{cd,4} = 5.4E-8$$

$$\Phi_{cd,6} = 9.4E-8$$

$$\Phi_{cd,13} = 3.6E-8$$

4.5.1.2 Seismic Initiating Events

Seismic root causes of interest are Root Causes 9 and 12. For seismic Root Cause 9, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = P_{d,w,cp} \bullet \int (P_{quake}(g) \bullet LOSP(g) \bullet P_{rand,DG}(g) \bullet P_{sys\ breach}(g) \bullet P_{FPS\ non\text{-}div}(g))\, dg$$

and in the generic case:

$$\Phi_{cd,9} = 1.7E\text{-}6$$

For seismic Root Cause 12, the core damage sequence scenario is a seismic event causes a diversion of FPS agent from a central supply, a seismically induced fire occurs and is subsequently not suppressed.

Core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = \int P_{quake}(g) \bullet LOSP(g) \bullet P_{rand,DG}(g) \bullet P_{fire}(g) \bullet P_{FPS\ div}(g)\, dg$$

and in the generic case:

$$\Phi_{cd,12} = 1.2E\text{-}5$$

4.5.2 Emergency Electrical Switchgear Rooms with a Deluge Water FPS Case

4.5.2.1 Non-Seismic Initiating Events

Deluge water systems are used in emergency electrical switchgear rooms in four commercial nuclear power plant applications. The logic behind the cutset development is the same as for wetpipe water systems, except that in the case of a deluge system, it is assumed that it covers the entire space. Therefore, an area ratio is not required. The following relationship represents the core damage frequency for releases of water in deluge water systems:

$$\Phi_{cd,x} = k_{Wd,b} \bullet P_{d,w,cp} \bullet M \bullet P_{rand,cc}$$

and for Root Causes 4,6, and 13:

$$\Phi_{cd,4} = 1.8E\text{-}6$$

$$\Phi_{cd,6} = 1.1E\text{-}6$$

$$\Phi_{cd,13} = 2.5E\text{-}6$$

Because of observed locations of components and configuration of emergency electrical switchgear rooms, Root Cause 5 is considered in the generic case for deluge (open nozzle) systems. The scenario for this root cause is a steam, feed or condensate line break, intrusion of steam into a FPS controller, and subsequent damage to electromechanical components. There are two cases (cut sets) for consideration here. In one case, suppressant might be released into both of the generic emergency electrical switchgear rooms due to initiation by the same steam leak. A second cut set considers the case where one emergency electrical switchgear room is damaged due to such a release in combination with a random system failure of emergency feedwater or high pressure coolant injection. In both cases, random failure of the power conversion system is also required. The core damage frequency relationship for release of suppressant from the deluge system is:

$$\Phi_{cd,x} = \delta_y \bullet P_{d\ control,s} \bullet (P_{d,w,cp})^2 \bullet M + \delta_y \bullet P_{d\ control,s} \bullet P_{d,w,cp} \bullet P_{rand,cc} \bullet M$$

and in the generic case:

$$\Phi_{cd,5} = 2.2E-6$$

4.5.2.2 Seismic Initiating Events

Seismic root causes of interest are combinations of Root Causes 7/8 and Root Cause 12. For this analysis, manual non-recovery probabilities are increased to their upper bound because of the disruptive conditions in the plant following the initiating earthquake. For the seismic Root Causes 7/8, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = P_{d,w,cp} \bullet \int (P_{quake}(g) \bullet LOSP(g) \bullet P_{rand,DG}(g) \bullet P_{FPS\ non\text{-}div}(g) \bullet R_{chat}(g))dg$$

and in the generic case:

$$\Phi_{cd,7/8} = 1.7E-5$$

For seismic Root Cause 12, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = \int P_{quake}(g) \bullet LOSP(g) \bullet P_{rand,DG}(g) \bullet P_{fire}(g) \bullet P_{FPS\ div}(g)\ dg$$

and in the generic case:

$$\Phi_{cd,12} = 1.2E-5$$

4.5.3 Emergency Electrical Switchgear Rooms with a Halon FPS Case

4.5.3.1 Non-Seismic Initiating Events

Halon systems are used in emergency electrical switchgear rooms in 15 commercial nuclear power plant applications. The following relationship represents the core damage frequency for releases of suppressant in Halon systems:

$$\Phi_{cd,x} = \kappa_{H,b} \bullet P_{d,H,cp} \bullet M \bullet P_{rand,cc}$$

and for Root Causes 4,6, and 13:

$$\Phi_{cd,4} = 1.5E-8$$

$$\Phi_{cd,6} = 2.0E\text{-}8$$

$$\Phi_{cd,13} = <1.0E\text{-}8$$

Root Cause 5 is also considered in the generic case for Halon (open nozzle) systems. The scenario for this root cause is a steam, feed or condensate line break, intrusion of steam into an FPS controller, and subsequent damage to electromechanical components. The following relationship represents the core damage frequency for releases of suppressant in Halon systems:

$$\Phi_{cd,x} = \delta_y \bullet P_{d\ control,s} \bullet (P_{d,H,cp})^2 \bullet M + \delta_y \bullet P_{d\ control,s} \bullet P_{d,H,cp} \bullet P_{rand,cc} \bullet M$$

and in the generic case:

$$\Phi_{cd,5} = 4.3E\text{-}8$$

4.5.3.2 Seismic Initiating Events

Seismic root causes of interest are combinations of Root Causes 7/8 and Root Cause 12. For the seismic Root Causes 7/8, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = P_{d,H,cp} \bullet \int (P_{quake}(g) \bullet LOSP(g) \bullet P_{rand,DG}(g) \bullet P_{FPSnondiv}(g) \bullet R_{chat}(g))dg$$

and in the generic case:

$$\Phi_{cd,7/8} = 7.9E\text{-}7$$

For seismic Root Cause 12, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = \int P_{quake}(g) \bullet LOSP(g) \bullet P_{rand,DG}(g) \bullet P_{fire}(g) \bullet P_{FPS\ div}(g)\ dg$$

and in the generic case:

$$\Phi_{cd,12} = 3.9E\text{-}6$$

4.5.4 Emergency Electrical Switchgear Rooms with a CO_2 FPS Case

4.5.4.1 Non-Seismic Initiating Events

CO_2 systems are used in emergency electrical switchgear rooms in 18 commercial nuclear power plant applications. The following relationship represents the core damage frequency for releases of suppressant from CO_2 systems:

$$\Phi_{cd,x} = KCO_{2,b} \bullet P_{d,CO_2,cp} \bullet M \bullet P_{rand,cc}$$

and for Root Causes 4,6, and 13:

$$\Phi_{cd,4} = 1.5E-8$$

$$\Phi_{cd,6} = 2.5E-8$$

$$\Phi_{cd,13} = 2.1E-8$$

Root Cause 5 is also considered in the generic case for CO_2 (open nozzle) systems. The scenario for this root cause is a steam, feed or condensate line break, intrusion of steam into an FPS controller, and subsequent damage to electromechanical components. The following relationship represents the core damage frequency for releases of suppressant in CO_2 systems:

$$\Phi_{cd,x} = \delta_y \bullet P_{d\ control,s} \bullet (P_{d,CO_2,cp})^2 \bullet M + \delta_y \bullet P_{d\ control,s} \bullet$$

$$P_{d,CO_2,cp} \bullet P_{rand,DG} \bullet M$$

and in the generic case:

$$\Phi_{cd,5} = 1.2E-7$$

4.5.4.2 Seismic Initiating Events

Seismic root causes of interest are combinations of Root Causes 7/8 and 12. For the seismic Root Causes 7/8, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = P_{d,CO_2,cp} \bullet \int (P_{quake}(g) \bullet LOSP(g) \bullet P_{rand,DG}(g) \bullet$$

$$P_{FPS\ non\text{-}div}(g) \bullet R_{chat}(g))\ dg$$

and in the generic case:

$$\Phi_{cd,7/8} = 1.6E-6$$

For seismic Root Cause 12, core damage frequency is governed by the following relationship:

$$\Phi_{cd,x} = \int P_{quake}(g) \bullet LOSP(g) \bullet P_{rand,DG}(g) \bullet P_{fire}(g) \bullet P_{FPS\ div}(g)\ dg$$

and in the generic case:

$$\Phi_{cd,12} = 9.4E-6$$

4.6 Generic Core Damage Frequency Summary and Uncertainty Analysis

This section provides tabular summaries of core damage frequency in each of the selected generic fire zones, the cable spreading room, diesel generator room(s), and the emergency electrical switchgear room(s). Within each zone, a separate summary table is provided. Core damage frequency is summarized by root cause. Tables 4.5 through 4.7 present

Table 4.5

Core Damage Frequency - Generic Cable Spreading Room

Root Cause	Preaction Water	Wetpipe Water	Deluge Water	Halon	CO_2
4	<1.0E-8	4.7E-8	1.6E-6	6.0E-7	2.3E-7
5			2.6E-8	2.6E-8	2.6E-8
6	<1.0E-8	8.4E-8	9.1E-7	8.8E-7	3.9E-7
7			2.6E-5		
8			1.1E-6	1.3E-6	1.2E-6
9		5.3E-8			
7/8					
8/9	1.3E-8				
7/8/9					
11			4.4E-7		
12	1.2E-5	1.2E-5	1.2E-5	3.2E-6	9.1E-6
13	<1.0E-8	2.9E-8	2.3E-6	1.4E-7	3.0E-7
TOTAL	1.2E-5	1.2E-5	4.4E-5	6.2E-6	1.1E-5

Table 4.6

Core Damage Frequency - Generic Diesel Generator Room

Root Cause	Preaction Water	Wetpipe Water	Deluge Water	CO_2
4				
5				
6				
7				
8				1.8E-5
9		1.6E-8		
7/8			8.8E-5	
8/9				
7/8/9	<1.0E-8			
11				
12				
13				
TOTAL	<1.0E-8	1.6E-8	8.8E-5	1.8E-5

Table 4.7

Core Damage Frequency - Generic Emergency Electrical Switchgear Room

Root Cause	Wetpipe Water	Deluge Water	Halon	CO_2
4	5.4E-8	1.8E-6	1.5E-8	1.5E-8
5		3.1E-6	6.2E-8	2.6E-7
6	9.4E-8	1.1E-6	2.0E-8	2.5E-8
7				
8				
9	1.7E-6			
7/8		1.7E-5	7.9E-7	1.6E-6
8/9				
7/8/9				
11				
12	1.2E-5	1.2E-5	3.9E-6	9.4E-6
13	3.6E-8	2.5E-6	<1.0E-8	2.1E-8
TOTAL	1.4E-5	3.8E-5	4.8E-6	1.1E-5

these data. The details of core damage frequency and uncertainty calculations are given in Appendix F. Totals for a set of typical rooms are presented in Section 4.9.

4.7 Offsite Dose and Risk Assessment and Uncertainty Analysis

This section addresses the derivation of generic offsite dose calculations and presents generic risk values for each of the applicable root causes. The MARCH (Meltdown Accident Response Characteristics) code approach (Ref. 4.14) to core damage consequence analysis is used. This analytic approach includes the following considerations:

- Analysis of in-vessel thermal-hydraulic processes during the accident sequence
- Vessel water boil-off and core support failure.
- Core debris-concrete interaction.
- Containment volume steam-condensing and heat-removal system performance.
- Containment failure modes including considerations for:
    - In-vessel steam explosion.
    - Containment leakage.
    - Hydrogen burn overpressure.
    - Ex-vessel steam pressure spiking.
    - Steam and non-condensible gas overpressure.
    - Base mat melt through.

Fission product release analysis includes consideration of several categories of releases (i.e. cladding rupture release, fuel melting, fission product vaporization, steam explosion/fuel oxidation, etc.) each evaluated for eight groups of radionuclides (noble gasses, molecular iodine, organic iodine, cesium-rubidium, tellurium, barium-strontium, ruthenium, and lanthanum). Radionuclides are removed from the release by settling, deposition, spray condensing, scrubbing, filtering, etc.

Once outside the containment, the behavior of the released fission product radionuclides is analyzed with a Gaussian dispersion calculation, taking into account the thermodynamic properties of the source (plume buoyancy), meteorological conditions, surface roughness, etc. Finally offsite human radiation dosage results are calculated considering such factors as population distribution, respiration rates, food-chain ingestion paths, sheltering and evacuation plans, radiation dose/organ conversion factors, etc.

Actual calculations have been simplified by development of plant specific factors that can be used in conjunction with core damage frequency and the accident sequence type to obtain summed offsite human dosage. Here it is important to remember that the type of accident sequence determines the containment failure mode. Thus it is possible that two accident sequences with the same core damage frequency might result in a markedly different offsite dose, because of the difference in containment performance during the different accident types (sequences).

In the following calculations of offsite dose attributable to core damage frequency associated with the generic cable spreading room, diesel generator room, and emergency electrical switchgear room, the following assumptions are made:

- Offsite dose calculations are confined to the summation of dose received within a 50 mile radius of the plant. While in the most catastrophic accidents this radius limitation is insufficient, for credible accidents a 50 mile radius is sufficient.

- Offsite dose calculations are based on an estimated 20 year lifetime remaining in the generic plant.

- Dose estimates are calculated based on a central estimate, and an upper and lower bound. The central estimate is 30% of the upper bound and the lower bound is 10% of the upper bound. This methodology is that used in Reference 4.15 and in the analyses associated with investigation of U.S. Nuclear Regulatory Commission Unresolved Safety Issue 45 (USI-45), "Decay Heat Removal Requirements.".

- In the generic PWR configurations, the containment failure mode accident sequence type is:

  - For sequences involving the cable spreading room, control power is lost to the containment systems. For sequences involving the diesel generator rooms and the emergency electrical switchgear rooms, loss of all AC power includes the loss of power to the containment systems. Thus in all these sequences, there is failure of the containment overpressure protection systems (COP) and the post-accident radioactivity removal system (PARR)

- In the generic BWR configurations, containment failure mode accident sequence types are:

  - For the diesel generator room and emergency electrical switchgear rooms, the sequences result in loss of coolant injection capability and station blackout.

  - For the cable spreading room, the sequences result in transients with loss of coolant injection capability.

The following variables are used in the discussion of generic plant risk:

$R_{x,s}$ = Risk resulting from Root Cause x accident s in person-REM for the population within 50 mile radius of generic plant, summed over 20 reactor-years of remaining plant life.

$\Phi_{cd,x,s}$ = Frequency of core damage resulting from Root Cause x from accident sequence s per reactor-year.

$CF_{m,s}$ = Containment failure probability for failure mode m applicable to sequences.

$FPR_{c,s}$ = Fission product release category c applicable to sequence s.

For this analysis, values for $CF_{m,s}$ and $FPR_{c,s}$ have been taken from the specific plants studied and are considered typical for a PWR and BWR. In that there are literally hundreds of variables involved in their development, they are highly plant specific. However, for this analysis generic 20 reactor-year risk is based on these values, and is calculated by the following relationship:

$$R_{x,s} = \Phi_{cd,x,s} \bullet \Sigma\{CF_{m,s} \bullet FPR_{c,s}\} \bullet 2.0E+1$$

Values for $CF_{m,s}$ and $FPR_{c,s}$ for the PWR generic case are shown in Tables 4.8 and 4.9.

Similarly, for the generic BWR, values for $CF_{m,s}$ and $FPR_{c,s}$ are provided in Tables 4.10 and 4.11.

Using the values for the generic PWR and BWR source terms and containment factors, the risk associated with the generic cable spreading room, diesel generator rooms and emergency electrical switchgear rooms are presented in Table 4.12 through 4.17. The details of these calculations and risk uncertainty analyses are shown in Appendix G.

4.8 Sensitivity Studies

Many parameters could be chosen for consideration in sensitivity studies concerning the effects of FPS system actuation on safety-related equipment. While not all assessed in this analysis, these studies might include:

- Non-Seismic considerations:
  - Pdam of components for Halon set to zero.
  - Reduced Pdam of components and cable by CO_2.
  - Reduced Pdam of components and cable by water.
  - Barrier failure probability reduced by a factor of X.

Table 4.8

Estimated PWR Containment Failure Modes - $CF_{m.s}$

Accident Sequence	Containment Failure Mode with Probability and Release Category α	β	γ, δ_e	δ_1	ε
COP/PARR Fail	1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6

Where,

α = In-vessel steam explosion,
β = Containment leakage,
γ = Hydrogen burn overpressure,
δ_e = Ex-vessel steam spike,
δ_1 = Steam and non-condensible gas overpressure, and
ε = Base mat melt through.

Table 4.9

PWR Release Categories - $FPR_{c,s}$
(Person-REM/reactor-year within 50 miles of plant)

Release Category	Upper Bound	Central Estimate Case	Lower Bound
1	7.9E+5	4.7E+5	2.9E+5
2	7.5E+5	4.9E+5	3.5E+5
3	5.9E+5	3.5E+5	2.3E+5
4	3.2E+5	2.1E+5	1.4E+5
5	2.0E+5	1.2E+4	6.5E+4
6	5.4E+4	2.3E+4	1.0E+4
7	6.0E+3	2.4E+3	2.1E+3

Table 4.10

Estimated BWR Containment Failure Modes - $CF_{m.s}$

Accident Sequence Type	Containment Failure Mode with Probability and Release Category				
	α	β	γ'	γ	δ
LOCA, Loss of Coolant Injection Systems and Station Blackout	1.0E-2 1	1.0E-2 2	1.8E-1 2	7.3E-1 3	1.0E-2 4
Transients with Loss of Coolant Injection Systems	1.0E-2 1	n/a	2.0E-1 2	7.8E-1 3	1.0E-2 4

Where,

α = Containment failure from in-vessel steam explosion
β = Containment failure from in-containment steam explosion,
γ' = Containment failure from overpressure with release direct to atmosphere burn overpressure,
γ = Containment failure from overpressure with release through the reactor building, and,
δ = Containment isolation failure.

Table 4.11

BWR Release Categories - $FPR_{c.s}$
(Person-REM/reactor-year within 50 miles of plant)

Release Category	Upper Bound	Central Estimate Case	Lower Bound
1	4.3E+5	3.8E+5	2.3E+5
2	6.2E+5	4.7E+5	2.8E+5
3	5.0E+5	2.9E+5	1.8E+5
4	9.2E+4	5.8E+4	3.2E+4

Table 4.12

Twenty Year Risk (Person-REM) - PWR Cable Spreading Room

Root Cause	Preaction Water	Wetpipe Water	Deluge Water	Halon	CO_2
4	6.5E-3	3.4E-2	1.0E+0	9.0E-1	3.6E-1
5			4.4E-2	4.6E-2	4.2E-2
6	6.8E-3	5.9E-2	6.0E-1	1.3E+0	5.5E-1
7			4.5E+1		
8			1.8E+1	2.2E+0	2.0E+0
9		9.1E-2			
7/8					
8/9	2.1E-2				
7/8/9					
11			7.0E-1		
12	2.0E+1	2.0E+1	2.0E+1	5.4E+0	1.5E+1
13	1.1E-2	2.7E-2	1.5E+0	2.3E-1	4.4E-1
TOTAL	2.0E+1	2.0E+1	2.2E+1	5.6E+0	1.5E+1

Table 4.13

Twenty Year Risk (Person-REM) - PWR Diesel Generator Room

Root Cause	Preaction Water	Wetpipe Water	Deluge Water	CO_2
4				
5				
6				
7				
8				3.1E+1
9		2.9E-2		
7/8			1.5E+2	
8/9				
7/8/9	1.1E-2			
11				
12				
13				
TOTAL	1.1E-2	2.9E-2	1.5E+2	3.1E+1

Table 4.14

Twenty Year Risk (Person-REM)-PWR Emergency Electrical Switchgear Room

Root Cause	Wetpipe Water	Deluge Water	Halon	CO_2
4	4.0E-2	1.1E+0	2.0E-2	2.3E-2
5		5.2E+0	1.6E-1	3.3E-1
6	6.9E-2	6.7E-1	2.9E-2	3.7E-2
7				
8				
9	2.9E+0			
7/8		2.7E+1	1.3E+0	2.6E+0
8/9				
7/8/9				
11				
12	2.0E+1	2.0E+1	7.0E+0	1.6E+1
13	3.1E-2	1.7E+0	5.2E-3	2.9E-2
TOTAL	2.3E+1	5.6E+1	8.5E+0	1.9E+1

Table 4.15

Twenty Year Risk (Person-REM) - BWR Generic Cable Spreading Room

Root Cause	Preaction Water	Wetpipe Water	Deluge Water	Halon	CO_2
4	5.3E-2	2.7E-1	1.0E+1	3.7E+0	1.5E+0
5			1.8E-1	1.9E-1	1.7E-1
6	5.5E-2	5.0E-1	5.6E+0	5.3E+0	2.3E+0
7			1.8E+2		
8			7.5E+0	8.8E+0	8.0E+0
9		3.6E-1			
7/8					
8/9	8.6E-2				
7/8/9					
11			2.9E+0		
12	8.2E+1	8.2E+1	8.2E+1	2.2E+1	6.2E+1
13	5.5E-2	1.7E-1	1.4E+1	9.2E-1	1.8E+0
TOTAL	8.2E+1	8.3E+1	3.0E+2	4.1E+1	7.6E+1

Table 4.16

Twenty Year Risk (Person-REM)- BWR Generic Diesel Generator Room

Root Cause	Preaction Water	Wetpipe Water	Deluge Water	CO_2
4				
5				
6				
7				
8				1.2E+2
9		1.0E-1		
7/8			5.7E+2	
8/9				
7/8/9	4.2E-2			
11				
12				
13				
TOTAL	4.2E-2	1.0E-1	5.7E+2	1.2E+2

Table 4.17

Twenty Year Risk (Person-REM)- BWR Generic Emergency Electrical Switchgear Room

Root Cause	Wetpipe Water	Deluge Water	Halon	CO_2
4	3.0E-1	9.6E+0	7.8E-2	8.6E-2
5		2.0E+1	4.4E-1	1.3E+0
6	5.4E-1	5.8E+0	1.1E-1	1.5E-1
7				
8				
9	1.1E+1			
7/8		1.1E+2	5.0E+0	1.1E+1
8/9				
7/8/9				
11				
12	7.4E+1	7.4E+1	2.5E+1	6.1E+1
13	1.9E-1	1.5E+1	2.0E-2	1.1E-1
TOTAL	8.6E+1	2.3E+2	3.1E+1	7.4E+1

- Variability of PWR versus BWR.

- Seismic considerations:

    - LLNL versus EPRI seismic hazard curves.

    - Average East Coast versus West Coast seismic hazard curves.

    - High end and low end of East and West Coast seismic hazard curves verses the median curves.

    - Fragility of relays, high and low versus the median value.

    - Fragility of mercury actuated control relays.

    - Decrease in the probability of fire due to tipping or sliding of electrical cabinets.

    - Fragility of FPS piping systems, high and low versus a median value.

    - Sensitivity of recovery factor by variability of required recovery time for diesel generator rooms.

    - Variability of PWR versus BWR.

    - Seismic vs non-seismic qualification of water suppressant FPS.

While all these variables are candidates for sensitivity analysis, many can be eliminated from further consideration based on analysis of the risk associated with the root causes to which they apply. Examination of the summary tables of core damage frequency, Tables 4.5 through 4.7, indicated that the root causes of most concern (those with core damage frequencies >1.0E-6) are only the seismic root causes. Therefore, for purposes of sensitivity analysis, only the seismic area will be investigated. Halon damage to cables and active electromechanical components is questionable (Ref. 4.16). Even with the damage values assigned in the generic base cases, the total core damage frequency for any of the Halon protected rooms analyzed was only in the mid-1E-6 range. Halon systems damage to cable and component sensitivity is assessed without the use of a sensitivity table. If there is no damage that results from exposure to Halon, then the root causes associated with that damage mechanism have core damage frequencies of zero.

Remaining for investigation in sensitivity analysis in this study are the water and CO_2 systems in seismic events. The parameters that will be investigated are as follows:

- Comparison of the use of the LLNL (Ref. 4.17) versus the EPRI (Ref. 4.18) seismic hazard curves. This comparison is on-going in the risk analysis community and must be considered. At this time, both sets of hazard curves are viewed by the USNRC as being equally credible. As such, calculations of the seismic core damage frequencies can be made for both sets of hazard curves and the results viewed as a measure of methodological uncertainty in the hazard curve development process. In the base case analysis, the LLNL seismic hazard curves were utilized to calculate the CDF contribution for each of the applicable seismic root causes to be consistent with the NUREG-1150 studies. As a point of comparison, the CDF contribution from the seismic root causes were also calculated using the EPRI seismic hazard curves. All other values were kept the same as in the base case study. Figures 4.2 and 4.3 present the LLNL hazard curves and the EPRI hazard curves, respectively.

- Investigation of the use of mercury (high seismic fragility) relays in the control systems for preaction water, deluge water, and CO_2 systems. These relays were found in fire protection system controls in two of the three plants subjected to detailed analysis. Relays of moderate fragility were used in the base case. This study demonstrates the increase in risk that results from these relays which are highly susceptible to seismic actuation.

- Reduction of the conditional probability of electrical cabinet fire, given cabinet tipping or sliding in a seismic event. For the base case, given tipping or sliding, a fire was assumed to occur 50 percent of the time. This value is based on engineering judgement, with little supporting data, and is believed to be conservative. The sensitivity study lowers this estimate for fire occurrence to only 10 percent of the time.

The results of these sensitivity studies are presented in Tables 4.18 through 4.28

4.9 "Most Vulnerable/Typical/Least Vulnerable Case" Generic Plants

This section examines core damage frequency (CDF) and 20-year risk in three generic plant cases, with risk calculated for both the PWR and BWR subcases. The following caveats must be kept in mind when reviewing the data presented in these cases:

- The generic plants examined are represented by only three fire zones: the cable spreading room, the diesel generator rooms, and the emergency electrical switchgear rooms. In any given specific plant, other fire zones may be significant or even dominant contributors to CDF and risk, and specific plant analysis must be conducted to evaluate an individual plant.

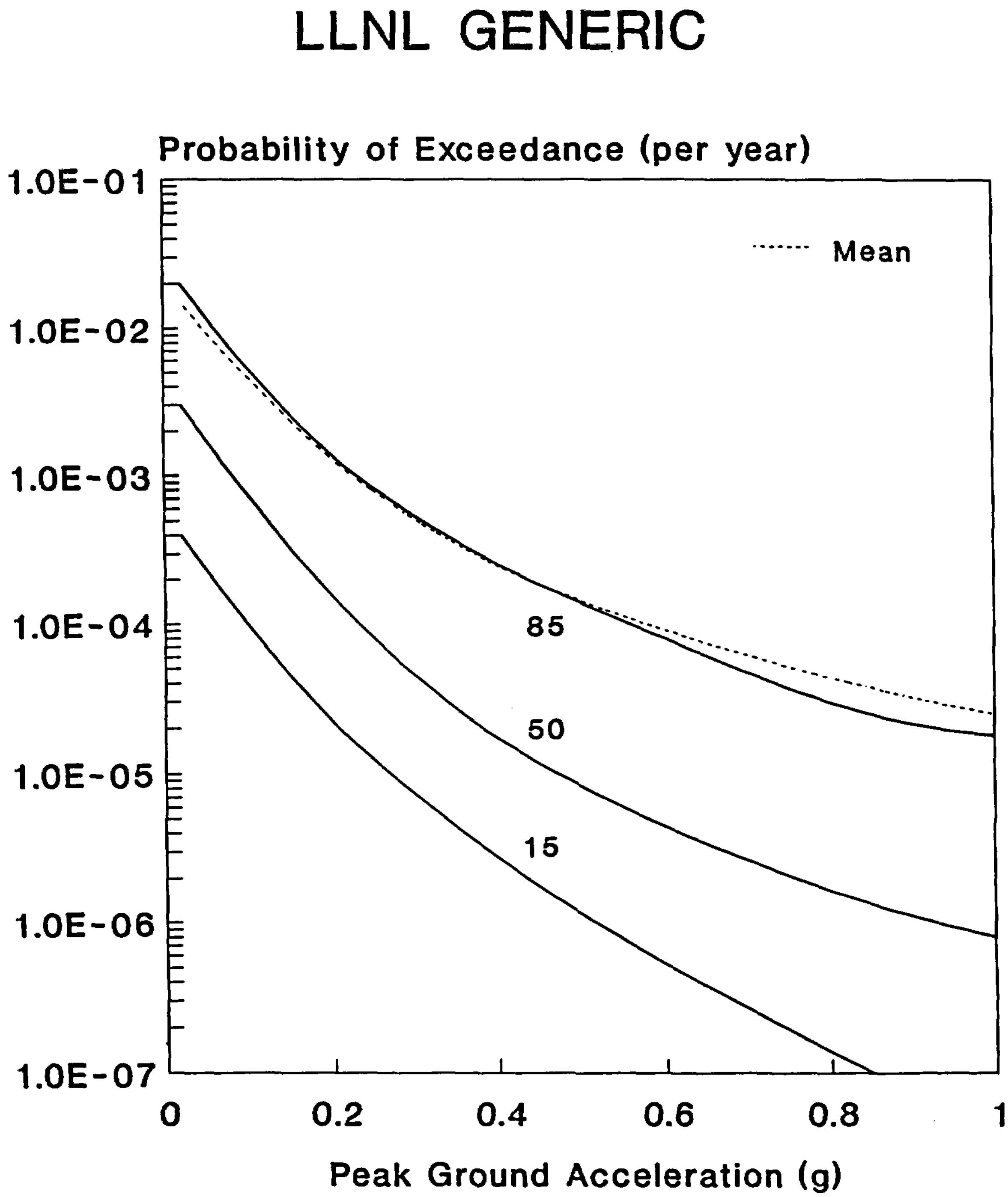

Figure 4.2. LLNL Average East Coast Hazard Curves: Mean, Median 85th and 15th Percentile Curves for Generic Plant

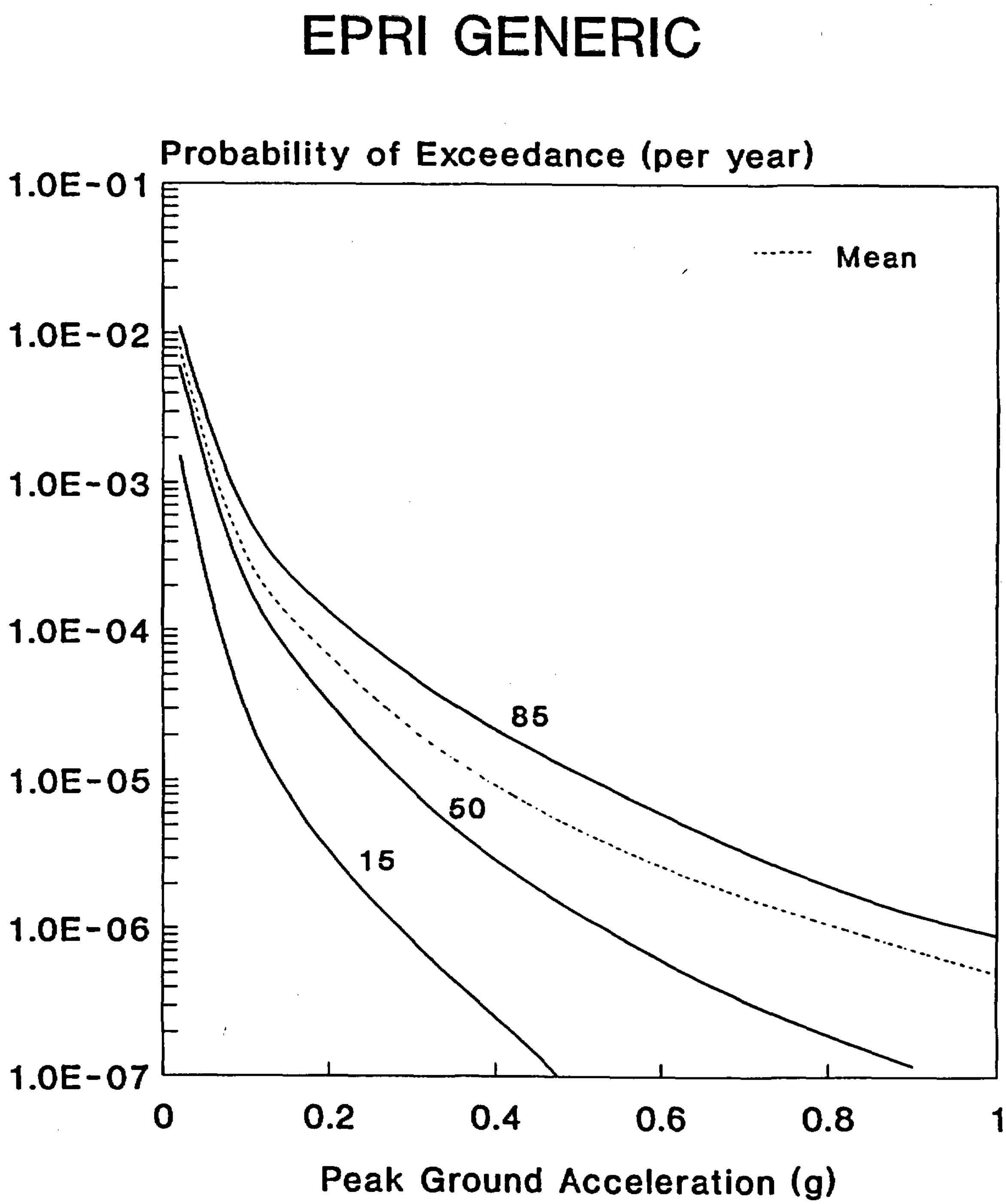

Figure 4.3. EPRI Average East Coast Hazard Curves: Mean, Median 85th and 15th Percentile Curves for Generic Plant

Table 4.18

Generic Cable Spreading Room (Preaction Water FPS) Summary of Sensitivity Results in Terms of Core Damage Frequency (Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
8/9.	1.3E-8	6.3E-10	5.3E-8	1.3E-8
12.	1.2E-5	5.3E-7	1.2E-5	2.5E-6

Table 4.19

Generic Cable Spreading Room (Wetpipe Water FPS) Summary of Sensitivity Results in Terms of Core Damage Frequency (Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
9.	5.3E-8	3.9E-9	5.3E-8	1.3E-8
12.	1.2E-5	5.3E-7	1.2E-5	2.5E-6

Table 4.20

Generic Cable Spreading Room (Deluge Water FPS)
Summary of Sensitivity Results in Terms of Core Damage Frequency
(Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
7.	2.6E-5	3.6E-6	2.6E-5	2.6E-5
8.	1.1E-6	1.2E-7	2.6E-5	1.1E-6
12.	1.2E-5	5.3E-7	1.2E-5	2.5E-6

Table 4.21

Generic Cable Spreading Room (CO_2 FPS)
Sensitivity of Sensitivity Results in Terms of Core Damage Frequency
(Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
8.	1.2E-6	1.3E-7	2.5E-5	1.2E-6
12.	9.1E-6	3.8E-7	9.1E-6	1.8E-6

Table 4.22

Generic Diesel Generator Room (Preaction Water FPS)
Summary of Sensitivity Results in Terms of Core Damage Frequency
(Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
8/9.	6.6E-9	2.7E-10	1.6E-8	6.6E-9

Table 4.23

Generic Diesel Generator Room (Wetpipe Water FPS)
Summary of Sensitivity Results in Terms of Core Damage Frequency
(Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
9.	1.6E-8	7.4E-10	1.6E-8	1.6E-8

Table 4.24

Generic Diesel Generator Room (Deluge Water FPS)
Summary of Sensitivity Results in Terms of Core Damage Frequency
(Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
8.	8.8E-5	1.0E-5	2.2E-3	8.8E-5

Table 4.25

Generic Diesel Generator Room (CO_2 FPS)
Summary of Sensitivity Results in Terms of Core Damage Frequency
(Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
8.	1.8E-5	1.9E-6	3.9E-4	1.8E-5

Table 4.26

Generic Emergency Electrical Switchgear Room (Wetpipe Water FPS)
Summary of Sensitivity Results in Terms of Core Damage Frequency
(Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
9.	1.7E-6	9.0E-8	1.7E-6	1.7E-6
12.	1.2E-5	4.8E-7	1.2E-5	2.4E-6

Table 4.27

Generic Emergency Electrical Switchgear Room (Deluge Water FPS)
Summary of Sensitivity Results in Terms of Core Damage Frequency
(Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
7/8.	1.7E-5	1.3E-6	2.2E-4	1.7E-5
12.	1.2E-5	4.8E-7	1.2E-5	2.4E-6

Table 4.28

Generic Emergency Electrical Switchgear Room (CO_2 FPS)
Summary of Sensitivity Results in Terms of Core Damage Frequency
(Mean Values Per Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Use of Mercury FPS Relays	Study 3 Reduced Seismic Fire Hazard
8.	1.6E-6	1.1E-7	1.6E-5	1.6E-6
12.	9.4E-6	3.8E-7	9.4E-6	1.9E-6

- Because the level of damage to electrical cables and electromechanical components that results from short term exposure to Halon FPS agent is not clear from experimental and historical data, Halon FPS systems are not considered in evaluation of the "least vulnerable case" plant. If the assumption is made that Halon presents no short term threat to cables and components, then the incremental CDF and risk associated with a generic plant with all Halon FPS systems would be only that resulting from Root Cause 12 (Seismic/fire interaction). CDF and risk associated with Halon suppressant agent damage to cables and components would be zero.

- CDF and risk data in the tables representing the generic cases are mean values for the individual root cause and room totals, and the sum of mean values for the overall plant values. Uncertainty calculations were not accomplished in examining the overall generic plant CDF and risk values. However, uncertainty could be expected to be distributed in a way similar to that in the specific plant analysis. From the 5% to the 95% point in composite CDF, the range was about two orders of magnitude. For risk, the range was about a factor of 20.

- In the typical and least vulnerable generic plant cases, while in the CDF and risk tables a value is presented for the emergency electrical switchgear rooms, it must be recognized that data in Appendix D indicates that an automatic FPS is installed in these rooms in only about 20 percent of U.S. commercial nuclear power plants. In the remaining 80 percent of these plants, there is no automatic FPS system installed. Thus for these generic cases, two overall values for CDF are provided, one for the case with an FPS in the emergency electrical switchgear rooms, and one for the case with no automatic FPS in those rooms. It is suspected that although Generic Issue 57 associated CDF and risk for the case with no automatic FPS installed is lower, the overall plant CDF and risk may not be lower because of the likelihood that the CDF associated with fire in these rooms is higher without an automatic FPS systems. A detailed analysis of this issue was not conducted.

- For all the cases, the LLNL seismic hazard curves were used.

4.9.1 "Most Vulnerable Case" Generic Plant

In the most vulnerable case plant, for each of the three rooms examined (cable spreading, diesel generator, and emergency electrical switchgear) the FPS system resulting in the highest CDF was selected. In all cases, the resulting FPS system is deluge water. Additionally, for this case only, mercury wetted contact type relays were assumed to be installed in the deluge FPS control system. Core damage frequency and risk data (PWR and BWR) for this case are presented in Tables 4.29 through 4.31. For the most vulnerable case generic plant, CDF is calculated to be 2.5E-3/reactor-year. For the most vulnerable case generic PWR, 20 year risk is 3800 Person-REM. For the most vulnerable case generic BWR, 20 year risk is 15,000 Person-REM.

Table 4.29

Core Damage Frequency - "Most Vulnerable Case" Plant

Fire Zone:	Cable Spreading Room	Diesel Generator Rooms	Emergency Electrical Switchgear Rooms
FPS Type:	Deluge Water	Deluge Water	Deluge Water
Root Cause			
4	1.6E-6		1.8E-6
5	2.6E-8		3.1E-6
6	9.1E-7		1.1E-6
7	2.6E-5		
8	2.6E-5		
9			
7/8		2.2E-3	2.2E-4
8/9			
7/8/9			
11	4.4E-7		
12	1.2E-5		1.2E-5
13	2.3E-6		2.5E-6
Total	6.9E-5	2.2E-3	2.4E-4

Total CDF for "Most Vulnerable Case" Generic Plant: 2.5E-3/reactor-year.

Table 4.30

Risk (Person-REM - "Most Vulnerable Case" PWR Plant

Fire Zone:	Cable Spreading Room	Diesel Generator Rooms	Emergency Electrical Switchgear Rooms
FPS Type:	Deluge Water	Deluge Water	Deluge Water
Root Cause			
4	1.2E-1		1.4E-1
5	2.0E-3		2.4E-1
6	6.9E-2		8.4E-2
7	2.0E+0		
8	2.0E+0		
9			
7/8		1.7E+2	1.7E+1
8/9			
7/8/9			
11	3.4E-2		
12	9.1E-1		9.1E-1
13	1.8E+0		1.9E-1
Total	5.3E+0	1.7E+2	1.8E+1

Total risk for "Most Vulnerable Case" PWR Generic Plant: 190 Person-REM/reactor-year

Table 4.31

Risk (Person-REM) - "Most Vulnerable Case" BWR Plant

Fire Zone:	Cable Spreading Room	Diesel Generator Rooms	Emergency Electrical Switchgear Rooms
FPS Type:	Deluge Water	Deluge Water	Deluge Water
Root Cause			
4	5.2E-1		5.5E-1
5	8.4E-3		9.5E-1
6	3.0E-1		3.4E-1
7	8.4E+0		
8	8.4E+0		
9			
7/8		6.7E+2	6.7E+1
8/9			
7/8/9			
11	1.4E-1		
12	3.9E+0		3.7E+0
13	7.5E-1		7.6E-1
Total	2.3E+1	6.7E+2	7.4E+1

Total risk for "Most Vulnerable Case" BWR Generic Plant: 760 Person-REM/reactor-year.

4.9.2 "Typical Case" Generic Plant

In the typical case plant, for each of the three rooms examined, the FPS was chosen that represents the most common installation in commercial U.S. nuclear power plants. Based on survey data (Appendix D to NUREG/CR-5580), there are 69 cable spreading rooms with wetpipe water FPS systems, 49 diesel generator rooms with CO_2 FPS systems, and 18 switchgear rooms with CO_2 FPS systems. Core damage frequency and risk data (PWR and BWR) associated with these systems in the generic plant are shown in Table 4.32 through 4.34. For the typical case generic plant, CDF is calculated to be 3.5E-5/reactor-year with an automatic FPS in the emergency electrical switchgear rooms, and 3.3E-5/reactor-with no automatic systems installed in the emergency electrical switchgear rooms. For the typical case PWR, 20-year risk is 54 Person-REM with an automatic FPS installed in the emergency electrical switchgear room, and 51 Person-REM without. For the typical case BWR, 20 year risk is 220 Person-REM with an automatic FPS installed in the emergency electrical switchgear room, and 210 Person-REM without.

4.9.3 "Least Vulnerable Case" Generic Plant

In the least vulnerable case plant several assumptions are made to optimize the plant for minimum CDF associated with Generic Issue 57. The assumptions are based on the information gained from the study. The assumptions are:

- For each of the three rooms examined, the FPS system resulting in the lowest CDF was selected. Accordingly, a CO_2 FPS was selected for the cable spreading room, a preaction water FPS was selected for the diesel generator rooms, and a CO_2 FPS was selected for the emergency electrical switchgear rooms.

- For the cable spreading room, it was assumed that there are no electrical cabinets in the room to act as fire sources in a seismic event. This assumption is consistent with some, but not all of the individual plants walked-down. This eliminates CDF associated with Root Cause 12 (seismic/fire interaction) in this space.

- For the emergency electrical switchgear rooms, the electrical cabinets that remain energized in a LOSP event are assumed to be seismically restrained against sliding or tipping, eliminating the CDF associated with Root Cause 12 (seismic/fire interaction). This kind of cabinet restraint was observed in some, but not all of the plants walked-down.

- For all three fire zones, relays in the FPS control systems are assumed to be seismically qualified, and the CDF associated with relay chatter is reduced by a factor of 10 from that in the typical generic plant case. Such seismically qualified relays were found in some, but not all, of the plants walked-down.

Table 4.32

Core Damage Frequency - "Typical" Plant

Fire Zone:	Cable Spreading Room	Diesel Generator Rooms	Emergency Electrical Switchgear Rooms
FPS Type:	Wetpipe Water	CO_2	CO_2
Root Cause			
4	4.7E-8		1.5E-8
5			2.6E-7
6	8.4E-8		2.5E-8
7			
8		1.2E-5	
9	5.3E-8		
7/8			1.6E-6
8/9			
7/8/9			
11			
12	1.2E-5		9.4E-6
13	2.9E-8		2.1E-8
Total	1.2E-5	1.2E-5	1.1E-5

Total CDF for "Typical" Generic Plant: 3.5E-5/reactor-year with an automatic FPS installed in the emergency electrical switchgear room, 3.3E-5/reactor-year without

Table 4.33

Risk (Person-REM) - "Typical" PWR Plant

Fire Zone:	Cable Spreading Room	Diesel Generator Rooms	Emergency Electrical Switchgear Rooms
FPS Type:	Wetpipe Water	CO_2	CO_2
Root Cause			
4	3.6E-3		1.1E-3
5			2.0E-2
6	6.4E-3		1.9E-3
7			
8		9.1E-1	
9	4.0E-3		
7/8			1.2E-1
8/9			
7/8/9			
11			
12	9.1E-1		7.2E-1
13	2.2E-3		1.6E-3
Total	9.3E-1	9.1E-1	8.6E-1

Total risk for "Typical" PWR Generic Plant: 2.7 Person-REM/reactor-year with an automatic FPS installed in the emergency electrical switchgear room, 2.6 Person-REM/reactor-year without.

Table 4.34

Risk (Person-REM) - "Typical" BWR Plant

Fire Zone:	Cable Spreading Room	Diesel Generator Rooms	Emergency Electrical Switchgear Rooms
FPS Type:	Wetpipe Water	CO_2	CO_2
Root Cause			
4	1.5E-2		4.6E-3
5			7.9E-2
6	2.7E-2		7.6E-3
7			
8		3.7E+0	
9	1.7E-2		
7/8			4.9E-1
8/9			
7/8/9			
11			
12	3.9E+0		2.9E+0
13	9.4E-3		6.4E-3
Total	3.9E+0	3.7E+0	3.5E+0

Total risk for "Typical" BWR Generic Plant: 11 Person-REM/reactor-year with an automatic FPS installed in the emergency electrical switchgear room, 11 Person-REM/reactor-year without.

Core damage frequency and risk data (PWR and BWR) for this case are shown in Tables 4.35 through 4.37. It should be noted that to achieve further reductions in CDF, the contributions due to non-seismic root causes must be reduced. The principal factors involved that must be reduced are the conditional probabilities for damage of cables and active electro-mechanical components, given that they are wetted by a fire suppressant agent. In this study, some of these values had to be established using zero data point bounding methods, while the remainder are based on very few documented actual damage events. A testing program could better define these conditional probabilities, and in all likelihood result in reduced calculated values for non-seismic root cause contributions to CDF.

For the least vulnerable case generic plant, CDF is calculated to be 1.6E-6/reactor-year with an automatic FPS in the emergency electrical switchgear rooms, and 1.1E-6/reactor-year with no automatic systems installed in the emergency electrical switchgear rooms. For the least vulnerable case PWR, 20 year risk is 2.4 Person-REM with an automatic FPS installed in the emergency electrical switchgear room, and 1.7 Person-REM without. For the least vulnerable case BWR, 20 year risk is 10 Person-REM with an automatic FPS installed in the emergency electrical switchgear room, and 7.1 Person-REM without.

4.10 Specific Plant Analysis Results

The methodology used in the analysis of the generic plant has been applied to specific operating commercial nuclear power plants. The results of the specific analysis are presented here to provide benchmarks against which the generic analysis can be evaluated. Three plant analyses are presented, that for a Westinghouse Pressurized Water Reactor, a Babcock and Wilcox Pressurized Water Reactor, and a General Electric Boiling Water Reactor. Core damage frequency and risk values are presented for the base case. Additionally, sensitivity study results are provided.

4.10.1 Westinghouse Pressurized Water Reactor

A risk evaluation for an operating Westinghouse Pressurized Water Reactor Plant (Ref. 4.1) was conducted to assess the effects of fire protection system actuation on safety-related equipment. The results of the quantification found a total mean contribution to annual core damage frequency of 7.3E-6/reactor-year, and a total 20 reactor-year dose of 6.8 person-REM. The results of the quantification of CDF and risk are presented in Tables 4.38 and 4.39.

4.10.2 Babcock and Wilcox Pressurized Water Reactor

A risk evaluation for an operating Babcock and Wilcox Pressurized Water Reactor (Ref. 4.2) was conducted to assess the effects of fire protection system actuation on safety-related equipment. The results of the quantification found a total mean contribution to annual core damage frequency of 5.6E-5/reactor-year, and a total 20 reactor-year dose of 100 person-REM. The results of the quantification of CDF and risk are presented in Tables 4.40 and 4.41.

Table 4.35

Core Damage Frequency - "Least Vulnerable Case" Plant

Fire Zone:	Cable Spreading Room	Diesel Generator Rooms	Emergency Electrical Switchgear Rooms
FPS Type:	CO_2	Preaction Water	CO_2
Root Cause			
4	2.3E-7		1.5E-8
5	2.6E-8		2.6E-7
6	3.9E-7		2.5E-8
7			
8			
9			
7/8	1.2E-7		1.6E-7
8/9			
7/8/9		<1.0E-8	
11			
12	<1.0E-8		<1.0E-8
13	3.0E-7		2.1E-8
Total	1.1E-6	<1.0E-8	4.8E-7

Total CDF for "Least Vulnerable Cable Case" Generic Plant: 1.6E-6/reactor-year with an automatic FPS installed in the emergency electrical switchgear room, 1.1E-6/reactor-year without.

Table 4.36

Risk (Person-REM) - "Least Vulnerable Case" PWR Plant

Fire Zone:	Cable Spreading Room	Diesel Generator Rooms	Emergency Electrical Switchgear Rooms
FPS Type:	CO_2	Preaction Water	CO_2
Root Cause			
4	1.8E-2		1.1E-3
5	2.0E-3		2.0E-2
6	3.0E-2		1.9E-3
7			
8			
9			
7/8	9.1E-3		1.2E-2
8/9			
7/8/9		7.6E-4	
11			
12	7.6E-4		7.6E-4
13	2.9E-2		1.6E-3
Total	8.2E-2	7.6E-4	3.7E-2

Total risk for "Least Vulnerable Case" PWR Generic Plant: 0.12 Person-REM/reactor-year with an automatic FPS installed in the emergency electrical switchgear room, 0.083 Person-REM/reactor-year without.

Table 4.37

Risk (Person-REM) - "Least Vulnerable Case" BWR Plant

Fire Zone:	Cable Spreading Room	Diesel Generator Rooms	Emergency Electrical Switchgear Rooms
FPS Type:	CO_2	Preaction Water	CO_2
Root Cause			
4	7.5E-2		4.6E-3
5	8.4E-3		7.9E-2
6	1.3E-1		7.6E-3
7			
8			
9			
7/8	3.9E-2		4.9E-2
8/9			
7/8/9		3.1E-3	
11			
12	3.3E-3		3.1E-3
13	9.8E-2		6.4E-3
Total	3.5E-1	3.1E-3	1.5E-1

Total risk for "Least Vulnerable Case" BWR Generic Plant: 0.50 Person-REM/reactor-year with an automatic FPS installed in the emergency electrical switchgear room, 0.36 Person-REM/reactor-year without.

Table 4.38

Westinghouse PWR Summary of Sensitivity Results
in Terms of Core Damage Frequency (Per Reactor Year)**

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Decrease in Probability of a Seismic Fire	Study 3 No Halon Damage
1.	Not applicable for plant under consideration.			
2.	Not applicable for plant under consideration.			
3.	Not applicable for plant under consideration.			
4.	1.4E-6	N/A*	N/A	3.0E-7
5.	1.1E-6	N/A	N/A	5.9E-8
6.	2.1E-6	N/A	N/A	3.9E-7
7.	Not applicable for plant under consideration.			
8.	2.6E-7	3.2E-8	N/A	N/A
9.	Not applicable for plant under consideration.			
10.	Not applicable for plant under consideration.			
11.	4.2E-7	N/A	N/A	1.7E-7
12.	1.4E-6	2.0E-7	2.8E-7	N/A
13.	5.7E-7	N/A	N/A	3.0E-7
Total	7.3E-6	5.9E-6	6.2E-6	2.9E-6

* All entries listed as N/A were not requantified from the base case. Therefore, the total for each sensitivity study can be obtained by using the base case frequency for these entries.

** All entries in this table represent mean values of uncertainty

Table 4.38 (Concluded)

Westinghouse PWR Summary of Sensitivity Results in Terms of Core Damage Frequency (Per Reactor Year)**

Root Cause	Base Case	Study 1 Reduced CO_2 Damage to Cable	Study 2 Barrier Failure = .01	Study 3 All Combined
1.	Not applicable for plant under consideration.			
2.	Not applicable for plant under consideration.			
3.	Not applicable for plant under consideration.			
4.	1.4E-6	1.2E-6	N/A	6.0E-8
5.	1.1E-6	1.1E-6	1.1E-6	1.2E-8
6.	2.1E-6	1.8E-6	N/A	8.0E-87
7.	Not applicable for plant under consideration.			
8.	2.6E-7	2.5E-7	N/A	3.2E-8
9.	Not applicable for plant under consideration.			
10.	Not applicable for plant under consideration.			
11.	4.2E-7	3.0E-7	N/A	3.4E-8
12.	1.4E-6	N/A*	N/A	4.0E-8
13.	5.7E-7	3.3E-7	N/A	6.0E-8
Total	7.3E-6	6.4E-6	7.3E-6	3.2E-7

* All entries listed as N/A were not requantified from the base case. Therefore, the total for each sensitivity study can be obtained by using the base case frequency for these entries.

** All entries in this table represent mean values of uncertainty analysis results given in Appendix A to the plant analysis report.

Table 4.39

Westinghouse PWR Summary of Sensitivity Study Results
in Terms of Risk (Person-REM over 20 Reactor Year)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Decrease in Probability of a Seismic Fire	Study 3 No Halon Damage
1.	Not applicable for plant under consideration.			
2.	Not applicable for plant under consideration.			
3.	Not applicable for plant under consideration.			
4.	5.8E-1	N/A*	N/A	1.2E-1
5.	1.1E-1	N/A	N/A	<0.1
6.	9.3E-1	N/A	N/A	1.4E-1
7.	Not applicable for plant under consideration.			
8.	5.6E-1	<0.1	N/A	N/A
9.	Not applicable for plant under consideration.			
10.	Not applicable for plant under consideration.			
11.	3.8E-1	N/A	N/A	1.7E-1
12.	3.5	0.5	0.7	N/A
13.	6.7E-1	N/A	N/A	3.4E-1
Total	6.8	3.2	3.4	4.8

* N/A reflects no modification from the base case.

** All values listed in table are mean values.

Table 4.39 (Concluded)

Westinghouse PWR Summary of Sensitivity Study Results in Terms of Risk (Person-REM over 20 Reactor Years)

Root Cause	Base Case	Study 1 Reduced CO_2 Damage to Cable	Study 2 Barrier Failure = .01	Study 3 All Combined
1.	Not applicable for plant under consideration.			
2.	Not applicable for plant under consideration.			
3.	Not applicable for plant under consideration.			
4.	5.8E-1	4.9E-1	N/A	1.8E-2
5.	1.1E-1	1.1E-1	1.1E-1	<1.0E-2
6.	9.3E-1	8.0E-1	N/A	3.6E-2
7.	Not applicable for plant under consideration.			
8.	5.6E-1	5.6E-1	N/A	<1.0E-2
9.	Not applicable for plant under consideration.			
10.	Not applicable for plant under consideration.			
11.	3.8E-1	2.5E-1	N/A	4.2E-2
12.	3.5	N/A*	N/A	1.0E-1
13.	5.7E-1	3.8E-1	N/A	4.8E-2
Total	6.8	6.1	6.8	2.4E-1

* N/A reflects no modification from the base case.

** All values listed in table are mean values.

Table 4.40

Babcock and Wilcox PWR Summary of Sensitivity Results in Terms of Core Damage Frequency (Per Reactor Year)**

Root Cause	Base Case	Study 1 EPRI Hazard Case	Study 2 Decrease in Probability of a Seismic/Fire
1.	Not applicable for plant under consideration.		
2.	Not applicable for plant under consideration.		
3.	Not applicable for plant under consideration.		
4.	2.3E-6	N/A*	N/A
5.	Not applicable for plant under consideration.		
6.	1.4E-6	N/A	N/A
7.	Not applicable for plant under consideration.		
8.	1.5E-6	3.1E-8	N/A
9.	<1.0E-8	<1.0E-8	N/A
10.	Not applicable for plant under consideration.		
11.	6.4E-7	N/A	N/A
12.	4.7E-5	1.8E-6	9.4E-6
13.	2.9E-6	N/A	N/A
Total	5.6E-5	9.1E-6	1.7E-5

* All entries listed as N/A were not requantified from the base case. Therefore, the total for each sensitivity study can be obtained by using the base case frequency for these entries.

** All entries in this table represent mean values of uncertainty analysis results given in Appendix A to the plant analysis report.

Table 4.40 (Concluded)

Babcock and Wilcox PWR Summary of Sensitivity Results in Terms of Core Damage Frequency (Per Reactor Year)**

Root Cause	Base Case	EFW Recovery	Reduced Water Damage to Cable	All Combined
1.	Not applicable for plant under consideration.			
2.	Not applicable for plant under consideration.			
3.	Not applicable for plant under consideration.			
4.	2.3E-6	N/A*	4.6E-7	4.6E-7
5.	Not applicable for plant under consideration			
6.	1.4E-6	N/A	2.8E-7	2.8E-7
7.	Not applicable for plant under consideration.			
8.	1.5E-6	N/A	3.0E-7	<1.0E-8
9.	<1.0E-8	N/A	N/A*	<1.0E-8
10.	Not applicable for plant under consideration.			
11.	6.4E-7	N/A	1.3E-7	1.3E-7
12.	4.7E-5	2.4E-5	N/A	1.8E-7
13.	2.9E-6	N/A	5.8E-7	5.8E-7
Total	5.6E-6	3.1E-5	4.9E-5	1.7E-6

* All entries listed as N/A were not requantified from the base case. Therefore, the total for each sensitivity study can be obtained by using the base case frequency for these entries.

** All entries in this table represent mean values of uncertainty analysis results given in Appendix A to the plant analysis report.

Table 4.41

Babcock and Wilcox PWR Summary of Sensitivity Results in Terms of Risk (Person-REM per 20 Reactor Years)**

Root Cause	Base Case	Study 1 EPRI Hazard Case	Study 2 Decrease in Probability of a Seismic/Fire
1.	Not applicable for plant under consideration.		
2.	Not applicable for plant under consideration.		
3.	Not applicable for plant under consideration.		
4.	3.4	N/A*	N/A*
5.	Not applicable for plant under consideration.		
6.	1.9	N/A	N/A
7.	Not applicable for plant under consideration.		
8.	2.6	.05	N/A
9.	0.02	<0.01	N/A
10.	Not applicable for plant under consideration.		
11.	1.0	N/A	N/A
12.	87	3.4	17
13.	4.6	N/A	N/A
Total	100	14	31

* N/A reflects no modification from the base case.

** All values listed in table are mean values.

Table 4.41 (Concluded)

Babcock and Wilcox PWR Summary of Sensitivity Results in Terms of Risk (Person-REM per 20 Reactor Years)

Root Cause	Base Case	EFW Recovery	Reduced FPS Damage to Cable	All Combined
1.	Not applicable for plant under consideration.			
2.	Not applicable for plant under consideration.			
3.	Not applicable for plant under consideration.			
4.	3.4	N/A*	0.68	0.68
5.	Not applicable for plant under consideration			
6.	1.9	N/A	0.38	0.38
7.	Not applicable for plant under consideration.			
8.	2.6	N/A	0.52	0.01
9.	0.02	N/A	N/A*	<0.01
10.	Not applicable for plant under consideration.			
11.	1.0	N/A	0.20	0.20
12.	87	44	N/A	0.34
13.	4.6	N/A	0.92	0.92
Total	100	58	90	2.5

* N/A reflects no modification from the base case.

** All values listed in table are mean values.

4.10.3 General Electric Boiling Water Reactor

A risk evaluation for an operating General Electric Boiling Water Reactor (Ref. 4.3) was conducted to assess the effects of fire protection system actuation on safety-related equipment. The results of the quantification found a total mean contribution to annual core damage frequency of 2.3E-5/reactor-year, and a total 20 reactor-year dose of 137 person-REM. The results of the quantification of CDF and risk are presented in Tables 4.42 and 4.43

Table 4.42

General Electric BWR Summary of Sensitivity Results in Terms of Core Damage Frequency (Per Reactor Year)**

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Decrease in Probability of a Seismic/Fire
1.	5.7E-7	N/A*	N/A
2.	Not applicable for plant under consideration.		
3.	Not applicable for plant under consideration.		
4.	3.3E-7	N/A	N/A
5.	2.3E-8	N/A	N/A
6.	5.4E-7	N/A	N/A
7.	3.3E-7	4.0E-8	N/A
8.	1.1E-5	1.2E-6	N/A
9.	Not applicable for plant under consideration.		
10.	6.9E-7	N/A	N/A
11.	5.7E-7	N/A	N/A
12.	8.6E-6	8.3E-7	1.7E-6
13.	4.4E-7	N/A	N/A
Total	2.3E-5	5.2E-6	1.6E-5

* All entries listed as N/A were not requantified from the base case. Therefore, the total for each sensitivity study can be obtained by using the base case frequency for these entries.

** All entries in this table represent mean values of uncertainty analysis results given in Appendix A to the plant analysis report.

Table 4.42 (Concluded)

General Electric BWR Summary of Sensitivity Study Results in Terms of Core Damage Frequency (Per Reactor Years)

Root Cause	Base Case	Study 1 Reduced CO_2 Damage to Cable	Study 2 Barrier Failure = .01	Study 3 All Combined
1.	5.7E-7	1.1E-7	5.7E-8	1.1E-8
2.	Not applicable for plant under consideration.			
3.	Not applicable for plant under consideration.			
4.	3.3E-7	6.6E-8	N/A	6.6E-8
5.	2.3E-8	<1.0E-8	<1.0E-8	<1.0E-8
6.	5.4E-7	1.1E-7	N/A	<1.0E-8
7.	3.3E-7	6.6E-8	N/A	<1.0E-8
8.	1.1E-5	N/A*	N/A	1.2E-6
9.	Not applicable for plant under consideration.			
10.	6.9E-7	1.4E-7	N/A	1.4E-7
11.	5.7E-7	1.1E-7	N/A	1.1E-7
12.	8.6E-6	N/A	N/A	1.7E-7
13.	4.4E-7	8.8E-8	N/A	8.8E-8
Total	2.3E-5	2.0E-5	2.3E-5	1.9E-6

* All entries listed as N/A were not requantified from the base case. Therefore, the total for each sensitivity study can be obtained by using the base case frequency for these entries.

** All entries in this table represent mean values of uncertainty analysis results given in Appendix A to the plant analysis report.

Table 4.43

General Electric Summary of Sensitivity Study Results in Terms of Risk (Person-REM per 20 Reactor Years)

Root Cause	Base Case	Study 1 EPRI Hazard Curves	Study 2 Decrease in Probability of a Seismic/Fire
1.	3.2	N/A	N/A
2.	Not applicable for plant under consideration.		
3.	Not applicable for plant under consideration.		
4.	1.9	N/A	N/A
5.	0.1	N/A	N/A
6.	3.5	N/A	N/A
7.	2.0	0.2	N/A
8.	63.2	6.9	N/A
9.	Not applicable for plant under consideration.		
10.	4.3	N/A	N/A
11.	3.3	N/A	N/A
12.	53.2	5.1	10.5
13.	2.6	N/A	N/A
Total	137.0	30.8	94.3

* All entries listed as N/A were not requantified from the base case. Therefore, the total for each sensitivity study can be obtained by using the base case frequency for these entries.

** All entries in this table represent mean values of uncertainty analysis results given in Appendix A to the plant analysis report.

Table 4.43 (Concluded)

General Electric BWR Summary of Base Case and Sensitivity Study Results in Terms of Risk (Person-REM per 20 Reactor Years)

Root Cause	Base Case	Study 1 Reduced FPS Damage to Cable	Study 2 Barrier Failure-0.01	Study 3 All Combined
1.	3.2	0.6	0.3	0.1
2.	Not applicable for plant under consideration.			
3.	Not applicable for plant under consideration.			
4.	1.9	0.4	N/A*	0.4
5.	0.1	<0.1	<0.1	<0.1
6.	3.5	0.7	N/A	0.7
7.	2.0	0.4	N/A	<0.1
8.	63.2	N/A	N/A	6.9
9.	Not applicable for plant under consideration.			
10.	4.3	0.9	N/A	0.9
11.	3.3	0.6	N/A	0.6
12.	53.2	N/A	N/A	1.1
13.	2.6	0.5	N/A	0.5
Total	137.0	120.5	120.2	11.2

* N/A reflects no modification from the base case.

** All values listed in table are mean values.

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4.18 Electric Power Research Institute, Seismic Hazard Methodology for the Central and Eastern United States, EPRI NP-4726, Vols. 1-10, July 1986.

5.0 GENERIC COST/BENEFIT ANALYSES

To prevent or mitigate the adverse impacts of advertent or inadvertent actuations of fire protection systems, modifications are possible that could reduce the overall risk of such actuations. To make an assessment of the cost-effectiveness (from a risk-reduction viewpoint) of possible modifications to the plant which would reduce or eliminate the risk increment due to inadvertent and advertent actuations of the fire protection systems, the insights gained from the analyses in References 5.1, 5.2, and 5.3 were utilized and possible plant modifications were identified as presented in Table 5.1. The list of possible plant modifications was analyzed and prioritized. The modifications presented in bold print will have their cost estimated in this report.

The cost analyses are comprehensive and follow the guidelines of NUREG/CR-3568, "A Handbook for Value-Impact Assessment," (Ref. 5.4) and NUREG/CR-4627, Revision 1, "Generic Cost Estimates," (Ref. 5.5). The computer code FORECAST 3.0 (Ref. 5.6), which incorporates this knowledge-based information, was used to develop cost estimates for the proposed plant modifications. The details of the cost estimating methodology are presented in Section 5.1. Generic plant modification cost estimates are presented in Section 5.2. Section 5.3 summarizes the risk calculations for the modifications examined. Finally, Section 5.4 presents the results in terms of cost/benefit.

5.1 Cost Estimating Methodology

This section presents the general methodology employed in preparing cost analyses of potential plant modifications judged to reduce the risk associated with actuations of fire protection systems in vital areas of nuclear power plants.

Cost analyses for the various subtasks required for each proposed plant modification were performed according to standard engineering practices. This involved an initial design evaluation of the plant modification, identification of equipment and materials necessary for the modification, and an assessment of the work areas within the plant in which the proposed modification would take place. All plant cost estimates are presented in 1992 dollars and represent implementation costs for the specific improvements, i.e., one-time cost incurred by the licensee. There are no annual costs, i.e., recurring costs, associated with any of the proposed modifications.

In addition to the cost of physical modifications, the cost analyses include costs for engineering and quality assurance, radiation exposure, health physics support, and radioactive waste disposal. Considered also are licensee costs associated with re-writing operating and testing procedures, staff training, and other technical tasks, as well as costs incurred by the USNRC.

Table 5.1

Potential Generic Cost Analyses

Modification 1:	**Upgrade the FPS with Seismically Qualified Printed Circuit Boards**
Modification 2:	**Replace Smoke Detector Actuated FPS with a Heat Detector Actuated System**
Modification 3:	**Reroute Safety-Related Cables**
Modification 4:	**Seismically Qualify the CO_2 Tank, Outlet Piping and Battery Rack**
Modification 5:	**Seismically Quality an FPS Battery Rack**
Modification 6:	**Upgrade the FPS Water Quality**
Modification 7:	**Replace Deluge FPS with Preaction Sprinkler System**
Modification 8:	**Seal Safety-Related Cabinets to Prevent Water Intrusion**
Modification 9:	**Replace Low Fragility Control Relays with Hardened Relays**
Modification 10:	**Seismically Anchor Electrical Cabinets**
Modification 11:	Provide Fire Wraps for Safety-Related Cabling in a Safety Significant Fire Zone
Modification 12:	Divide Fire Area by Physical Barrier Between Redundant Trains
Modification 13:	Installation of HEPA Filters in Outside Air Intake Duct
Modification 14:	Provide Shielding for Safety-Related Cabling to Prevent Damage from FPS Agents
Modification 15:	Install FPS Requiring Dual-logic Detector Actuation for FPS Agent Release (CO_2 or Halon)
Modification 16:	Install Smoke Detector Actuated Ventilation Barriers
Modification 17:	Seismically Qualify Fire Water System (Piping, Tank and Pumps)
Modification 18:	Upgrade Seismic Characteristics of Ceramic Insulators

Table 5.1

Potential Generic Cost Analyses (Concluded)

Modification 19:	Provide Dedicated Smoke Control and Removal Ventilation System
Modification 20:	Upgrade Barriers to Internal Flooding in Areas Protected by Water-Based FPSs.

5.1.1 Assumptions

The following general assumptions were made in developing cost estimates for the potential plant improvements:

a. Modifications will be made during normal plant operations or during scheduled shutdowns such that, if possible, no additional replacement energy costs will be incurred by the utility.

b. Socio-economic impacts will be considered minimal and will not be included as an increment of cost.

c. Although equipment/component locations for a proposed modification are specific to the reference plant, the environmental factors chosen to estimate worker productivity reductions are appropriate for the entire PWR or BWR plant population.

5.1.2 Cost Categories Considered

All costs were derived by FORECAST 3.0 based on the input provided by the cost analyst. The cost analyses were limited to the following:

a. Licensee Costs

    1. Cost of equipment, materials, and structures

    2. Installation and removal labor cost (where applicable) and associated overhead

    3. Engineering and quality assurance costs

    4. Radiation exposure costs (assuming $1,000/person-REM/benefit screening value)

    5. Health physics support costs

    6. Generated radioactive waste disposal costs (where applicable)

    7. Licensee costs for re-writing procedures, staff training, and other technical subtasks

b. USNRC costs - costs associated with the review of the plant modifications

c. Onsite averted costs - these costs represent the averted onsite property damages, including allowances for cleanup, repair, and replacement energy costs.

The following subsection explains the overall methodology and parameters used by FORECAST 3.0 in the cost derivations.

5.1.3 Key Cost Parameters

5.1.3.1 Labor and Equipment/Materials Costs

The Energy Economic Data Base (EEDB), Reference 5.7 (embedded in the FORECAST code), and R.S. Means Cost Guides (Ref. 5.8) provided the basis for the equipment/material cost and labor estimates. The EEDB incorporates "as-built" cost information (both material unit cost and installation labor hours) for nuclear plant construction activities. The material and labor information from R.S. Means Cost Guides required adjustment to the specialized EEDB basis to properly reflect the nuclear plant level of effort and equipment/material specifications. Two factors, derived for and used in a previous cost study (Ref. 5.9) were employed: Means-EEDB equipment/materials costs were adjusted by multiplying by 2.1 and Means-EEDB labor hours were adjusted by multiplying by 2.7.

Additionally, for operating nuclear power plants there are a number of workplace characteristics which significantly reduce the level of productivity and thus increase the number of labor hours required to accomplish a task. These characteristics, discussed in detail in FORECAST 3.0, include access, congestion and interference, radiation, and task management. Since EEDB reflects only new (or "as-built") plant conditions, the installation labor hours were adjusted to properly consider actual conditions existing at operating nuclear plants.

The total labor costs associated with the proposed modifications include overhead charges (at 100 percent of direct labor) to account for contractor management, administrative support, rent, insurance, etc.

5.1.3.2 Engineering and Quality Assurance/Control Costs

These costs reflect the cost of engineering and design, as well as quality assurance/control (QA/QC) activities associated with implementing the requirements. For requirements affecting structures/systems already in-place (operating plants) the guidelines of Abstract 6.4 of "Generic Cost Estimates," (Ref. 5.5) recommend that a 25 percent engineering and QA/QC factor be applied to the <u>direct</u> cost (i.e., labor and materials cost but without any overhead charges). All cost estimates developed in this study included this engineering and QA/QC cost component.

5.1.3.3. Radiation Exposure

Worker radiation exposure estimates were derived based on guidelines presented in Abstract 5.1 of Reference 5.5. The collective radiation exposure associated with the implementation of a proposed plant modification is estimated by taking the product of the in-field labor hours necessary to perform the task and the work area dose rate associated with that particular task.

In this study the work areas in which the modifications would take place are considered to be either low-dose contaminated areas (cable vault/tunnel) or clean areas (diesel generator rooms). Therefore, radiation exposure is either minimal or zero for the modifications proposed in this study.

5.1.3.4 Health Physics Support Costs

Health physics requirements for the potential plant enhancements were developed based on information and guidelines presented in Abstract 2.1.6 of Reference 5.5. Two factors were considered; the size of the work crew and the magnitude of the radiation field. The plant health physics (HP) personnel perform radiological surveys that are conducted throughout the time required to perform the modification, staff radiological checkpoints, set up anti-contamination clothing removal areas, as well as determine badging requirements.

Some of the modifications are performed in low radiation but contaminated work areas, such as the cable vault/tunnel. Therefore, the health physics support costs are highest for this type of improvement. However, a minimum health physics cost increment is associated even with physical modifications conducted in clean work areas since area radiological surveys and other HP activities still have to be performed.

5.1.3.5 Anti-Contamination Clothing Costs

Cost estimates for anti-contamination (anti-c) clothing used while performing the plant modifications were derived based on Abstract 2.1.5 of Reference 5.5. The cost per suit-up assumes that each member of the work crew requires two complete sets of anti-c clothing per eight-hour shift. Included in the cost per suit-up are the cost of purchasing the anti-c clothing set, its wear-out rate, laundering costs, etc. Only work tasks conducted in contaminated plant areas were considered to include this cost increment.

5.1.3.6 Radioactive Waste Disposal Cost

The costs for disposal of radioactive wastes generated during plant modifications were derived based on guidelines of Abstract 2.1.4 of Reference 5.5. For this study the cost increment associated with the disposal of radioactive wastes is applicable only to those plant modifications that necessitated removal of existing system components located in a contaminated area. The costs are, however, insignificant (less than five percent of total cost).

5.1.3.7 Other Licensee Costs

Other costs incurred by the utility as a result of implementing the proposed physical plant modifications included the costs of re-writing procedures, training the staff (both maintenance and operations), and changing recordkeeping or reporting requirements. For each of the above stated cost categories, the costs were derived following the guidelines presented in Abstracts 2.2.2, 2.2.3, and 2.2.4, respectively, of Reference 5.5. In this study, for some of the plant modifications proposed, these costs represent a significant portion of the total cost.

5.1.4 USNRC Costs

These costs represent USNRC implementation costs. They account for such USNRC activities as developing inspection guidelines and procedures, assuring compliance with the proposed regulatory action, and other technical tasks. In this study, the cost estimates associated with the USNRC were primarily derived from guidelines and input provided by References 5.4 and 5.6.

5.1.5 Onsite Averted Costs

In addition to the costs associated with the modifications, an evaluation of the costs associated with the potential reduction of severe onsite consequences were evaluated. "A Handbook for Value-Impact Assessment" (Ref. 5.4) was used as the reference for performing this evaluation. The values for onsite averted cost were calculated using the following equation:

$$V_{op} = NU\ (F_o - F_n)$$

where

V_{OP} = the cost of avoided onsite property damage

N = the number of affected facilities

U = the present value of onsite property damage given a release

F_O = the original core damage frequency (base case)

F_N = the core damage frequency after implementing an option

and

$$U = \frac{C}{m}\left[\left(e^{-rti}\right)/r^2\right]\left[1-e^{-r(tf-ti)}\right]\left[1-e^{-rm}\right]$$

where

C = cleanup, repair, and replacement power costs

t_f = years remaining until end of plant life

t_i = years before reactor begins operation

m = period of time over which damage costs are paid out

r = discount rate (for 10%, r=0.10)

The cost handbook (Ref. 5.4) recommends best estimate values for input to calculating U (percent value of onsite property damage given a release) as follows:

$C = \$1{,}650 \times 10^6$

$m = 10$ reactor-years

$r = 0.10$

$t_f = 20$ reactor-years

$t_i = 0$ reactor-years

Using the above values for calculating U yields the following result:

Best estimate - $9.0 billion/severe accident event

This value is then applied to the potential change in accident frequency, or for these analyses, change in core damage frequencies for each option.

5.1.6 Cost Estimating Uncertainty

The areas of uncertainty associated with the cost estimating model for this study included the following:

1. Labor rate variations due to plant site location
2. Contingency allowance
3. Variability of in-plant work environment conditions
4. Licensee procedural/administrative/analytical cost
5. NRC procedural/administrative/analytical cost
6. Discount rate variation in the recurring cost module
7. Waste disposal cost module

Each cost estimate was evaluated to determine all areas of uncertainty applicable. Specific numerical value were used for each individual cost analysis. However, the following general assumptions were made:

a. Labor rate variations due to plant site location are considered when calculating labor costs. The assumed labor rate variation was as follows: best estimate is 100 percent of the labor cost, the high cost estimate is 112 percent, and the low cost estimate is 88 percent.

b. The contingency factor provides the user with a means for including an allowance for uncertainty and cost variations at the summary cost level. A contingency percentage can be applied to some or all of the applicable licensee cost categories (i.e., physical modification costs, replacement energy costs, recurring costs and procedural/analytical costs) and for NRC recurring and procedural/analytical costs. A 20 percent contingency factor was assumed applicable to cost categories for the high cost estimate. For the best estimate, a 10 percent contingency factor was used. No contingency factor was applied to the low cost estimate.

c. The working environment characteristics are reflected in the labor productivity factor. Radiation, congestion, and access and handling conditions in the area where the modification is being performed increase the amount of time spent for that task. For the best estimate cost calculation, the labor productivity factor for typical work conditions (as derived by the FORECAST code) were used. A labor productivity factor for an environment reflecting less radiation and congestion was used for the low cost estimate calculation. Conversely, a higher value for the labor productivity factor (an environment reflecting more radiation and congestion) was used for the high cost estimate calculation.

d. The Licensee Procedural/Administrative/Analytical cost module in FORECAST allows modifications to reflect uncertainty in the following cost categories: technical specification change, writing procedures, staff training, recordkeeping and other technical costs. In the case of the technical specification change, writing procedures and recordkeeping the FORECAST 3.0 package allows a change from "simple" to "complex" depending on the plant specific situation. The value developed for the simple change was used for the low cost estimate and the value developed for the complex change was used for the high cost estimate. For both, the staff training and other technical costs, the uncertainty involved varying the number of students or the hours required for the training/evaluation. Training costs will be greater in those plants with additional operators needing training (e.g., five operator teams versus six teams in other plants).

e. The NRC Procedural/Administrative/Analytical cost module in FORECAST allows modifications to reflect uncertainty in the following cost categories: implementation and other technical costs. For the low, best, and high cost estimate calculations, implementation and other technical costs were modified by varying the staff hours and/or hourly rate.

f. In the recurring cost modules for the licensee and the NRC, all future costs considered (e.g., maintenance and testing costs) are discounted to reflect the present value using the high and low discount rates chosen. The defaults used are a 10 percent rate, appropriate for the low cost estimate, and a 5 percent rate, appropriate for the best cost estimate. No present value discounting rate was used for the high cost estimate.

g. The waste disposal cost module has an option to include a surcharge amount to be used in estimating surcharge costs. The default is $40 per cubic foot; however, this amount may be edited by the user.

5.2 Plant Modification Cost Estimates

A discussion of the proposed plant modifications is provided in subsections 5.2.1, through 5.2.8. These modifications were proposed based upon plant walkdowns and engineering judgement. Although the modifications were based upon configurations at specific plants it is anticipated that these modifications can be used as a basis for analyzing the FPSs at all nuclear power plants and possibly reducing risk associated with FPS actuations. Direct costs (labor and materials), indirect charges (overhead, etc.) engineering and QA/QC costs, licensee procedural/analytical costs, and escalation of all costs for each of the modification options are presented. The direct cost data from the R.S. Means Cost Guides (Ref. 5.8) and the Energy Economic Data Base (EEDB), Reference 5.7, are provided in 1989 dollars. These costs are first escalated to 1992 dollars and then overhead at 100 percent of direct labor is added to arrive at total loaded labor and material costs. Engineering and QA/QC costs are computed as 25 percent of the total direct material and labor costs. The low and high cost estimates for a particular option reflect the geographical variation ($\pm$12 percent) of the labor costs, uncertainty in the assessment of labor productivity due to environmental factors, judgements regarding the low and high estimates for licensee procedural and analytical costs, and contingencies in the high estimate for total material costs and NRC costs.

Procedural and analytical costs incurred by the licensee can include any of the following five components:

a. Technical Specification Changes
Based on a review of the proposed modifications, it was assumed for this analysis that technical specification changes would not be required.

b. Writing Procedures
Procedure writing can include routine and/or complex changes to operating, maintenance, surveillance, testing, or other procedures required to prepare the new plant modification for operation and to provide documentation for future personnel use.

c. Staff Training
Where required, the costs for classroom instruction (one hour/student @ $26/hour), on-the-job training (three hours/student @ $8/hour), and the cost for personnel salaries (ten students @ $25/hour) were included in the estimates. Training costs are estimated at about $1600 for the modifications where training was deemed necessary.

d. <u>Recordkeeping</u>
Costs for records management include costs for evaluation of the proposed change, revising procedures or other documents, reviewing the revised documentation, and implementing all modifications to associated plant records. For complicated changes, these costs are much higher (about $2K per modification).

e. <u>Technical/Analytical</u>
For modifications requiring special analyses, the manhours (and engineering rates) for each such analysis were estimated.

For all of the modifications, estimates for security, and fire watch personnel (where required) are provided.

5.2.1 Modification 1 - Upgrade a FPS Actuation Controller with Seismically Qualified Printed Circuit Boards

Because of concerns and industry experience with relay chatter during seismic events it may be prudent to investigate replacing existing relays in the FPS controller cabinets with printed circuit boards. Based on plant specific walkdowns many types of FPS actuation relays were found. For some plants, mercury wetted relays for FPS actuation and/or the annunciation of alarms and isolation of room cooling was found. Given a seismic event, there is a high likelihood of actuation for some of these relay types. The intent of this modification is to replace the relays with printed circuit boards and prevent any damage which may result from inadvertent FPS actuation during a seismic event. For each of the areas modified, the contribution to the core damage frequency from Root Cause 8 (relay chatter in a seismic event) would be eliminated.

For the purposes of this estimate, it is assumed that this modification could be performed during a planned unit outage. Therefore, costs associated with unit shutdown or startup and replacement power costs are not included in the estimate. It is assumed that this type of activity has been done many times before, requiring no learning curve adjustments. No significant radwaste disposal is involved. Also, the costs for security and fire watch personnel are estimated.

The total cost to implement this plant modification ranges from $13,000 to $17,000. Table 5.2 provides a summary of the costs for one FPS controller upgrade while Table 5.3 provides the detailed breakdown.

5.2.2 Modification 2 - Replace Smoke Detector Actuated FPS with a Heat Detector Actuated System

Replacing an existing smoke detector actuated system with a heat detector actuated system will eliminate contributions from Root Cause 1, 7 and 10 which are smoke detector specific. However, to provide an additional detection capability, it may be prudent to leave the existing smoke detectors intact for indication purposes only.

Table 5.2

Cost Summary for Modification 1

Category	Low	Base	High
Hardware and Equipment	4820	5303	6363
Installation Labor	4373	4970	5694
Removal Labor			
Engineering/QA-QC	1752	1947	2336
Anti-C Clothing			
Health Physics			
Radwaste Disposal			
Licensee Procedural	964	1095	1314
Total Licensee Costs	11909	13314	15977
NRC One-Time Costs	627	712	854
TOTAL (Rounded)	13000	14000	17000

Table 5.3 Cost Estimate Worksheet - Modification 1

Modification 1: Seismically Upgrade the FPS Controller

All costs in 1992 dollars

Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
Control Panel	1	ea	2295.49	2.1	16.00	2.2	2.7	63	29.44	1844	4821
Fire Watch	1	ea						25	12.79	320	
Security Guard	1	ea						25	12.79	320	
Cost Summary								113	direct	2485	4821
									overhead	2485	
									total	4970	4821

As with the upgrade of FPS controllers in the last section, it is assumed that this project could be completed during a scheduled outage. No costs associated with shutdown, startup, and replacement power are included. A general set of productivity factors representative of a cable spreading room is used. Costs for security personnel and a fire watch (306 manhours) are included in the total installation labor cost.

The cost to implement this option would range from $78,000 to $105,000. Table 5.4 provides a summary of the cost bases for this modification while Table 5.5 provides a detailed breakdown of the cost estimate.

5.2.3 Modification 3 - Reroute Safety-Related Cables

Through plant walkdowns it has been found that there are certain "pinch points" located in the plant where cabling for certain redundant safety-related systems are run together. Given a fire or FPS actuation that could damage these cables, these safety systems are vulnerable to simultaneous failure. Therefore the intent of this modification is to reroute one of the sets of cabling to remove it from a common failure vulnerability.

For the purposes of this estimate, it is assumed that the old cable run would be abandoned in place and that the new cable installation could be completed during a planned unit outage. Therefore costs associated with unit shutdown or startup and replacement power costs are not included in the estimate. It is further assumed that one length of cable would be required (only control) and that the cable would need to be qualified for harsh environments. Thus, an estimate is made for cable subject to the requirements of the plant equipment qualification program. Costs for cable in conduit are used for all of the cable runs. The total length of cable run in conduit required is assumed to be approximately 500 feet. Ten penetrations are required, and terminations are needed at both ends of the cable run.

Separate environmental and labor productivity factors are used for the two main plant areas. It is assumed that this type of activity has been done many times before, requiring no learning curve adjustments. It is assumed that part of the rerouting will be done in a radiation area and as such appropriate factors and costs for anti-c clothing and HP support are included. No significant radwaste disposal is involved. Also, the costs for security and fire watch personnel are estimated.

The total cost to implement this plant modification ranges from $136,000 to $185,000. Table 5.6 provides a summary of the overall costs for each room and the project total while Table 5.7 provides the detailed breakdown of the cost estimate.

5.2.4 Modification 4 - Seismically Qualify the CO_2 Tank, Outlet Piping, and Battery Rack

A seismic walkdown and subsequent fragility analysis for a typical CO_2 system found a high likelihood of suppressant agent diversion given an earthquake. Failure of the tank or its outlet piping or the FPS battery

Table 5.4

Cost Summary for Modification 2

Category	Low	Base	High
Hardware and Equipment	13091	14400	17280
Installation Labor	46560	52909	63490
Removal Labor			
Engineering/QA-QC	9093	10214	12256
Anti-C Clothing			
Health Physics			
Radwaste Disposal			
Licensee Procedural	7516	8541	10249
Total Licensee Costs	76259	86063	103276
NRC One-Time Costs	1253	1424	1709
TOTAL (Rounded)	78000	87000	105000

Table 5.5 Cost Estimate Worksheet - Modification 2

Modification 2: Replace Smoke Detector Actuated FPS With Heat Detector Actuated System

All costs in 1992 dollars

Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
Fire Alarm Wiring	550	lin ft	1.56	2.1	0.01	3.1	2.7	37	29.44	1094	1804
Conduit	550	lin ft	3.93	2.1	0.18	3.1	2.7	472	29.44	13909	4541
Control Panel	1	ea	2295.49	2.1	16.00	3.1	2.2	63	29.44	1852	4821
Heat Detectors	40	ea	22.93	2.1	1.00	3.1	2.7	193	29.44	5683	1926
Fire Watch	1	ea						153	12.79	1958	
Security Guard	1	ea						153	12.79	1958	
Cost Summary								1072	direct	26454	13091
									overhead	26454	
									total	52909	13091

Table 5.6

Cost Summary for Modification 3

Category	Low	Base	High
Hardware and Equipment	19459	21405	25686
Installation Labor	83679	95090	114108
Removal Labor			
Engineering/QA-QC	15325	17238	20685
Anti-C Clothing			
Health Physics	9234	10493	12591
Radwaste Disposal			
Licensee Procedural	7516	8541	10249
Total Licensee Costs	136466	154190	185029
NRC One-Time Costs	1253	1424	1709
TOTAL (Rounded)	136000	154000	185000

Table 5.7 Cost Estimate Worksheet - Modification 3

Modification 3: Reroute Safety-Related Cable

All costs in 1992 dollars

Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
Control Cable	375	lin ft	2.41	2.1	0.04	3	2.7	71	29.44	2086	1900
Conduit	375	lin ft	14.13	2.1	0.40	3	2.7	709	29.44	20863	11124
Conduit Elbows	4	ea	71.81	2.1	2.00	3	2.7	38	29.44	1113	603
Conduit Hangers	37	ea	5.57	2.1	0.23	3	2.7	40	29.44	1179	433
Wall Penetrations	7	ea	28.75	2.1	3.00	3	2.7	99	29.44	2921	423
Control Cable (rad area)	125	lin ft	2.41	2.1	0.04	3.91	2.7	28	29.44	831	633
Conduit (rad area)	125	ft	14.13	2.1	0.40	3.91	2.7	282	29.44	8311	3708
Conduit Elbows (rad area)	2	ea	71.81	2.1	2.00	3.91	2.7	23	29.44	665	302
Conduit Hangers (rad area)	13	ea	5.57	2.1	0.23	3.91	2.7	17	29.44	497	152
Wall Penetrations (rad area)	3	ea	28.75	2.1	3.00	3.91	2.7	51	29.44	1496	181
Security Guard	1	ea						543	13.97	7584	
Cost Summary								1901	direct	47545	19459
									overhead	47545	
									total	95090	19459

Dominated the overall probability of failure of the CO_2 system. This potential plant modification would seismically qualify the CO_2 tank, battery, and its immediate outlet piping.

For the modification, it is assumed that this project would not require a special plant shutdown to implement. Therefore, shutdown, startup and replacement power costs are not included. Given that there is not real potential for contamination, cost estimates for health physics support and anti-c clothing are not included. Labor hours were included for security and fire watch support.

The total cost for this upgrade ranges from $97,000 to $131,000. Table 5.8 summarizes the overall cost estimate while Table 5.9 provides the detailed cost breakdown.

5.2.5 Modification 5: Seismically Qualify A FPS Battery Rack

It was found during a plant specific walkdown, that a water based FPS for an entire plant had two electric driven fire pumps and one diesel driven fire pump. Given a LOSP, both electric driven pumps would fail due to being powered from non-vital busses. A failure of the starting battery for the diesel pump would lead to a loss of fire main pressure.

For this modification it is assumed that this project would not require a special plant shutdown to implement. Therefore, shutdown, startup and replacement power costs are not included. Given that there is not a real potential for contamination, cost estimates for health physics support and anti-c clothing are not included. Labor hours are included for security and fire watch support.

The total cost for this upgrade ranges from $35,000 to $42,000. Table 5.10 summarizes the overall cost estimate for this modification while Table 5.11 provides the detailed cost breakdown.

5.2.6 Modification 6: Upgrade the FPS Water Quality

Water is the most frequently used FPS agent at nuclear power plants. The most common delivery systems are pre-action sprinklers, deluge wet-pipe sprinklers, dry-pipe sprinklers and standpipe hose systems. It has been postulated that improvement of water quality will reduce the potential for damage to safety-related components from exposure to water suppressant. This is based on the thought that pure water, less conductive than normal fire fighting water, would be less likely to cause short circuits or grounds. To address this concern, a modification to upgrade the FPS water quality is performed. for this modification a water purification system would be required along with the associated piping and valves and a storage tank. It is assumed for this modification that the existing FPS piping and pumps would be utilized. also, it is assumed that this project could be performed during plant operation. Therefore, shutdown, startup and replacement power costs are not included. given that there is not a real potential for contamination,

Table 5.8

Cost Summary for Modification 4

Category	Low	Base	High
Hardware and Equipment	21120	23232	27879
Installation Labor	56338	64019	76824
Removal Labor			
Engineering/QA-QC	12322	13810	16573
Anti-c ClothinG			
Health Physics	4608	5236	6283
Radwaste Disposal			
Licensee Procedural	964	1095	1314
Total Licensee Costs	95351	107393	128872
NRC One-Time Costs	1253	1424	1709
TOTAL (Rounded)	97000	109000	131000

Table 5.9 Cost Estimate Worksheet - Modification 4

Modification 4: Seismically Qualify CO2 Tank, Outlet Piping and Battery Rack

All costs in 1992 dollars Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
New Anchors and Anchor Bolts	1	lot	5689.32	2.1	160.00	2.8	2.7	724	29.44	21300	11948
Replace Battery racks	2	ea	2184.00	2.1	24.00	3.7	2.7	262	29.44	7712	9173
Fire Watch	1	ea						107	13.97	1498	
Security Guard	1	ea						107	13.97	1498	
Cost Summary								1200	direct	32009	21120
									overhead	32009	
									total	64019	21120

Table 5.10

Cost Summary for Modification 5

Category	Low	Base	High
Hardware and Equipment	9173	10090	12108
Installation Labor	18725	21278	25534
Removal Labor			
Engineering/QA-QC	4634	5182	6219
Anti-C Clothing			
Health Physics			
Radwaste Disposal			
Licensee Procedural	964	1095	1314
Total Licensee Costs	33495	37646	45175
NRC One-Time Costs	1253	1424	1709
TOTAL (Rounded)	35000	39000	47000

Table 5.11 Cost Estimate Worksheet - Modification 5

Modification 5: Seismically Qualify FPS Battery Rack

All costs in 1992 dollars

Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
Replace Battery racks	2	ea	2184.00	2.1	24.00	3.7	2.7	262	29.44	7712	9173
Fire Watch	1	ea						105	13.97	1464	
Security Guard	1	ea						105	13.97	1464	
Cost Summary								472	direct	10639	9173
									overhead	10639	
									total	**21278**	**9173**

costs estimates for health physics support and anti-c clothing are not included. Labor hours are included for security and fire watch support.

The total cost for this upgrade ranges from $1,174,000 to $1,577,000. Table 5.12 summarizes the overall cost estimate for this modification while Table 5.13 provides the detailed cost breakdown.

5.2.7 Modification 7: Replace Deluge with Preaction Sprinkler FPS

Of all of the types of water based FPSs utilized in nuclear power plants the pre-action sprinkler system has been found to experience the least number of inadvertent actuations (Appendix A). A pre-action sprinkler system requires the opening of a deluge value, either automatically, with a control signal, or manually, and a rise in the ambient temperature to the melting point of the fusible links on the sealed sprinkler heads. A deluge FPS only requires the opening of a valve, either automatically or manually to discharge water under pressure to the open spray heads. This modification involves replacing a deluge FPS with a pre-action sprinkler FPS. This modification would reduce the frequency of inadvertent actuations and localize the application of FPS agent.

for this modification fusible link sealed sprinkler heads would need to be added to the existing deluge FPS. All of the existing hardware would be kept in place. It is also assumed the existing locations of sprinkler heads would be adequate for preaction system. This modification will not require a special plant shutdown to implement. Therefore, shutdown, startup and replacement power costs are not included. Also this modification will not require any health physics support or anti-c clothing. Labor hours are included for security and fire watch support.

The total cost for this upgrade ranges from $22,000 to $30,000. Table 5.14 summarizes the overall cost estimate for this modification while Table 5.15 provides the detailed cost breakdown.

5.2.8 Modification 8: Replace Electrical Cabinet with a Cabinet Designed to Prevent Water Intrusion

One study (ref. 5.10) has reported that electrical equipment failure modes related to water were mainly due to electrical shorting and long term corrosion. Wherever water intrusion is possible the equipment could be expected to fail through shorting, grounding, tripping of overcurrent devices, physical damage due to the velocity of direct hose streams or long term corrosion causing potential failure of electromechanical parts. These failure modes were found to be dependent on the national electrical manufacturers association (NEMA) rating or the configuration of the electrical enclosure. The appropriate NEMA rating to preclude water intrusion can potentially eliminate failures in electrical equipment due to water based FPSs. Enclosures that have a NEMA rating of 1 and 5 are subject to water intrusion under all water spray conditions applicable for this study. Enclosures with NEMA ratings of 2, 3, 3R, 3S, 4, 4S, 6, 6P, 11, 12, 12K and 13 are expected to prevent water intrusion

Table 5.12

Cost Summary for Modification 6

Category	Low	Base	High
Hardware and Equipment	434327	477759	573311
Installation Labor	553102	628524	754229
Removal Labor			
Engineering/QA-QC	177719	198005	237607
Anti-C Clothing			
Health Physics			
Radwaste Disposal			
Licensee Procedural	7516	8541	10249
Total Licensee Costs	1172664	1312830	1575396
NRC One-Time Costs	1253	1424	1709
TOTAL (Rounded)	1174000	1314000	1577000

Table 5.13 Cost Estimate Worksheet - Modification 6

Modification 6: Upgrade FPS Water Quality

All costs in 1992 dollars

Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
FPS Water Storage Tank	1	ea	217,942.00	1	5200.00	1	1	5200	29.44	153071	217942
Demineralized Water Syste	1	ea	216,384.80	1	3772.94	1	1	3773	29.44	111063	216385
Security Guard	1	ea						3589	13.97	50127	
Cost Summary								12562	direct	314262	434327
									overhead	314262	
									total	628524	434327

Table 5.14

Cost Summary for Modification 7

Category	Low	Base	High
Hardware and Equipment	754	829	995
Installation Labor	10782	12253	14703
Removal Labor			
Engineering/QA-QC	1536	1739	2087
Anti-c Clothing			
Health Physics			
Radwaste Disposal			
Licensee Procedural	7516	8541	10249
Total Licensee Costs	20589	23362	28035
NRC One-Time Costs	1253	1424	1709
TOTAL (Rounded)	22000	25000	30000

Table 5.15 Cost Estimate Worksheet - Modification 7

Modification 7: Replace Deluge FPS with Pre-action FPS

All costs in 1992 dollars

Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
Sprinkler Heads	32	ea	11.22	2.1	1.00	3.1	2.7	154	29.44	4546	754
Fire Watch	1	ea						62	12.79	790	
Security Guard	1	ea						62	12.79	790	
Cost Summary								278	direct	6126	754
									overhead	6126	
									total	12253	754

under direct hose stream and splashing water spray. Only those enclosures with a NEMA rating of 6 and 6P are expected to prevent water intrusion under temporary submersion due to flooding. The intent of this modification is to replace existing safety-related electrical cabinets and enclosures with NEMA spray-proof rated enclosures. Although the modified electrical cabinets may require internal cooling a cost for internal cooling was not included for this analysis.

Unlike the other generic modifications presented in this section this modification can not be completed during a scheduled outage since safety-related cabinets would need to be de-energized. Therefore, costs associated with shutdown, startup and replacement power are included. Additionally, costs for health physics support and anti-c clothing are included. Costs for security and fire watch are also included. The cost to implement this modification would range from $22,000 to $30,000. Table 5.16 summarizes the overall cost estimate while Table 5.17 provides a detailed cost breakdown.

5.2.9 Modification 9: Replace Low Fragility Control Relays with Hardened Relays

Because of concerns and industry experience with relay chatter during seismic events it may be prudent to investigate replacing existing relays in the FPS controller cabinets with seismically hardened relays. Based on plant specific walkdowns many types of FPS actuation relays were found. For some plants, mercury wetted relays for FPS actuation and/or the annunciation of alarms and isolation of room cooling was found. Given a seismic event, there is a high likelihood of actuation for some of these relay types. The intent of this modification is to replace the relays with seismically hardened relays and prevent any damage which may result from inadvertent FPS actuation during a seismic event. For each of the FPSs modified, the contribution to the core damage frequency from Root Cause 8 (relay chatter in a seismic event) would be eliminated.

For the purposes of this estimate, it is assumed that this modification could be performed during a planned unit outage. Therefore, costs associated with unit shutdown, startup and replacement power are not included in this estimate. It is assumed that his type of modification has been performed many times before, requiring no learning curve adjustments. No significant radwaste disposal is involved. Also, the costs for security and fire watch are included.

The total cost to implement this modification ranges from $13,000 to $17,000. Table 5.18 provides a summary of the cost for the replacement of one relay while Table 5.19 provides the detailed breakdown of the cost.

Table 5.16

Cost Summary for Modification 8

Category	Low	Base	High
Hardware and Equipment	4830	5313	6376
Installation Labor	11151	12671	15206
Removal Labor			
Engineering/QA-QC	2601	2912	3495
Anti-C Clothing			
Health Physics	1389	1578	1894
Radwaste Disposal			
Licensee Procedural	964	1095	1314
Total Licensee Costs	20935	23570	28284
NRC One-Time Costs	1253	1424	1709
TOTAL (Rounded)	22000	25000	30000

Table 5.17 Cost Estimate Worksheet - Modification 8

Modification 8: Replace Electrical Cabinets with NEMA Rated Cabinets to Prevent Water Intrusion

All costs in 1992 dollars

Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
NEMA 4 Electrical Cabinet	1	ea	2300.00	2.1	20.00	2.8	2.7	90	29.44	2663	4830
Removal of Electrical Cabinet	1	ea		2.1	20.00	2.8	2.7	90	29.44	2663	0
Security Guard	1	ea						72	13.97	1011	
Cost Summary								253	direct	6336	4830
									overhead	6336	
									total	12671	4830

Table 5.18

Cost Summary for Modification 9

Category	Low	Base	High
Hardware and Equipment	5250	5775	6930
Installation Labor	2586	2939	3526
Removal Labor			
Engineering/QA-QC	1636	1811	2173
Anti-C Clothing			
Health Physics	938	1065	1279
Radwaste Disposal			
Licensee Procedural	964	1095	1314
Total Licensee Costs	11373	12685	15222
NRC One-Time Costs	1253	1424	1709
TOTAL (Rounded)	13000	14000	17000

Table 5.19 Cost Estimate Worksheet - Modification 9

Modification 9: Replace Low Fragility Relays with Hardened Relays

All costs in 1992 dollars

Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
FPS Relay	1	lot	2500.00	2.1	8.00	2.8	2.7	36	29.44	1065	5250
Fire Watch	1	ea						14	13.97	202	
Security Guard	1	ea						14	13.97	202	
Cost Summary								65	direct	1469	5250
									overhead	1469	
									total	2939	5250

5.2.10 Modification 10: Seismically Anchor Safety-Related Cabinets Susceptible to Tipping/Sliding Failure

In a seismic event energized cabinets may present a potential source for fire. Although it is assumed that in a seismic event offsite power will be lost, thus deenergizing many electrical cabinets, there will be a number of safety-related cabinets energized by alternative power sources (batteries, diesel generators). These energized cabinets are susceptible to tipping/sliding failure possibly leading to a fire.

It is assumed that this modification would eliminate the potential for seismically induced fires due to the tipping/sliding failure of an energized electrical cabinet. For this modification a special plant shutdown will not be required. Therefore, shutdown, startup and replacement power costs are not included. Additionally, it is assumed there is no real potential for contamination so that cost estimates for health physics support and anti-c clothing are not included. Labor hours are included for security watch suppport.

The total cost for this modification ranges from $67,000 to $91,000. Table 5.20 summarizes the overall cost estimate for this modification while Table 5.21 provides the detailed cost breakdown.

5.3 Cost/Benefit Analysis

The cost and risk estimates presented in Section 5.2 and 4.7, respectively are used to calculate the dollar-to-person-REM averted ratio (DPR) for the proposed generic plant modifications. This section provides the results of the cost/benefit analysis applied to both a generic PWR and BWR. Tables 5.22 through 5.30 provide the results for each of the proposed generic plant modifications. These tables present the cost/benefit results in $K/person-REM with and without onsite averted costs (OSAC). The remaining plant operational lifetime for this analysis was assumed to be 20 years.

5.3.1 Modification 1 - Upgrade an FPS Actuation Controller with Seismically Qualified Printed Circuit Boards

A cost for this modification of $14K (one FPS controller) was estimated. This modification is assumed to eliminate any contribution of risk from Root Cause 8 or any combination of Root Cause 8 scenarios. It is assumed that for the cable spreading room (CSR) one FPS controller will be replaced and for the emergency diesel generator (EDG) and emergency switchgear (ESGR) areas two FPS controllers will be replaced.

Table 5.22 presents the cost/benefit results for this modification. Even without OSAC included this option appears to be beneficial for EDG areas with a deluge or CO_2 system and for the ESGR with a deluge FPS. When OSAC are included a Halon FPS in the CSR for both a PWR and BWR and a deluge and CO_2 FPS in the CSR for a BWR appear to be beneficial.

Table 5.20

Cost Summary for Modification 10

Category	Low	Base	High
Hardware and Equipment	11948	13142	15771
Installation Labor	44603	50686	60823
Removal Labor			
Engineering/QA-QC	8562	9621	11546
Anti-C Clothing			
Health Physics			
Radwaste Disposal			
Licensee Procedural	964	1095	1314
Total Licensee Costs	66077	74544	89543
NRC One-Time Costs	1253	1424	1709
TOTAL (Rounded)	67000	76000	91000

Table 5.21 Cost Estimate Worksheet - Modification 10

Modification 10: Seismically Anchor Electrical Cabinet

All costs in 1992 dollars Material Description	Quantity	Unit	Material Unit Cost	Material EEDB Factor	Unit Labor Hours	Labor Prod Factor	Labor EEDB Factor	Total Hours Labor	Labor Rates	Labor Cost	Material Cost
New Anchors and Anchor Bolts	1	lot	5689.32	2.1	160.00	2.8	2.7	724	29.44	21300	11948
Security Guard	1	ea						289	13.97	4042	
Cost Summary								1013	direct	25343	11948
									overhead	25343	
									total	50686	11948

TABLE 5.22 COST/BENEFIT RATIO
MODIFICATION 1
($K/PERSON-REM AVERTED)

	Preaction	Deluge	Wetpipe	CO2	Halon
PWR					
CSR	6.7E+02	7.8E+00		7.0E+00	6.4E+00
EDG	2.5E+03	1.9E-01		9.0E-01	
ESGR		1.0E+00		1.1E+01	2.2E+01
CSR	**6.6E+02**	**2.3E+00**		**1.6E+00**	**1.0E+00**
EDG	**2.5E+03**	**<1.0E+00**		**<1.0E+00**	
ESGR		**<1.0E+00**		**5.2E+00**	**1.6E+01**
BWR					
CSR	1.6E+02	1.9E+00		1.8E+00	1.6E+00
EDG	6.7E+02	4.9E-02		2.3E-01	
ESGR		2.5E-01		2.5E+00	5.6E+00
CSR	**1.6E+02**	**5.5E-01**		**4.0E-01**	**2.6E-01**
EDG	**6.7E+02**	**<1.0E+00**		**<1.0E+00**	
ESGR		**<1.0E+00**		**1.2E+00**	**4.2E+00**

Bold type indicates OSAC included.

5.3.2 Modification 2 - Replace Smoke Detector Actuated FPS with a Heat Detector Actuated System

A cost for this modification of $87K was estimated. This modification was proposed only for the cable spreading room and would eliminate any incremental contribution from Root Cause 7 to core damage frequency from this area. The risk reduction for this modification is 45 person-REM for the PWR analyzed and 180 person-REM for the BWR analyzed. Table 5.23 presents the cost/benefit results for this modification. If OSAC are not included, this modification appears to be beneficial only for the BWR analyzed. If OSAC are included both the PWR and BWR analyzed indicate a beneficial option.

5.3.3 Modification 3 - Reroute Safety-Related Cables

A cost for this modification of $154K was estimated. This modification is assumed to apply only to the cable spreading room and was assumed to reduce risk in the CSR by an order of magnitude. If OSAC are not included this modification appears to only be beneficial for a deluge FPS in a BWR. With OSAC included, a deluge FPS in both a PWR and a BWR and a wetpipe and preaction FPS in a BWR indicate a beneficial modification. Table 5.24 presents the cost/benefit results for this modification.

5.3.4 Modification 4 - Seismically Qualify the CO_2 Tank, Outlet Piping and Battery Rack

For this modification it was assumed that there is one common CO_2 tank per plant. A cost for this modification of $109K was estimated. Implementing this modification eliminates the incremental contribution to core damage frequency from Root Cause 12 (seismic/fire interaction) for CO_2 systems. When OSAC are included for the cable spreading room and emergency switchgear room for the BWR analyzed, a beneficial modification is indicated. Table 5.25 presents the cost/benefit results for this modification.

5.3.5 Modification 5 - Seismically Qualify A FPS Battery Rack

The cost for this modification was estimated to be $39K. This modification is assumed to eliminate Root Cause 12 core damage frequency contributions from water-based FPSs. Table 5.26 presents the cost/benefit results for this modification. For the BWR examined a beneficial modification is indicated for preaction, deluge and wetpipe FPSs in the cable spreading room. For the emergency switchgear room a beneficial modification is indicated for deluge and wetpipe FPS. If OSAC are included all water-based FPSs considered (both PWR and BWR) would indicate a beneficial modification.

TABLE 5.23 COST/BENEFIT RATIO
MODIFICATION 2
($K/PERSON-REM AVERTED)

	Preaction	Deluge	Wetpipe	CO2	Halon
PWR					
CSR		1.9E+00			
EDG					
ESGR					
CSR		**<1.0E+00**			
EDG					
ESGR					
BWR					
CSR		4.8E-01			
EDG					
ESGR					
CSR		**<1.0E+00**			
EDG					
ESGR					

Bold type indicates OSAC included.

TABLE 5.24 COST/BENEFIT RATIO
MODIFICATION 3
($K/PERSON-REM AVERTED)

	Preaction	Deluge	Wetpipe	CO2	Halon
PWR					
CSR EDG ESGR	1.1E+01	3.0E+00	1.1E+01	1.5E+01	4.0E+01
CSR EDG ESGR	**3.2E+00**	**<1.0E+00**	**3.2E+00**	**4.8E+00**	**2.1E+01**
BWR					
CSR EDG ESGR	2.8E+00	7.4E-01	1.9E+00	2.9E+00	5.4E+00
CSR EDG ESGR	**7.9E-01**	**<1.0E+00**	**5.3E-01**	**9.5E-01**	**2.8E+00**

Bold type indicates OSAC included.

TABLE 5.25 COST/BENEFIT RATIO
MODIFICATION 4
($K/PERSON-REM AVERTED)

	Preaction	Deluge	Wetpipe	CO2	Halon
PWR					
CSR				7.3E+00	
EDG					
ESGR				6.8E+00	
CSR				**1.8E+00**	
EDG					
ESGR				**1.5E+00**	
BWR					
CSR				1.8E+00	
EDG					
ESGR				1.8E+00	
CSR				**4.4E-01**	
EDG					
ESGR				**4.0E-01**	

Bold type indicates OSAC included.

TABLE 5.26 COST/BENEFIT RATIO
MODIFICATION 5
($K/PERSON-REM AVERTED)

	Preaction	Deluge	Wetpipe	CO2	Halon
PWR					
CSR	2.0E+00	2.0E+00	2.0E+00		
EDG					
ESGR		**<1.0E+00**	**<1.0E+00**		
CSR					
EDG					
ESGR					
BWR					
CSR	4.8E-01	4.8E-01	4.8E-01		
EDG					
ESGR		**<1.0E+00**	**<1.0E+00**		
CSR					
EDG					
ESGR					

Bold type indicates OSAC included.

5.3.6 Modification 6 - Upgrade the FPS Water Quality

A cost for this modification of $1314K was estimated. The magnitude of risk reduction that would be achieved by improvement of water quality is not clear. Therefore a cost/benefit ratio was not calculated. Further study on the effect of water-based FPSs on safety-related equipment and the improvement of the FPS water quality and its effect may provide the data necessary to examine this issue quantitatively.

5.3.7 Modification 7 - Replace Deluge with Preaction Sprinkler FPS

This modification applies to all three plant areas examined and would require one FPS replacement in the CSR and two in both the EDG and the ESGR area. However, a cost/benefit was not performed for the ESGR since a preaction FPS was not part of the configurations studied based on Appendix D. The cost for this modification was estimated to be $25K for one plant area. The reduction in risk associated with this modification is the difference in risk between a deluge FPS and a preaction FPS. A beneficial modification is indicated for both the cable spreading room and the emergency diesel generator area without OSAC included for both the PWR and BWR examined. Table 5.27 presents the cost/benefit results for this modification.

5.3.8 Modification 8 - Replace Electrical Cabinet with a Cabinet Designed to Prevent Water Intrusion

This modification was examined on a plant area basis assuming two cabinets in each of two EDG areas, two cabinets in the CSR and ten cabinets in each of two ESGR areas. These numbers were estimated based on plant walkdowns and may vary for specific plants. The cost for this modification was estimated to be $25K for one cabinet replacement. The reduction in risk associated with this modification is the elimination of all water-based core damage frequency contributions except for Root Cause 12. For the deluge FPS this modification appears to be beneficial in the cable spreading room and emergency diesel generator areas for both the PWR and BWR examined without OSAC included. If OSAC are included, this modification also appears to be beneficial for a wetpipe FPS in an EDG area in a BWR. The results of the cost/benefit are presented in Table 5.28.

TABLE 5.27 COST/BENEFIT RATIO
MODIFICATION 7
($K/PERSON-REM AVERTED)

	Preaction	Deluge	Wetpipe	CO2	Halon
PWR					
CSR		4.5E-01			
EDG		3.3E-01			
ESGR					
CSR		**<1.0E+00**			
EDG		**<1.0E+00**			
ESGR					
BWR					
CSR		1.1E-01			
EDG		8.8E-02			
ESGR					
CSR		**<1.0E+00**			
EDG		**<1.0E+00**			
ESGR					

Bold type indicates OSAC included.

TABLE 5.28 COST/BENEFIT RATIO
MODIFICATION 8
($K/PERSON-REM AVERTED)

	Preaction	Deluge	Wetpipe	CO2	Halon
PWR					
CSR	8.1E+02	9.1E-01	1.6E+02		
EDG	9.1E+03	6.7E-01	3.4E+03		
ESGR		1.3E+01	1.7E+02		
CSR	**8.0E+02**	**<1.0E+00**	**1.5E+02**		
EDG	**9.1E+03**	**<1.0E+00**	**3.4E+03**		
ESGR		**6.8E+00**	**1.6E+02**		
BWR					
CSR	2.0E+02	2.3E-01	1.3E+00		
EDG	2.4E+03	1.8E-01	1.0E+03		
ESGR		3.2E+00	4.2E+01		
CSR	**2.0E+02**	**<1.0E+00**	**1.3E+00**		
EDG	**2.4E+03**	**<1.0E+00**	**1.8E-01**		
ESGR		**1.7E+00**	**4.0E+01**		

Bold type indicates OSAC included.

5.3.9 Modification 9 - Replace Low Fragility Control Relays with Hardened Relays

A cost for this modification of $14K (one FPS relay) was estimated. This modification is assumed to eliminate any contribution of risk from Root Cause 8 or any combination of Root Cause 8 scenarios. It is assumed that for the cable spreading room (CSR) one FPS relay will be replaced and for the EDG and ESGR areas two FPS relays will be replaced.

Table 5.29 presents the cost/benefit results for this modification. Even without OSAC included this option appears to be beneficial for EDG areas with a deluge or CO_2 system and for the ESGR with a deluge FPS. When OSAC are included, a Halon FPS in the CSR for both a PWR and a BWR and a deluge and CO_2 FPS in the CSR for a BWR appear to be beneficial.

5.3.10 Modification 10 - Seismically Anchor Safety-Related Cabinets Susceptible to Tipping/Sliding Failure

This modification applies to the CSR and ESGR areas and the cost appropriately reflects the number of electrical cabinets in each area as discussed in Section 5.3.9. It is recognized that some of the cabinets may already be seismically anchored, but for the purposes of this estimate it is assumed that all cabinets will require the seismic anchoring modification. The cost for seismically anchoring one cabinet is estimated to be $76K. The reduction in risk associated with this modification is the elimination of Root Cause 12 scenarios in the CSR and ESGR areas. Table 5.30 presents the cost/benefit results for this modification. This modification appears to be beneficial when OSAC are included for a preaction, deluge and wetpipe FPS in the CSR for the BWR examined.

5.4 Frontfit Analysis

The plant modifications presented in the cost/benefit analysis as part of the Generic Issue 57 generic plant analysis were intended to be backfits for existing plants and as such these proposed plant modifications were determined from the insights provided by the three individual plant analyses and the generic plant analysis. However, some of these plant modifications may be considered as frontfits as part of the Advanced Light Water Reactor (ALWR) design program. All of the modifications considered as frontfits would avoid cost contributions for health physics support, radiation exposure, waste disposal and removal labor. Additionally, cost contributions for licensee (procedural) and NRC costs would differ. The following subsections specifically discuss the proposed plant modifications, as applicable, for frontfits. Licensee and NRC costs were not recalculated and are not expected to differ significantly from the backfit estimates. Table 1 presents the entire list of potential plant modifications analyzed as backfits as part of GI-57 with the modifications that were analyzed presented in bold type.

TABLE 5.29 COST/BENEFIT RATIO
MODIFICATION 9
($K/PERSON-REM AVERTED)

	Preaction	Deluge	Wetpipe	CO2	Halon
PWR					
CSR	6.7E+02	7.8E+00		7.0E+00	6.4E+00
EDG	2.5E+03	1.9E-01		9.0E-01	
ESGR		1.0E+00		1.1E+01	2.2E+01
CSR	**6.6E+02**	**2.3E+00**		**1.6E+00**	**1.0E+00**
EDG	**2.5E+03**	**<1.0E+00**		**<1.0E+00**	
ESGR		**<1.0E+00**		**5.2E+00**	**1.6E+01**
BWR					
CSR	1.6E+02	1.9E+00		1.8E+00	1.6E+00
EDG	6.7E+02	4.9E-02		2.3E-01	
ESGR		2.5E-01		2.5E+00	5.6E+00
CSR	**1.6E+02**	**5.5E-01**		**4.0E-01**	**2.6E-01**
EDG	**6.7E+02**	**<1.0E+00**		**<1.0E+00**	
ESGR		**<1.0E+00**		**1.2E+00**	**4.2E+00**

Bold type indicates OSAC included.

TABLE 5.30 COST/BENEFIT RATIO
MODIFICATION 10
($K/PERSON-REM AVERTED)

	Preaction	Deluge	Wetpipe	CO2	Halon
PWR					
CSR	7.6E+00	7.6E+00	7.6E+00	1.0E+01	2.8E+01
EDG					
ESGR		7.6E+01	7.6E+01	9.5E+01	2.2E+02
CSR	**2.2E+00**	**2.2E+00**	**2.2E+00**	**4.7E+00**	**2.3E+01**
EDG					
ESGR		**7.1E+01**	**7.1E+01**	**9.0E+01**	**2.1E+02**
BWR					
CSR	1.9E+00	1.9E+00	1.9E+00	2.5E+00	6.9E+00
EDG					
ESGR		2.1E+01	2.1E+01	2.5E+01	6.1E+01
CSR	**5.4E-01**	**5.4E-01**	**5.4E-01**	**1.1E+00**	**5.6E+00**
EDG					
ESGR		**1.9E+01**	**1.9E+01**	**2.4E+01**	**5.9E+01**

Bold type indicates OSAC included.

5.4.1 Modification 1 - Upgrade a FPS Actuation Controller with Seismically Qualified Printed Circuit Boards

Because of concerns and industry experience with relay chatter during seismic events it may be prudent to investigate replacing existing relays in the FPS controller cabinets with printed circuit boards. Based on plant specific walkdowns many types of FPS actuation relays were found. For some plants, mercury wetted relays for FPS actuation and/or the annunciation of alarms and isolation of room cooling was found. Given a seismic event, there is a high likelihood of actuation for some of these relay types. The cost determined for this design is for one FPS actuation controller and would eliminate Root Cause 8 (relay chatter in a seismic event) contributions in the area of installation only. If this modification is performed as a frontfit it would be included as part of the overall FPS design. However, it is assumed that the costs associated with including this relay type as part of a new design would be similar to that of the backfit costs minus the costs of the original relay. The total cost of this relay ranges from $13,000 to $17,000.

5.4.2 Modification 2 - Replace Smoke Detector Actuated FPS with a Heat Detector Actuated FPS

Designing a FPS to actuate on heat detectors rather than smoke detectors will eliminate contributions from Root Cause 1, 7 and 10 which are smoke detector specific. The cost for this system as a frontfit would be similar to the costs estimated for the components considered as part of an existing plant backfit. However, the costs in the new plant FPS utilizing heat detectors will also include all of the other associated FPS hardware. The cost to implement this option would range from $78,000 to $105,000.

5.4.3 Modification 3 - Reroute Safety-Related Cables

The intent of this modification is to reroute one set of redundant cabling to remove it from a common failure vulnerability. This plant modification would not be considered as a frontfit and would be proposed as part of the new plant design.

5.4.4 Modification 4 - Seismically Qualify the CO_2 Tank, Outlet Piping, and Battery Rack

A seismic walkdown and subsequent fragility analysis for a typical CO_2 system found a high likelihood of suppressant agent diversion given an earthquake. Failure to the tank or its outlet piping or the FPS battery dominated the overall probability of failure of the CO_2 system. This potential plant system design would seismically qualify the CO_2 tank, battery, and its immediate outlet piping. The overall costs for this frontfit would not differ from the backfit costs significantly. The total cost for this design ranges from $97,000 to $131,000.

5.4.5 Modification 5: Seismically Qualify a FPS Battery Rack

The cost for this system as a frontfit would be similar to the costs estimated for an existing plant backfit. The total cost for this upgrade ranges from $35,000 to $47,000.

5.4.6 Modification 6: Upgrade the FPS Water Quality

Water is the most frequently used FPS agent at nuclear power plants. The most common delivery systems are pre-action sprinklers, deluge wet-pipe sprinklers, dry-pipe sprinklers and standpipe hose systems. It has been postulated that improvement of water quality will reduce the potential for damage to safety-related components from exposure to water suppressant. To address this concern, a plant design to upgrade the FPS water quality is considered. For this design a water purification system would be required along with the associated piping and valves and a storage tank. Additional costs to be considered for a frontfit would be the FPS sprinkler piping and sprinkler heads. The total cost for this system design would range from $1,174,000 to $1,577,000.

5.4.7 Modification 7: Replace Deluge with Preaction Sprinkler FPS

This modification would not be considered a frontfit, but part of an overall FPS design. However, it is anticipated that for the components of the FPS analyzed as part of the backfit analysis utilized as part of the new FPS design the costs will be similar.

5.4.8 Modification 8: Replace Electrical Cabinet with a Cabinet Designed to Prevent Water Intrusion

One study has reported that electrical equipment failure modes related to water were mainly due to electrical shorting and long term corrosion. Wherever water intrusion is possible the equipment could be expected to fail through shorting, grounding, tipping of overcurrent devices, physical damage due to the velocity of direct hose steams or long term corrosion causing potential failure of electromechanical parts. These failure modes were found to be dependent on the National Electrical Manufactures Associated (NEMA) rating or the configuration of the electrical enclosure. The appropriate NEMA rating to preclude water intrusion can potentially eliminate failures in electrical equipment due to water based FPSs. Enclosures that have NEMA rating of 1 and 5 are subject to water intrusion under all water spray conditions applicable for this study. Enclosures with NEMA ratings of 2, 3, 3R, 3S, 4, 4S, 6, 6P, 11, 12, 12K and 13 are expected to prevent water intrusion under direct hose stream and splashing water spray. Only those enclosures with a NEMA rating of 6 and 6P are expected to prevent water intrusion under temporary submersion due to flooding. The intent of this modification is to replace existing safety-related electrical cabinets and enclosures with NEMA spray-proof rated enclosures.

The cost for this system as a frontfit would be similar to the costs estimated for an existing plant backfit with the exception of the labor for the removal of old electrical cabinets. The cost to implement this modification as a frontlift would range from $22,000 to $30,000 per cabinet.

5.4.9 Modification 9: Replace Low Fragility Control Relays with Hardened Relays

Because of concerns and industry experience with relay chatter during seismic events it may be prudent to investigate existing relays in the FPS controller cabinets with seismically hardened relays. Based on plant specific walkdowns many types of FPS actuation relays were found. For some plants, mercury wetted relays for FPS actuation and/or the annunciation of alarms and isolation of room cooling were found. Given a seismic event, there is a high likelihood of actuation for some of these relay types. The intent of this modification is to replace the relays with seismically hardened relays and prevent any damage which may result from inadvertent FPS actuation during a seismic event. For the FPSs designed with these relays, the contribution to the core damage frequency from Root Cause 8 (relay chatter in a seismic event) would be eliminated. Although in a new plant design, the relay would be part of the overall FPS design (and cost) the specific relay cost would be similar as a backfit or part of the new design. The total cost to implement this modification as a frontfit would range from $13,000 to $17,000.

5.4.10 Modification 10: Seismically Anchor Safety-Related Cabinets Susceptible to Tipping/Sliding Failure

In a seismic event energized cabinets present a potential source for fire. Although it is assumed that in a seismic event offsite power will be list, thus deenergizing many electrical cabinets, there will be a number of safety-related cabinets energized by alternative power sources (batteries, diesel generators). These energized cabinets are susceptive to tipping/sliding failure possibly leading to a fire. It is assumed that this modification would eliminate the potential for seismically induced fires due to the tipping/sliding failure of an energized electrical cabinet. The cost for this system as a frontfit would be similar to the costs estimated for an existing plant backfit. The total cost for this design ranges from $67,000 to $91,000.

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6.0 TECHNICAL INSIGHTS FOR THE ADVANCED LIGHT WATER REACTOR (ALWR)

6.1 Introduction

This chapter provides insights with regard to potential risk associated with fire protection system actuation for the ALWR design. These insights stem from the experience base developed from the detailed study of four operating light water reactor designs, as well as the study of a generic light water plant developed in Chapter 4 of this report. These insights are provided in two parts; first, considerations for conducting an analysis on the ALWR design, similar to the specific plant analysis conducted on individual plants. Second, insights gained on specific plant designs found on plant walkdowns and detailed analyses.

6.2 FPS Risk Analysis on ALWR Design

There are many design choices in the ALWR overall plant design that could be made to minimize risks associated with the effects of fire protection system actuation on safety-related equipment. In order to understand the choices, and the impact on risk of these choices, the designer would require the development of a probabilistic risk assessment for the plant, and a vital area analysis associated with the 13 root causes for risk identified in this report. In lieu of plant walkdowns, a detailed review of installation drawings would be required to assess cable routing paths, fire-source-to-critical-cable distances, equipment locations, fire suppressant spray nozzle locations and coverage, etc. The kind of design choices that could be made, given the analytic tools available, include:

- Type of fire suppression systems to be installed.

- Suppressant agents to be used.

- Type of fire suppressant control system sensors to be installed.

- Location and routing of cables for critical safety-related equipment.

- Physical or fire barrier separation of potential seismic/fire sources (e.g., cabinets remaining energized on a LOSP accident sequence) from safety-related cables and other active electromechanical equipment.

- Paths for migration of heat and smoke from one fire zone to another.

- Susceptibility of FPS control systems to actuation from heat/moisture created from high energy fluid system leaks or dust raised during a seismic event.

- Susceptibility of FPS control systems to actuation from smoke from fires external to the power plant.

- Minimizing the potential for suppressant diversion by specifying seismically qualified designs for:

  a. Fire suppression agent storage and distribution systems (pipes, pumps, cylinders, tanks, and valves).

  b. Fire suppressant agent dispensing nozzles (design clearance from structural members and other components).

- Minimizing the potential for unintended system actuation by specifying seismically qualified designs for FPS control systems (relays).

- Interaction between FPS system controls and emergency power generation system controls.

- Prevention of suppressant agent intrusion into active electro-mechanical components, electrical/electronic panels and cabinets, and cable junction boxes by appropriate selection of NEMA rated enclosures.

- Purity of water when used as a suppressant agent.

6.3 Technical Insights from Specific and Generic Plant Analyses

6.3.1 Suppressant Diversion

The potential for fire suppressant diversion was found in the analysis of individual plants. All of these sources related to diversion resulting from a seismic event. Details of the diversion scenarios follow:

- A water FPS system was found that was supplied by two fire pumps, one electrically driven and the other driven by a diesel engine. The electric pump power source was non-vital power, and thus was lost in a seismic sequence involving LOSP. The diesel driven pump starting system power supply consisted of a set of heavy duty lead-acid batteries that provided 12 volt DC starting power to the engine starter motor. However, the batteries were located on a metal storage rack, that was weakly anchored to the concrete floor. The batteries were not fastened in any way to the storage rack. A detailed analysis was performed on the rack, with the conclusion that, given a seismic event, there was a high likelihood that the rack would collapse and the batteries would fall off, with the consequent spilling of electrolyte and breaking of the intercell connecting cables. This would result in loss of starting power for the diesel pump. Thus in a seismic event, the fire main would not remain pressurized. At this plant, water was the agent used in the FPSs for the cable spreading room, the emergency diesel generator rooms, and many other areas.

- A CO_2 FPS was found that was supplied by a common tank that was not seismically mounted. The batteries that supplied power to the tank outlet valve were weakly anchored, and the shelf on which the batteries rested had no restraints on the ends of the shelf. A detailed analysis was performed on the batteries, tank, and outlet piping, with the conclusion that, given a seismic event, there was a high likelihood that either the tank outlet piping would be damaged, or there would be no power available for actuation of the outlet valve. In this plant, CO_2 was the FPS agent for the cable spreading room, the emergency diesel generator rooms, and other plant areas.

- A Halon FPS was found that was supplied by Halon bottles which were not seismically mounted. The bottles were attached to a non-seismically qualified wall by a single metal strap. A detailed analysis was performed on the support straps, with the conclusion that, given a seismic event, there was a high likelihood that the bottle outlet piping would be damaged, and the Halon would not be distributed when demanded. In this plant, the Halon was the suppressant agent for the cable spreading room.

6.3.2 Mercury Relays

Cases of the use of mercury-wetted contact type relays were found in the control systems for fire protection systems in the plants examined in detail. These relays have almost no hardness against seismic actuation. The following examples were found:

- In one plant, it was found that just prior to recently completed modifications, there were mercury actuation relays in the CO_2 FPS control system for the diesel generator rooms. This control system, when actuated, would lock-out the diesel generators as well as isolate the diesel generator room cooling. Therefore, even relatively low acceleration levels had the potential to result in loss of the diesel generators.

- In another plant, even though the actuation relays were not of the mercury type for the High Pressure Coolant Injection (HPCI) pump room FPS control system, auxiliary mercury relays were found in those control circuits that interacted with the room cooling system. In this case even low seismic acceleration levels would isolate cooling to the HPCI room, resulting in the eventual loss of the HPCI pump as the room overheated.

- In another plant, mercury relays were in the actuation circuits for the Halon system that served the control room. Three separate systems protect the subfloor and ceiling areas. An inadvertent release of Halon would increase the noise levels in the control room, and require either donning of emergency breathing apparatus (compounding communications problems) or abandonment of the control room.

6.3.3 Seismic Dust/Smoke Detectors

Smoke detectors signals were found to be used in the FPS actuation control systems in all of the plants examined. These detectors can be expected to actuate on the dust that rises during a seismic event. One case was found where smoke detectors alone could actuate a CO_2 FPS in a cable spreading room. When this single sensor type control system is coupled with a water deluge, CO_2, or Halon system, a seismic event has the potential to lead to inadvertent suppressant release.

6.3.4 Water Deluge Systems

Plant fire protection strategy survey data (Appendix D) indicates that water deluge systems are installed in 44 cable spreading rooms/areas, 4 switchgear rooms, and 10 diesel generator rooms in commercial nuclear power plants. If deluge system spray heads are located such that safety-related components are sprayed by the system, and the cabinets or components sprayed are not sprayproof, then the components may be susceptible to damage.

As an example, for three diesel generator rooms, it was found that critical cabinets with open conduit penetrations on top were located almost directly under spray heads for a preaction system. If this system had been of the deluge type, then in an accident sequence where LOSP occurs (seismic event), and the diesel FPS system was actuated due to relay chatter, station blackout may result from the water damage to active electromechanical components associated with the diesel generators.

In the switchgear rooms in which deluge systems are installed, such installations would have a potential vulnerability to damage in a seismic event unless:

- The FPS control system is seismically qualified, or,
- The cabinets (switchgear) are sealed against water intrusion.

6.3.5 Diesel Generator Controls

In some plants, diesel generator control panels have been found to be co-located. If these panels can be exposed to FPS agent and are not sealed against such an application, then the potential exists for loss of the diesel generators. In a seismic sequence, this would result in station blackout.

6.3.6 Switchgear Fires

In the emergency electrical distribution rooms, while Root Cause 12 (seismic/fire interaction) is a contributor to risk, the fire sources (the switchgear itself) cannot be removed. In some of the switchgear rooms reviewed, all critical cables were routed along the tops of the switchgear, in a position such that large numbers of these cables were vulnerable to a fire in any of the of the cabinet subdivisions. To reduce the potential risk associated with these areas, the following options are available:

- Reduction of suppressant diversion probability as discussed in Section 6.3.1 above.

- Reduction of fire probability through seismic anchoring of cabinets to prevent tipping or sliding (many of these cabinets have been found to be anchored during plant walkdowns).

- Distancing the safety-related cables from the fire source or separating safety-related equipment cables by distance or barriers.

- Routing of some cables other than out of the top of the switchgear to reduce the likelihood that a single cubicle fire would damage a large number of safety-related cables.

6.3.7 Electromechanical Components in Cable Spreading Rooms

In three of four plants studied, the cable spreading rooms contained electrical cabinets. Accordingly, these rooms were subject to incremental risk due to Root Cause 12. In one case, the cable spreading room had no such equipment installed, thus removing the potential for a seismically induced fire.

6.3.8 Diesel Generator/FPS Interaction

In two of the plants studied, the FPS control system interacted with the diesel generator systems:

- In one case, the FPS system, when actuated, sent a lockout signal to the diesel generator located in its associated zone.

- In another plant, the FPS system, when actuated, caused the engine to shutdown due to starvation (high CO_2/low oxygen in the engine intake).

In the first case the engine would shutdown immediately. In the second case the engine would quickly starve because of the lack of oxygen for combustion. In both of these cases, if the FPS control system was susceptible to seismic actuation (relay chatter), then in a seismic event, station blackout could occur.

7.0 SUMMARY

Analysis of USNRC Generic Issue 57 involved development of a detailed understanding of the potential safety significance of U.S. commercial nuclear power plant fire protection system (FPS) advertent and inadvertent actuations. In this report an extensive review of operational experiences involving such FPS actuations is presented. A methodology for the quantification of effects of fire protection system actuation on safety-related equipment has been developed. This methodology has been applied to specific plants, one boiling water reactor (BWR) and two pressurized water reactors (PWRs). In addition, analysis of a third PWR was independently conducted by the Idaho National Engineering Laboratory. For this third PWR, Sandia National Laboratories conducted an independent evaluation of the risk associated with the seismic root causes. In applying the methodology, extensive plant walkdowns were conducted in addition to detailed reviews of plant documentation. Building on the insights gained as a result of the analysis of these four plants, a risk assessment was made for generic light water plant cable spreading rooms, diesel generator rooms, and emergency electrical switchgear rooms. For these rooms, both core damage frequency and incremental risk are calculated. An uncertainty analysis is performed on both core damage frequency and risk. A cost/benefit assessment was then performed for candidate modifications for several plant as-found conditions that were demonstrated to be contributors to risk. Technical insights from all of these analyses are presented for the use of those involved with the design of the advanced light water reactor (ALWR).

7.1 Review of Fire Protection System Actuation Events and Performance

A review of fire protection system actuation events at U.S. commercial nuclear power plants during the time period of January 1, 1980 to December 31, 1989 were presented. Included in this section was a discussion of how the data review was conducted and a summary of both the inadvertent and advertent FPS actuations identified in the study. Data from two additional sources were reviewed. These additional data included information concerning FPS actuations at foreign nuclear power plants, and at U.S. naval shore facilities. A review was conducted of FPS performance data collected after the Loma Prieta earthquake of 1989.

7.2 Methodology for Evaluation of Potential Accident Scenarios Caused by FPS Actuations

A methodology was developed for the evaluation of potential accident scenarios caused by FPS actuations. Thirteen root causes (both seismic and non-seismic) leading to increases in core damage frequency and risk were identified. The methodology uses a plant internal event probabilistic risk assessment (PRA) as a basis, with the addition of a vital area analysis for safety-related components and cables. While helpful, the existance of a fire PRA is not required for the application of the methodology. The methodology can be readily applied to a specific plant, as has been done to three PWRs and one BWR, the results of which are separately reported in References 7.1 through 7.4. In this report, a generic plant model was developed and analyzed using the methodology.

7.3 Generic Plant Analysis

A generic plant analysis was conducted, based on insights gained from the individual plant analytic work as well as the survey of fire protection strategies at U.S. commercial nuclear power plants. For the generic plant, core damage frequency was calculated for a generic cable spreading room, diesel generator rooms, and emergency electrical switchgear rooms. For each space, fire protection systems in use in such spaces in U.S. commercial nuclear power plants were assessed. The FPSs analyzed were wetpipe water, preaction water, deluge water, Halon, and CO_2. After core damage frequency was calculated, generic containment systems were modeled for a PWR and a BWR. Using these models, generic risk was assessed. For core damage frequency, both an uncertainty analysis and sensitivity studies were conducted. For generic plant risk, an uncertainty analysis was performed.

7.4 Cost/Benefit Assessment

During the course of the analysis of individual plants, several design issues were identified as potential contributors to risk. These were identified for individual plants during documentation reviews, plant walkdowns, and application of the risk assessment methodology. Additional issues were identified while performing the analysis of the generic plant. For eleven of these issues, costing of risk-reduction modifications was done, and then cost-benefit ratios (dollars/person-REM averted) for the potential modifications were calculated.

7.5 Technical Insights for the Advanced Light Water Reactor (ALWR)

Technical insights are discussed for use in the ALWR program. These are insights only, not specific recommendations or design requirements. Presented for the use of the ALWR designer are summaries of findings of existing design features that result in contribution of risk due to the effects of fire protection system actuation on safety-related equipment. Additionally presented for consideration was the concept of applying the FPS risk assessment methodology to the ALWR in the design phase, in an effort to optimize the plant design against risk from the effects of FPS actuation on safety-related equipment.

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(2-89)
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BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, If any.)

NUREG/CR-5580
SAND90-1507
Vol. 1

2. TITLE AND SUBTITLE

Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment

Main Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1992

4. FIN OR GRANT NUMBER

L1334

5. AUTHOR(S)

J. Lambright, J. Lynch[1], M. Bohn, S. Ross[1], D. Brosseau[2]

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION — NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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9. SPONSORING ORGANIZATION — NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

Nuclear power plants have experienced actuation of fire protection systems (FPSs) under conditions for which these systems were not intended to actuate, and also have experienced advertent actuations with the presence of a fire. These actuations have often damaged safety-related equipment. A review of past occurrences of both types of such events and their impact on plant safety systems, an analysis of the risk impacts of such events on nuclear power plant safety, and a cost-benefit analysis of potential corrective measures has been performed.

Thirteen different scenarios leading to actuation of fire protection systems due to a variety of causes were identified. A quantification of these thirteen root causes, where applicable, was performed on generically applicable scenarios.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Probabilistic Risk Assessment, Risk, nuclear power plant, fire, fire protection system, fire protection system actuation, core damage frequency, pressurized water reactor, PWR, boiling water reactor, BWR, cost/benefit, licensee event report, LER

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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