NUREG/CR-4826
UCID-20948
Vol. 1

Seismic Margin Review of the Maine Yankee Atomic Power Station

Summary Report

Prepared by P. G. Prassinos, R. C. Murray, G. E. Cummings

Lawrence Livermore National Laboratory

Prepared for
U.S. Nuclear Regulatory
Commission

NUREG/CR-4826
UCID-20948
Vol. 1

Seismic Margin Review of the Maine Yankee Atomic Power Station

Summary Report

Manuscript Completed: March 1987
Date Published: March 1987

Prepared by
P. G. Prassinos, R. C. Murray, G. E. Cummings

Lawrence Livermore National Laboratory
7000 East Avenue
Livermore, CA 94550

Prepared for
Division of Engineering Safety
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN A0461

ABSTRACT

This Summary Report is the first of three volumes for the Seismic Margin Review of the Maine Yankee Atomic Power Station. Volume 2 is the Systems Analysis of the first trial seismic margin review. Volume 3 documents the results of the fragility screening for the review. The three volumes demonstrate how the seismic margin review guidance (NUREG/CR-4482) of the Nuclear Regulatory Commission (NRC) Seismic Design Margins Program can be applied.

The overall objectives of the trial review are to assess the seismic margins of a particular pressurized water reactor, and to test the adequacy of this review approach, quantification techniques, and guidelines for performing the review. Results from the trial review will be used to revise the seismic margin methodology and guidelines so that the NRC and industry can readily apply them to assess the inherent quantitative seismic capacity of nuclear power plants.

TABLE OF CONTENTS

VOLUME 1. SUMMARY REPORT

Page

ABSTRACT....iii
LIST OF TABLES.... xi
LIST OF FIGURES....xii
PREFACE....xiii
EXECUTIVE SUMMARY.... xv

1. INTRODUCTION....1-1

1.1 Organization of the Report....1-4
1.2 Scope of the Seismic Margin Review of Maine Yankee....1-4

2. SEISMIC MARGINS APPROACH....2-1

2.1 Systems Screening....2-2
2.2 Fragility Screening....2-4
2.3 Approach for Performing Seismic Margin Reviews....2-6
2.4 Trial Guidelines for Seismic Margin Reviews....2-7

3. RESULTS OF SEISMIC MARGIN REVIEW....3-1

3.1 Overall Results....3-2
3.2 Systems Results....3-13
3.3 Fragility Results....3-17

4. INSIGHTS AND LESSONS LEARNED....4-1

4.1 Approach and Methodology....4-1
4.2 Guidelines....4-7
4.3 Effort to Perform the Review....4-15
4.4 Findings and Their Resolution....4-16

5. CONCLUSIONS....5-1

5.1 Conclusions Regarding the Methodology, Guidelines, and Insights....5-3
5.2 Conclusion Regarding the Maine Yankee Licensing Issue....5-4

6. RECOMMENDATIONS TO IMPROVE METHODOLOGY....6-1

REFERENCES R-1

APPENDICES

A Expert Panel Correspondence....A-1
B Peer Review Group Correspondence....B-1
C Utility Comments....C-1

TABLE OF CONTENTS

VOLUME 2. SYSTEMS ANALYSIS

Page

ABSTRACT........iii
LIST OF TABLES........vii
LIST OF FIGURES........ix

1. INTRODUCTION........1-1

2. METHODOLOGY........2-1

2.1 General Approach........2-1
2.2 Step 2 - Initial Systems Review........2-1
2.2.1 Gather System Information........2-1
2.2.2 Classify Front-Line Systems........2-1
2.2.3 Identify Group A Front-Line Components........2-2
2.2.4 Classify Support Systems........2-2
2.2.5 Identify Group A Support System Components........2-2
2.2.6 Identify Plant-Unique Features........2-3
2.2.7 Prepare for First Plant Walkdown........2-3
2.3 Step 4 - First Plant Walkdown........2-3
2.3.1 Peer Review Group Meeting........2-3
2.3.2 Plant Walkdown........2-3
2.3.3 Plant Staff Discussions........2-4
2.4 Step 5 - Systems Modeling........2-4
2.4.1 Review Event Trees........2-4
2.4.2 Develop Fault Trees........2-5
2.4.3 Develop Data Base........2-7
2.4.4 Determine System Cut Sets........2-7
2.5 Step 6 - Second Plant Walkdown........2-8
2.6 Step 7 - Determine Minimal Cut Sets........2-8
2.6.1 Finalize Event Trees........2-8
2.6.2 Finalize Fault Trees........2-8
2.6.3 Link Fault Trees........2-8
2.6.4 Determine Preliminary Boolean Equation........2-9
2.6.5 Determine Final Boolean Equation........2-9

3. SYSTEMS ANALYSIS........3-1

3.1 System Identification........3-1
3.1.1 Group A Systems........3-1
3.1.2 Systems Removed from Group A........3-1
3.2 Systems Analysis of Front-Line Systems........3-3
3.2.1 Auxiliary Feedwater System........3-3
3.2.2 High Pressure Safety Injection System........3-5
3.2.3 Power-Operated Relief Valves (Feed and Bleed)........3-8
3.3 Systems Analysis of Support Systems........3-9
3.3.1 Component Cooling Water Systems........3-9

TABLE OF CONTENTS (Cont.)

VOLUME 2. SYSTEMS ANALYSIS

Page

3.3.2 Service Water System....3-11
3.3.3 Electric Power System....3-13
3.3.4 Actuation Systems....3-16
3.4 Probability Calculations....3-16

4. ACCIDENT SEQUENCE ASSESSMENT AND RESULTS....4-1

4.1 Event Trees....4-1
4.1.1 No LOCA Case....4-1
4.1.2 Small LOCA Case....4-2
4.2 Core Damage Sequences....4-3
4.2.1 T_1LP_1....4-3
4.2.2 T_1LD....4-3
4.2.3 S_2D....4-4
4.2.4 S_2LP_2....4-4
4.2.5 S_2LD....4-4
4.3 Boolean Equations for No LOCA and LOCA Cases....4-4
4.3.1 No LOCA Case....4-5
4.3.2 Small LOCA Case....4-6
4.4 Plant-Level Boolean Equation....4-6

5. ENGINEERING AND METHODOLOGY INSIGHTS....5-1

5.1 Engineering and Operational Insights....5-1
5.2 Insights on the Methodology and Execution....5-2
5.2.1 System Calssification and Screening Guidelines....5-2
5.2.2 Preparation for Walkdowns and Documentation....5-4
5.2.3 Systems Analysis and Pruning Process....5-4
5.2.4 Minimal Cut Set Evaluation Process....5-5
5.2.5 Schedule and Resources....5-6

REFERENCES....R-1

APPENDIX

A Identifiers and Symbols....A-1
B Auxiliary Feedwater System....B-1
C High Pressure Safety Injection System....C-1
D Primary Pressure Relief System....D-1
E Primary Component Cooling Water System....E-1
F Secondary Component Cooling Water System....F-1
G Service Water System....G-1
H Electric Power System....H-1
I Actuation Systems....I-1

TABLE OF CONTENTS

VOLUME 3. FRAGILITY ANALYSIS

Page

ABSTRACT....iii
LIST OF TABLES....vii
LIST OF FIGURES....ix

1. INTRODUCTION 1-1

1.1 Background....1-1
1.2 Objective of the Study....1-2
1.3 Scope of the Study....1-2
1.4 Organization of the Report....1-3

2. REVIEW EARTHQUAKE LEVEL....2-1

2.1 Review of Available Seismic Hazard Studies....2-1
2.2 Implication of the Selected Review Earthquake Level....2-1
2.2.1 Screening of Components....2-1
2.2.2 Estimation of Seismic Capacities....2-5
2.3 Different Methods for Specifying Review Earthquake Level....2-5

3. DESCRIPTION OF PLANT STRUCTURES, SYSTEMS AND COMPONENTS....3-1

3.1 Maine Yankee Plant/Structures and Systems....3-1
3.1.1 Structures....3-1
3.1.2 Systems....3-1
3.2 Seismic Design Criteria....3-2
3.3 Availability of Plant Design Data....3-2

4. REVIEW OF PLANT INFORMATION AND WALKDOWN....4-1

4.1 Initial Screening of Components....4-1
4.2 Review of Design-Analysis and Seismic Reevaluation Reports....4-4
4.3 Plant Walkdown....4-6
4.3.1 Identification of Target Areas for First Walkdown....4-6
4.3.2 Walkdown Procedures....4-22
4.3.2.1 Walkdown Team....4-22
4.3.2.2 Procedures for Structures and Equipment....4-23
4.3.3 Walkdown Documentation....4-29
4.3.4 Walkdown Results....4-30
4.3.4.1 Walkdown Findings....4-30
4.3.4.2 Plant Unique Features....4-56
4.4 Maine Yankee Component Modifications....4-56

5. EVALUATION OF SEISMIC CAPACITIES OF COMPONENTS AND PLANT....5-1

5.1 Review of Structural Models....5-1
5.2 Simplified Analysis and Use of Screening Tools....5-2
5.3 Second Walkdown....5-4
5.4 HCLPF Capacity of Components....5-4

TABLE OF CONTENTS (Cont.)

VOLUME 3. FRAGILITY ANALYSIS

Page

5.4.1 Fragility Analysis Method....5-4
5.4.1.1 Methodology....5-4
5.4.1.2 Steel Structures....5-23
5.4.1.3 Block Walls....5-30
5.4.1.4 Flat Bottom Storage Tanks....5-32
5.4.1.5 Inverter....5-40
5.4.1.6 Diesel Fuel Oil Day Tank Fragility Evaluation....5-46
5.4.1.7 Containment Spray Fans....5-52
5.4.2 CDFM Method....5-53
5.4.2.1 Circulating Water Pumphouse....5-54
5.4.2.2 Block Wall SB 35-3....5-54
5.4.2.3 Refueling Water Storage Tank....5-55
5.5 HCLPF Capacity of Plant....5-56
5.5.1 Accident Sequences....5-56
5.5.2 Probabilistic Method....5-59
5.5.3 Deterministic Method....5-67
5.5.4 Sensitivity Studies....5-68

6. COMMENTS ON SEISMIC MARGIN REVIEW METHODOLOGY....6-1

6.1 Selection of Review Earthquake Level....6-1
6.2 Use of Screening Guidelines....6-2
6.2.1 Extent of Review Needed....6-2
6.2.2 Additional Screening Guidelines....6-2
6.2.3 Difficulties of Reviewing Certain Components....6-3
6.2.3.1 Reactor Internals and CEDM....6-3
6.2.3.2 Instrumentation within Containment....6-3
6.3 Availability of Qualification Data....6-4
6.4 Walkdown Procedures....6-5
6.5 Guidance on CDFM Capacity Calculation Procedures....6-6
6.6 Staffing Requirements and Schedule....6-7
6.6.1 Staffing Requirements....6-7
6.6.2 Schedule....6-8
6.7 Applicability of Methodology to Other Plants....6-8

7. RESULTS AND CONCLUSIONS....7-1

7.1 Screened Out Systems and Components....7-1
7.2 HCLPF Capacities of Screened-in Components....7-2
7.3 HCLPF Capacity of Plant....7-32
7.4 Identification of Low Capacity Components....7-32
7.5 Conclusions....7-32

REFERENCES....R-1

TABLE OF CONTENTS (Cont.)

VOLUME 3. FRAGILITY ANALYSIS

Page

APPENDIX A

Maine Yankee Atomic Power Station Arrangement Drawings.............A-1

LIST OF TABLES

		Page
Table 3-1	Component Seismic Fragility Parameters	3-3
Table 3-2	Probabilities for Nonseismic Failures	3-4
Table 3-3	Summary of Plant Level HCLPF Capacities	3-9
Table 3-4	Front-line System vs Support System Dependency Matrix	3-14
Table 3-5	Support System vs Support System Dependency Matrix	3-15
Table 3-6	Minimal Cut Sets for Important Systems	3-18
Table 3-7	Maine Yankee Structures and Block Walls	3-21
Table 3-8	Maine Yankee Equipment List	3-23
Table 4-1	Cost Breakdown for Seismic Margins Trail Plant Review	4-11

LIST OF FIGURES

Page

Figure 2-1 Graphic rpresentation of the screening operations (Figure 2-6 from NUREG/CR-4482)....2-6

Figure 2-2 Example systematic event trees....2-8

Figure 2-3 Example of a simplified fault tree....2-11

Figure 2-4 Fault tree for failure of RCS integrity....2-13

Figure 2-5 Example fragility curves for a structure....2-16

Figure 3-1 Fragility curves for small LOCA core damage....3-7

Figure 3-2 Seismic fragility curves for NO-LOCA core damage....3-8

PREFACE

The seismic margin review of Maine Yankee Atomic Power Plant was performed for the U.S. Nuclear Regulatory Commission as part of a research program on quantification of seismic margins at nuclear power plants. The approach, methodology (NUREG/CR-4334), and guidelines (NUREG/CR-4482), were developed by the Expert Panel on the Quantification of Seismic Margins and its technical support personnel. Maine Yankee and Yankee Atomic provided necessary data and information to this margin review. The results were reviewed by a Peer Review Group and the NRC Seismic Design Margins Working Group. This report is a collective effort and presents the results of the seismic margins review of the Maine Yankee plant.

AUTHORS OF THE REPORT AND ANALYSIS TEAM

Volume 1. Summary Report

Lawrence Livermore National Laboratory

P. G. Prassinos
R. C. Murray, Project Manager
G. E. Cummings

Volume 2. Systems Analysis

Energy Incorporated

D. L. Moore, Systems Analysis Team Leader
D. M. Jones
M. D. Quilici
J. Young

Volume 3. Fragility Analysis

EQE Incorporated

M. K. Ravindra, Fragility Analysis Team Leader
M. J. Griffin
G. S. Hardy
P. S. Hashimoto

U.S. NUCLEAR REGULATORY COMMISSION

D. J. Guzy, Project Leader
A. J. Murphy, Section Leader, Engineering Branch
J. E. Richardson, Chief, Engineering Branch, Division of Engineering Safety

NRC Seismic Design Margins Working Group

N. Anderson, Cochairman
J. Richardson, Cochairman
G. Bagchi
L. Beratan
J. Chen
N. Chokshi
C. Grimes
D. Guzy
P. K. Niyogi
L. Reiter

UTILITY

Maine Yankee Atomic Power Company

S. V. Evans
G. D. Whittier

Yankee Atomic Electric Company

P. L. Anderson
W. E. Henries
J. T. McCumber
W. J. Metevia

R. P. Kennedy, Consultant

REVIEWERS

Peer Review Group

R. J. Budnitz (Chairman), Future Resources Associates
M. P. Bohn, Sandia National Laboratories
J. W. Reed, Jack R. Benjamin & Associates, Inc.
J. Thomas, Duke Power Company
L. A. Wyllie, H. J. Degenkolb & Associates

Expert Panel on the Quantification of Seismic Margins

R. J. Budnitz (Chairman), Future Resources Associates
P. J. Amico, Applied Risk Technology Corporation
C. A. Cornell, CAC, Inc.
J. W. Reed, Jack R. Benjamin & Associates, Inc.
M. Shinozuka, Columbia University

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission is sponsoring a research program to develop and demonstrate a method for assessing safety margins at nuclear power plants with respect to seismic events. This program is called the Seismic Margins Program and was started in 1984 at the Lawrence Livermore National Laboratory, [Cummings et al., 1984]. The need for such research results from the changing perception of the seismic hazard in certain localities which could require reassessment of the adequacy of seismic design at some power plant sites. It is generally accepted that nuclear power plants are capable of withstanding earthquake motion substantially greater than their design basis, but methods are needed to systematically demonstrate this. An "Expert Panel on the Quantification of Seismic Margins" was formed to develop such a margins assessment method [Budnitz et al., 1985] and [Prassinos et al., 1986], and this report discusses the application of that method to the Maine Yankee Atomic Power Station, a Combustion Engineering, three-loop pressurized water reactor located near Wiscasset, Maine.

The margin review process involves the screening of components based on their importance to safety and their seismic capacity. The products of the review are High Confidence of Low Probability of Failure (HCLPF) capacities for components, accident sequences and the plant. Mathematically, the HCLPF can be thought of as an estimate of the 5% failure probability point with 95% confidence. These HCLPF capacities, measured in terms of peak ground acceleration (pga), are compared to the peak ground acceleration predicted for the earthquake against which the plant is to be assessed, called the review level earthquake. This review level earthquake is chosen at some level above the design basis (safe shutdown earthquake, SSE) and a favorable comparison with the HCLPF capacity of the plant indicates that there is high confidence of a low probability of failure (core damage). The extent to which this plant HCLPF capacity is above the SSE level is a measure of the seismic margin of the plant.

Systems analysis is used to determine those plant systems and components (including structures) that are important contributors to plant seismic safety and thus allow focusing of effort on components requiring a margin review. By studying previous seismic probabilistic risk assessments (PRA) of pressurized water reactors (PWR), it was found that only systems and components needed to assure reactor subcriticality and early emergency core-coolant injection needed to be considered. Therefore, only systems and components related to these two functions (with a few exceptions) were considered at Maine Yankee.

Capacities of generic sets of components were estimated and given in NUREG/CR-4334 based on estimated capacities in previous PRA's, experience data gained from studying earthquake effects on industrial facilities and engineering analysis. These generic capacities were used to screen out from further consideration those components identified as important by systems analysis if the generic capacity was found to be higher than the review earthquake level for Maine Yankee. This review level earthquake was established by the NRC as

0.30g with a 50th percentile Newmark Hall spectra, as defined in NUREG/CR-0098. To make sure that the generic capacity of all important components could be properly predicted, by using the tables found in NUREG/CR-4334, these components underwent review and inspection before being screened out. In addition, any potential system interaction or plant-unique features discovered were added to the list of review items.

The components remaining after the systems and fragility screenings, plus the systems interaction and plant-unique global features, were then subjected to a margins quantification. Prior to this quantification, each remaining component was thoroughly inspected and studied, and systems models were developed to describe the possible seismic-initiated accident behavior of the plant. The quantification was accomplished by calculating the HCLPF capacities for each of these components using structural/mechanical analyses and then analyzing the minimal cut sets derived from the systems analysis using the rules of Boolean algebra and Discrete Probability Distribution methods to arrive at accident sequence, and plant level HCLPF capacities. Component HCLPF capacities were calculated for the important components remaining after screening, using the Fragility Analysis (FA) method. This method requires estimating the median failure capacity for the component, and its random and modeling uncertainties. Assuming a lognormal failure probability distribution, the HCLPF (5% failure with 95% confidence) capacity can be calculated. Checks on several key components were made using the Conservative Deterministic Failure Method (CDFM) which uses a deterministic, more design-oriented method of calculating component HCLPF's. Further work is underway to cross-check the two methods. Random, test and maintenance, and human error failure modes were also included in the analysis.

MARGINS REVIEW OF MAINE YANKEE

The seismic margins review of Maine Yankee was an eight-step process which involved Maine Yankee, Yankee Atomic Electric, NRC, LLNL as project manager, and fragility and system analysis teams (EQE Inc. and Energy Incorporated, respectively). The first step was to establish the review level earthquake (0.30g). Steps 2 and 3 involved information gathering and preliminary analysis by the two teams leading to step 4, a plant inspection (called a walkdown). In step 5, following the plant inspection, event and fault trees were constructed including those components not screened out. In step 6, a second visit was made to the plant to recheck the components remaining which might require further analysis. In step 7, minimal cut sets leading to core damage were determined. In step 8, the component HCLPF capacities were finalized and HCLPF capacities for accident sequences and the plant calculated. This entire effort took about 3.0 man-years of effort.

For Maine Yankee, two important accident sequences were identified. Both are initiated by a seismically induced loss of offsite power (LOSP) assumed to always occur at the review earthquake level. In one sequence there is a small loss-of-coolant accident (LOCA of 3/8 in. to 2 in. diameter equivalent area) assumed to occur because of seismically induced pipe breakage. HCLPF capacities for components which might cause other types of small LOCAs (pump seal or power-operated relief valve LOCAs) were sufficiently high so they could be screened out. The other accident sequence assumed no small LOCA.

The small LOCA accident sequence involved seismic failures only and resulted in a plant HCLPF of 0.21g. Many plausible arguments, including the component screening table of NUREG/CR-4334, indicate that a small LOCA could be screened out for the review level earthquake considered. Since the analysts involved in this review could not get inside the Maine Yankee containment to inspect the small primary system piping, they chose not to screen small LOCA out. If they had, this sequence would not be considered and the plant HCLPF would be above 0.30g.

The small LOCA accident sequence was composed of three singleton cut sets with the dominant contributor being failure of the Refueling Water Storage Tank (RWST) which had a HCLPF of 0.21g. Other singletons in the sequence had HCLPF's greater than 0.30g.

The second accident sequence, LOSP with no small LOCA, involved no singletons but a number of doubletons, some combining seismic and nonseismic (random, test and maintenance, human error) failures. The most important doubleton is the Demineralized Water Storage Tank (DWST, HCLPF = 0.17g) and the Circulating Water Pump House (HCLPF > 0.30g).

Nonseismic failures were found not to be important contributors. They made no contribution to the overall plant level HCLPF capacity. The most important nonseismic failure found was a common cause failure of the Auxiliary Feedwater System, caused by steam binding (median unavailability per demand of 1.2×10^{-4}).

It should be noted that during the review process, important components were found for which a low HCLPF would result or insufficient data was available to determine the HCLPF. These were the lead-antimony station batteries, the station service transformers, a block wall near HVAC equipment, parts of the PCCW and SCCW air conditioning heat exchangers, and anchorage of the diesel generator day tanks. To reduce the uncertainty in their capacities, these components are being replaced or upgraded and the results of this review are based on the upgraded configurations.

INSIGHTS AND LESSONS LEARNED

In addition to the results already described, the Maine Yankee seismic margins review provided some lessons applicable to future reviews. The event/fault tree methods used provided a complete description of dominant contributors and considered all important systems. The fragility screening table (NUREG/CR-4334) needs to be strengthened and more guidance given on how to select a review level earthquake. Also, more guidance is needed on how to use and compare the two methods used to determine component capacity (FA and CDFM) and how to combine seismic and nonseismic failures.

The systems analysis effort should start early so that component screening and plant inspections can be done efficiently. Information in the plant Final Safety Analysis Report can be effectively used for this effort. Plant inspections (walkdowns) need to be carefully planned, taking into account auxiliary systems such as the HVAC and actuation/control system, as well as important systems and components identified in the systems analysis. These walkdowns are essential to successful margin reviews. Specifically, they permit accurate data collection and allow identification of potential low capacity components.

More guidance is needed on the consideration of the seismically initiated small LOCA. This initiating event turned out to be particularly important at Maine Yankee. It may be impossible to review and inspect all the small primary pressure boundary piping, and other screening methods need to be developed, e.g., inspection of similar piping outside of containment.

Finally, there is a question about maintenance of hot shutdown for a specified period of time after an earthquake, e.g., 72 hours. The current methodology addresses the attainment of hot shutdown but not necessarily its maintenance. Review of previous seismic PRAs indicates that once hot shutdown is attained, the probability of maintaining it is large. Therefore, this issue may be of less importance.

VOLUME 1. SUMMARY REPORT

CHAPTER 1

1. INTRODUCTION

In 1984, the U.S. Nuclear Regulatory Commission (NRC) initiated the Seismic Design Margins Program (SMP) to address regulatory needs and a changing perception of the seismic hazard. The NRC formed the "Expert Panel on the Quantification of Seismic Margins" and charged it to work closely with an in-house NRC staff, "Working Group on Seismic Design Margins," to provide technical guidance on the assessment of seismic margins. The overall goal of the SMP is the development of a methodology and guidelines that can be readily used by the NRC and industry for assessing the inherent quantitative seismic capacity of nuclear power plants [Cummings et al., 1984].

The development of a soundly based, efficient and effective method for the assessment of how much margin actually exists in important components, systems, and the plant will serve to minimize the impact of changing regulatory requirements and licensing actions as the estimates of seismic hazards change. In addition, a seismic margins assessment can provide a basis for confidence in the capacity of nuclear power plants and this methodology can be applied when questions arise about their seismic capacity.

The most important regulatory need and the focus of the seismic margins effort is stated as follows:

> "There is a need to understand how much seismic margin exists at nuclear power plants. This seismic margin is to be expressed in terms of how much larger must an earthquake be above the safe shutdown earthquake (SSE) before it compromises the safety of the plant."

The Expert Panel and its technical support personnel studied the available information on the quantification of seismic capacity of nuclear power plants and other industrial facilities. The results of several seismic probabilistic risk assessments of nuclear power plants were reviewed along with the behavior of industrial facilities during earthquakes. These studies were used to develop a margin review approach that involves both the screening of components based on their importance in preventing seismic core melt and their inherent seismic capacity.

The seismic margins review approach has been documented in the report, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants" [Budnitz et al., 1985]. This document formed the basis for the development of guidelines for performing seismic margin reviews. These guidelines are given in "Recommendation to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants, Draft Report for Comment" [Prassinos et al., 1986].

The performance of a trial plant review is needed to verify and test the review methodology and guidelines. For the trial review, the NRC negotiated with and selected the Maine Yankee plant. Once the trial plant was selected, the two analysis teams (Systems Team and Fragility Team) were chosen. These teams were selected based on their technical approach and team composition. In addition, a Peer Review Group was selected and charters were developed for both this group and the Expert Panel. The seismic margin review was then organized allowing for participation by the plant owner and its representative (Yankee Atomic Electric Company), the NRC Working Group on Seismic Design Margins, and the appropriate NRC program managers.

The purpose of this report is to summarize the results of this first trial seismic margin review. The detailed results from this review can be found in Volumes 2 and 3 of this report. Volume 2, Systems Analysis, documents the results of the systems screening and analyses portion of the review, while Volume 3, Fragility Analysis, gives the component fragility screening and analyses results.

The overall objectives of this trial review are to assess the seismic margins of a pressurized water reactor, the Maine Yankee Atomic Power Station, and to test the adequacy of the seismic margin review approach and quantification techniques, and the guidelines for performing these reviews. There are four related objectives of this study:

- To demonstrate the use of the Expert Panel's approach (NUREG/CR-4334) and guidelines (NUREG/CR-4482) for seismic margin reviews.

- To provide a basis for revising and upgrading the approach and guidelines.

- To provide a benchmark for possible future seismic margin reviews, including an understanding of the level of effort in performing a seismic margin review.

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

The results from this trial review are intended to be used to revise the seismic margin methodology and guidelines so they are more prescriptive and ready for general use by the NRC and industry.

The role of the Expert Panel during the Maine Yankee study was to review the procedures being used at an early stage of the process and to assure that the methods and techniques being employed are consistent with the Panel's guidance and are relevant for performing a seismic margins review. When the trial plant review is completed, the Panel will study the results, examine how the review was implemented, and evaluate the overall effort. The Expert Panel will be expected to appraise the overall usefulness of the seismic margin review approach and identify any limitations whether previously recognized or not. Appendix A defines the role of the Expert Panel for this review and contains minutes of the panel's telephone conference call.

A Peer Review Group was selected and chartered to review the technical adequacy of this study including participation in the plant walkdowns. The objective of the Peer Review Group is to assure that the seismic margin review is executed in a fully competent and professional manner, uses appropriate methods, and follows the guidance established in NUREG/CR-4334 and NUREG/CR-4482. At the conclusion of the Maine Yankee review, the Peer Review Group provided its best judgment with regard to both the review procedures and the technical competence of the reviews, based on its collective expert opinion [Anderson, 1986]. Peer Review Group correspondence including its charter, meeting minutes and summary report on this trial review are given in Appendix B.

Appendix C contains comments received from the utility.

1.1 Organization of the Report

The remainder of this section gives a discussion of the scope of the effort. Chapter 2 presents a brief overview of the seismic margin review methodology and the guidelines as applied to this review. Chapter 3 provides the overall results of the Maine Yankee review, including a summary of the significant fragility and systems results. Insights and lessons learned are given in Chapter 4 followed by the conclusions in Chapter 5. Recommendations from this study are given in Chapter 6.

1.2 Scope of the Seismic Margin Review of Maine Yankee

The Maine Yankee Atomic Power Station is a Combustion Engineering (CE) three-loop pressurized water reactor (PWR) located approximately 3.9 miles south of Wiscasset, Maine. The architect engineer for the plant was Stone and Webster Engineering Corporation. The Maine Yankee plant started commercial operation in 1972. Its present net electrical power output is 825 megawatts electric (2630 megawatts thermal). A brief description of the plant configuration, including structures and systems, is given in Volumes 2 and 3. A detailed description of the Maine Yankee plant is given in the FSAR [Maine Yankee, FSAR].

All seismic "Class 1" structures and components of the plant which are important to nuclear safety, and could affect the health and safety of the public, are designed based on a minimum horizontal ground acceleration of 0.05g and a safe shutdown earthquake (SSE) horizontal acceleration of 0.1g [Maine Yankee, FSAR]. Damping at these acceleration levels is 2 percent and 5 percent, respectively. Vertical acceleration is taken as two-thirds of the horizontal acceleration and is considered to act simultaneously with each horizontal component.

The occurrence of two seismic events in the vicinity of the plant, one in 1979 and the other in 1982, prompted Maine Yankee to upgrade the capability of the plant to withstand a potential seismic event in excess of the original design-basis event. Based on these upgrades and the inherent design capacity of the plant, Maine Yankee concluded that the plant structures, systems, and components had sufficient strength to withstand a seismic event of at least 0.2g with a Regulatory Guide 1.60 [USNRC, 1973] spectrum and still shut down without danger to the public health and safety [Miraglia, 1986].

To assure the NRC that the plant could withstand earthquake motion greater than the design basis, the utility agreed to participate in the trial seismic margin review of the Maine Yankee plant. For this review, it was agreed the seismic margin review earthquake level would be 0.3g with a 50th percentile Newmark Hall Spectra defined in NUREG/CR-0098. Magnitude and duration criteria specified in NUREG/CR-4334 apply [Guzy, 1986], [Crutchfield, 1986]. The margins concept requires the HCLPF capacity to be associated with a defined response spectrum and a specified nonexceedance probability.

The HCLPF capacity used for screening as well as the calculated HCLPF capacities for particular components not initially screened out and the final plant level HCLPF capacity are considered to be valid provided ground motion from any earthquake does not exceed the review earthquake level spectrum for more than 16% of the spectral frequencies within the range of interest. The review earthquake level spectrum is a spectral shape defined by the 50% exceedance spectrum specified in NUREG/CR-0098 and anchored at 0.3g pga for the initial screening. The seismic margin for the components and plant is referenced to this spectrum but anchored to the pga corresponding to the HCLPF capacity.

This definition of spectra used to determine a HCLPF capacity does not in any way refer to the probability of occurrence of an earthquake. It is no more than an arbitrary spectrum used to define the HCLPF capacity that recognizes the dependency of a component capacity on the frequency content of the spectrum and not just the pga.

The results of the seismic margin study are interpreted as follows. The HCLPF capacity of the structures, equipment and plant are conditional on the actual site-specific spectrum not exceeding the target spectrum; exceedance is defined as the event when 16 percent of the spectral ordinates exceed the target spectrum over the frequency range of interest. It is assumed that the spectrum peak-to-peak and earthquake direction variabilities are removed from the hazard analysis leading to the selection of the review earthquake. The review earthquake is specified by the same spectrum in two horizontal directions and 2/3 of the horizontal spectrum in the vertical direction. It is also assumed that the review earthquake level is specified as the higher of the response spesctra from the two orthogonal horizontal directions.

CHAPTER 2

2. SEISMIC MARGINS APPROACH

Insights gained from the results of seven published probabilistic risk assessments (PRAs) were used in the development of a screening approach that combined systems insights and fragility information to simplify the margin review process. This approach is directed at reviewing a specific plant at a selected earthquake acceleration level greater than the SSE.

A general definition of "seismic margin" is stated below:

> "Seismic margin is expressed in terms of the earthquake motion level that compromises plant safety, specifically leading to melting of the reactor core. In this context, margin needs to be defined for the whole plant. The margin concept also can be extended to any particular function or component."

The adopted measure of margin is the earthquake level for which there is a High Confidence, Low Probability of Failure (HCLPF). The HCLPF is a conservative representation of capacity and in simple terms corresponds to the earthquake level at which it is extremely unlikely that core damage will occur. Generally, the median capacity is at least a factor of 2 greater than the HCLPF capacity and thus no proverbial "cliff" or sudden failure is expected to occur immediately upon exceeding the HCLPF capacity. From the mathematical perspective of a failure probability distribution on capacity (fragility), the HCLPF capacity is approximately equal to a 95 percent confidence (probability) of not exceeding about a 5 percent probability of failure. HCLPF capacity for specific types of components are derived from a combination of engineering data, either test data or data from real earthquake experience, and engineering analysis.

2.1 Systems Screening

The review of the available seismic PRAs indicated some key trends and insights useful in developing seismic margin review criteria. Chapter 4 of [Budnitz et al., 1985] discusses the details of the screening approach. These trends and insights could only be obtained for pressurized water reactors (PWRs), for which there were seven available PRAs.

The Expert Panel review indicated that systems screening must be performed at the functional level due to the diversity of plant designs and system configurations. This led to considering the general plant safety functions normally performed in PWRs:

1. Reactor Subcriticality.
2. Normal Cooldown.
3. Early Emergency Core Cooling (injection).
4. Late Emergency Core Cooling (recirculation).
5. Containment Heat Removal.
6. Early Containment Overpressure Protection (injection).
7. Late Containment Overpressure Protection (recirculation).

Examination of PRA results indicated that the dominant plant damage states to core melt for seismic events are generally early core melt with early containment failure. In addition, these plant damage states generally involve core melt induced by failure of the first three functions listed above followed by loss of containment integrity early in the accident progression. This insight led to the division of plant safety function into two groups:

Group A - Functions 1-3.
Group B - Functions 4-7.

The systems screening approach is based on this insight. The dominant plant damage states are caused by failure of the Group A functions and these plant damage states are also characterized by functional failure of the Group B functions. Therefore, we consider core damage to occur whenever there is a failure of the systems that provide the initial shutdown of the nuclear reaction and cooling of the reactor core. These failures are followed by failure of the systems that provide the Group B functions so as to preclude the mitigation of consequences by providing containment protection and cooling.

The normal cooldown function is not considered in our screening criteria because a loss of offsite power is assumed to occur in all seismic PRAs as a result of "low capacity" switchyard components.

The relationship between the Group A and Group B functions indicated that they are highly coupled so that the combined failure of Group A and success of Group B (or the combined success of Group A and failure of Group B) is virtually precluded since their weakest links are coupled. This insight indicates that only those systems and components needed to perform the Group A function must be considered in a seismic margins review.

The Expert Panel also discovered that seismic core melt can occur due to the failure of specific unique features. This implies that it will always be necessary to perform some kind of plant walkdown to look for any unique features requiring a margin review. It must be emphasized that only major, unique features are of concern. Examples of these are the presence of a dam upstream of a plant or a free-standing stack that could damage surrounding buildings if it collapsed.

The insights gained from our systems review of the PRAs and the development of a systems screening criteria to simplify the margin review process has resulted in the following conclusions:

1. It is possible only to come to conclusions regarding the relative importance of plant systems and safety function for PWRs for which six plants were studied (one by two different methods). No function/systemic conclusions can be made about boiling water reactors (BWRs) without examination of additional PRAs.

2. For PWRs, it is possible to categorize plant safety functions as belonging to one of two groups, one of which is important to the assessment of seismic margins and one of which is not.

3. The important group involves only two plant functions that must be considered for estimating seismic margin. These two functions are shutting down the nuclear chain reaction and providing cooling to the reactor core in the time period immediately following the seismic event.

4. It is possible to reasonably estimate the seismic margin of the plant by performing a margins study only involving the analysis of the plant systems and structures which are required in order to perform those two safety functions.

Using the systems screening criteria and combining it with fragility insights, we can establish functional/systemic guidelines for margin reviews. These guidelines show that it is possible to perform a reasonable seismic margin review by concentrating on those functions (and associated systems) which are required in the early part of the seismic event and eliminating from the review those functions which are not required until later. Further, depending on the level of earthquake for which it is desirable to define a margin, certain initiating events would not have to be considered (e.g., large LOCAs). This reduces the level of effort and scope of the analysis.

Event trees need only be constructed up to the point of determining whether or not there is an early core damage. Fault trees would only have to be constructed for those front-line systems (and their support systems) that appear on these abbreviated event trees. By combining a nonseismic failure probability and seismic fragility screening criteria with these systems models, it would only be necessary to include those components which have not been removed during the screening process.

2.2 Fragility Screening

The available fragility information that was reviewed and assessed is based primarily on the detailed analysis of nuclear power plants performed for PRAs. This PRA information was supplemented by recent systematic investigations of historic earthquakes. The information includes past earthquake performance data for eight classes of equipment obtained by the Seismic Qualification Utility Group (SQUG), and reviewed by the Senior Seismic Review and Advisory Panel (SSRAP). Work is ongoing to document the historic earthquake performance and qualification data for additional components such as piping, valve operators, penetrations, diesel generators, battery racks, and electrical equipment. This work is being conducted by SQUG, the Electric Power Research Institute, and the American Society of Civil Engineers Dynamic Analysis Committee [ASCE, 1986]. Additionally, the collective knowledge of the Expert Panel members with actual earthquake experience at nuclear and nonnuclear industrial facilities, performance test data, and other analyses were included in the assessment.

These available sources of fragility information were used to arrive at conclusions about which components should be assessed from a seismic capacity standpoint. In making statements about the need for capacity assessments for each component, three ranges, stated in peak ground acceleration (pga), were used: (1) less than 0.3g, (2) 0.3g to 0.5g, and (3) greater than 0.5g. Each type of nuclear power plant component was assessed to have a generic HCLPF capacity within one of these ranges. This resulted in an extensive table of components indicating at what earthquake level each component will require a margin review or be removed from the review process. This table is given in [Budnitz et al., 1985].

This categorization of components is based on the available information, and the fragility screening resulting from the use of this table should only be performed with consideration of the caveats, limitations, and assumptions presented in Chapter 5 of [Budnitz et al., 1985].

During the process of assessing the HCLPF capacities for the various nuclear power plant components, an extensive fragility information base was developed from the available seismic PRAs. This fragility information base is available on a diskette for use on personal computers, and is documented in [Campbell et al., 1985].

2.3 Approach for Performing Seismic Margin Reviews

The combined insights gained on plant functions and component fragilities were used to develop an outline of an approach for performing seismic margin reviews. The review approach consists of eight steps, the first of which is the selection of an earthquake review level for which it is desirable to demonstrate margin. The eight steps developed in [Prassinos et al., 1986] are outlined later in this section.

It is important to point out the assumptions and limitations of the approach.

1. The systems screening part of the approach presently applies only to PWRs.

2. The review approach focuses on earthquakes that could occur in the eastern part of the U.S., specifically east of the Rocky Mountains.

3. The assessment of component HCLPF capacities is limited to earthquakes of less than a magnitude of about 6.5, which are characterized by 3 to 5 strong motion cycles with a total duration of 10 to 15 seconds.

4. The effects of undiscovered design and construction errors are not covered.

5. Possible vulnerabilities in hydraulic systems associated with sensors and pneumatic systems are not fully covered.

6. Electrical and control systems are incompletely covered because unrecoverable relay chatter and breaker trip is not adequately treated at this time.

7. Evaluation of the effect of wear and aging on equipment function is not fully covered.

8. Possible adverse human responses caused by earthquake-induced stress are not explicitly covered.

The first three limitations listed above are based on the data from PRAs and industrial facilities that were used in the development of this approach and present true limitations on the methodology but not on the Maine Yankee review. Some of the remaining items represent limitation on our knowledge of how to adequately address these issues, while others require considerable effort and are beyond the scope of this analysis.

2.4 Trial Guidelines for Seismic Margin Reviews

The objective of the seismic margin review guidelines is to provide guidance for determining whether a plant can resist with high confidence a specified earthquake level greater than the SSE. To accomplish this objective, analyses are performed on components, systems, and the plant to determine what the HCLPF capacity is so that it can be compared to the specified earthquake level. Plant failure is defined as the onset of core damage.

A flow chart of the margin review process is shown in Figure 2-1 This process involves the screening of components based on their importance to plant safety and their seismic capacity. Inspection of Figure 2-1 indicates that Steps 2, 5, and 7 are primarily concerned with plant safety functions and systems, and are performed by a team of systems analysts. Steps 3, 6, and 8 are mainly concerned with capacity assessment and are performed by a team of fragility analysts. Step 4 is performed by both teams of analysts. The entire process requires close cooperation and interaction between the two teams of analysts and the utility.

The initial step in the review process is the selection of the margin review earthquake level. The margin review earthquake level selected for the Maine Yankee review was discussed in Section 1.2.

In Step 2, plant information gathering, review, and analysis is performed to determine those plant systems and components (structures and equipment) that are important contributors to plant safety and thus allow focusing of the effort on the components requiring a margin review. Also performed during Step 2 is an identification of the relevant seismic initiating events and the development of preliminary event trees that describe the systematic behavior of the plant following these initiating events.

The team of systems analysts review the plant information and determine those front-line systems that perform the two functions important to plant safety, reactor subcriticality and early emergency core cooling. Examples of these systems are the reactor scram, emergency boration, high pressure safety injection, and auxiliary feedwater systems. The support systems to these front-line systems are then determined. Examples of the support systems include electrical power, cooling, actuation, and control.

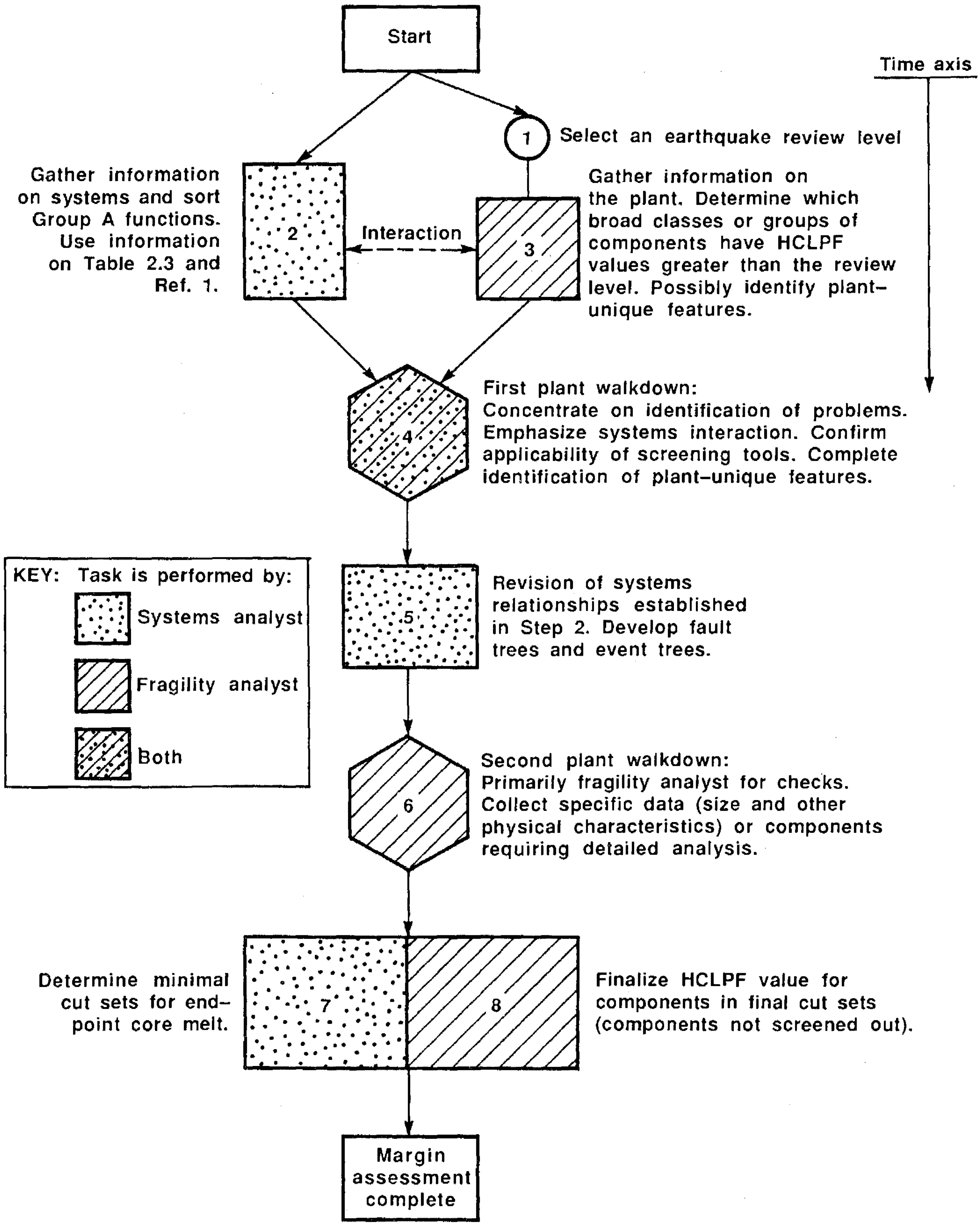


Figure 2-1 Graphic representation of the screening operations.

Once these systems have been determined, the components that make up these front-line and support systems are listed. This list is shared with the fragility analysts and these components become the focus of the first plant walkdown.

Using the identified front-line systems, preliminary systemic event trees are developed for seismic initiating events. An example of an event tree is shown in Figure 2-2. An event tree is a logic diagram that is used to determine the sequence of successes and failures of the systems of concern following a postulated initiating event.

An illustration of an accident sequence is shown in Figure 2-2 by the heavy dark line on the tree. This sequence assumes the initiation event (IE) has occurred followed by the failure of sys1, the failure of sys2, the success of sys3, and the failure of sys4. This accident sequence is represented by the Boolean expression:

$$AS1 = (IE) * (sys1) * (sys2) * \overline{(sys3)} * (sys4), \tag{1}$$

where the system identifier (sys1, sys2, etc.) with the bar indicate system success and the others indicate system failure. This equation reads "the initiating event must occur and sys1 must fail and sys2 must fail and sys3 must succeed and sys4 must fail" for AS1 to occur. The multiplication is the mathematical representation of the logical "and" operator where each event must occur for the occurrence of the outcome.

Based on the systems that fail and succeed in each accident sequence Boolean expression, it is determined whether that sequence of events leads to core damage (CD). If core damage is postulated to occur, we then need to determine the occurrence of each event within the accident sequence to determine the sequence outcome. Since we have assumed the initiating event always occurs given the occurrence of the margin review earthquake, the occurrence of the initiating event is taken as a certainty (probability = 1.0).

Normally, system success is quantified by one minus the probability of system failure. However, where system failure is small, the occurrence of each system success is assumed to be unity (probability = 1.0). This assumption is made for a seismic margin review at a review earthquake level of 0.3g. For higher review levels, system success may need to be considered.

A Boolean expression for the occurrence of each system failure is determined by fault tree analysis. Fault trees are used to determine the combinations of front-line and support system component failures that lead to system failure. For the Maine Yankee analysis, the component failures are represented by their fragility curves and nonseismic failure probabilities. Each accident sequence Boolean expression is then quantified by replacing each system failure with its Boolean expression and solving the resulting logical expression for the sequence HCLPF capacity.

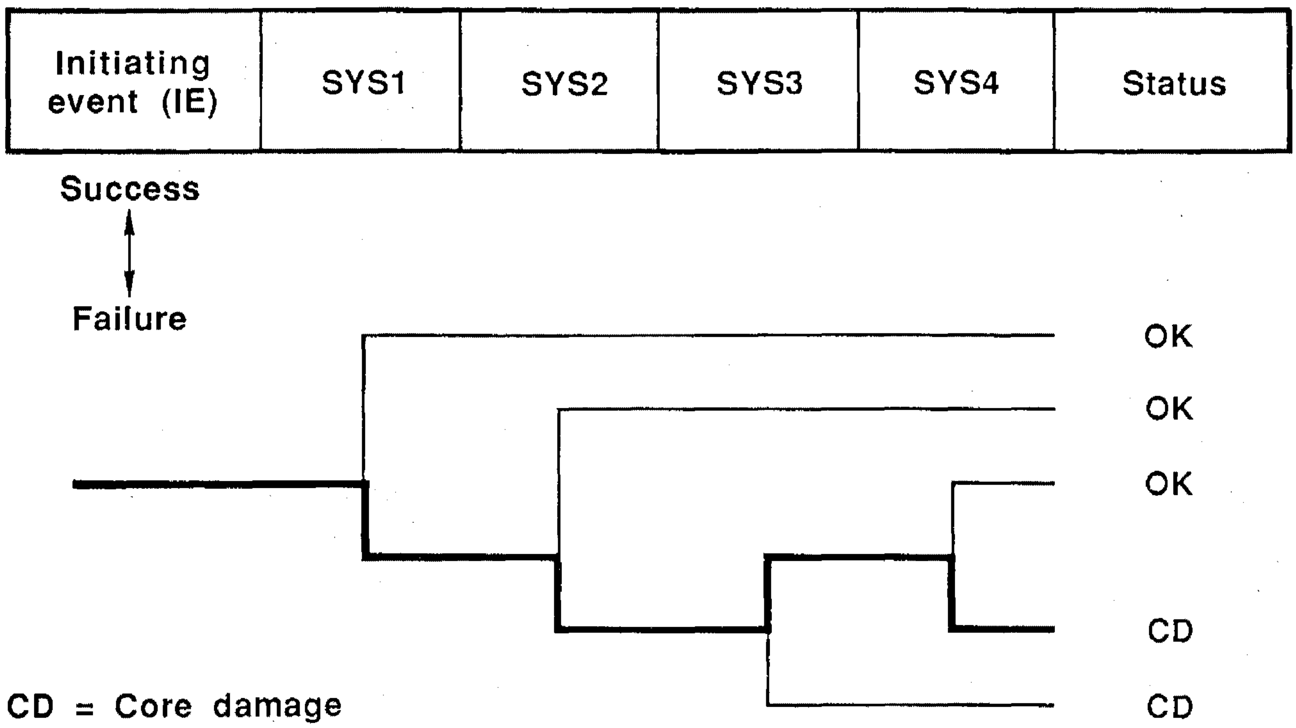


Figure 2-2 Example systemic event trees

A more detailed discussion of systems analysis and event tree development is given in the IREP Procedures Guide [Carlson, 1983] and the PRA Procedures Guide [USNRC, 1983],

For the Maine Yankee review, the initiating event was considered a seismic induced loss-of-offsite power (LOSP). An event tree was constructed that considered the systems that perform the reactor subcriticality and early emergency core cooling functions.

In addition, the event tree considered the integrity of the reactor coolant systems (RCS). This event considered whether a loss-of-coolant accident (LOCA) would occur as a result of the earthquake along with the LOSP. Only a small LOCA was considered to occur. A large LOCA was not considered because large RCS piping and their supports have generic capacities above the review earthquake level and were screened out. The break size of the small LOCA was considered to have an area equivalent to a 3/8-to-2-in.-diameter pipe and requires the operation of the high pressure injection system for mitigation.

The event trees developed in this step were preliminary. They were revised following the first plant walkdown and finalized after the second walkdown. If, however, the first walkdown uncovers reasons for including other systems, initiating events, or components within the analysis, these will be included.

Concurrently, in Step 3, knowledge gathered from the plant and knowledge of the inherent capacity of components is used to sort the components developed in Step 2 into two groups, those with a generic HCLPF capacity larger than the review earthquake level and those that have smaller HCLPF capacity.

For the Maine Yankee review, the culmination of steps 2 and 3 resulted in the identification of structures, block walls, equipment, and areas of the plant that need to be inspected and reviewed. In addition, the first walkdown was planned including organizing the walkdown teams, developing procedures for the review of the various components, developing data sheets for recording the findings, and making arrangements with the plant for the necessary health physics counting, badging and training.

A first plant walkdown is performed in Step 4. This walkdown is performed to inspect the plant and confirm that the plant's configuration is such that the rules developed for doing a margins assessment are applicable and that components can be screened out based on the generic evaluation. Assuming this is the case, the appropriate components are eliminated from further consideration. During this walkdown, any system interactions, system dependencies, and plant unique features will be identified along with confirmation of the accuracy of the system descriptions and configurations.

During the first walkdown of the Maine Yankee plant, the analysis team members formed into groups and inspected components and plant areas that were identified during the previous steps. The Peer Review Group and NRC personnel also formed into groups to walk down the plant. The walkdowns were performed with various levels of detail depending on the requirements of a particular group. Arrangements were made to have team meetings at the beginning and end of each day. Meetings were also arranged with knowledgeable plant personnel to discuss details about the plant and the review.

Following the completion of Step 4, many of the components identified as belonging to the two important plant safety functions were screened out based on the inspection and their generic HCLPF capacities being larger than 0.3g. In addition, plant information that was gathered during this review was used to revise the plant models and perform a conservative evaluation of the remaining component HCLPF capacities. Those components that were evaluated to have a HCLPF capacity larger than 0.3g were also screened out.

During Step 5, the information and understanding of the operation of the plant following Step 4 are used to review and revise, if necessary, the event trees developed in Step 3. Fault trees are developed during this step for systems that perform the two safety functions (subcriticality and early core cooling).

A system fault tree is a logic diagram that models the various parallel and sequential combinations of faults that will result in the occurrence of the predefined undesired top event. For the seismic margins review, the faults are associated with component seismic failure capacities, human error, and other pertinent nonseismic failure events which can lead to system failure following an earthquake. A fault tree thus depicts the logical interrelationships of basic events that lead to the occurrence of the top event, system failure.

A fault tree is a diagram of "gates" which serve to permit or inhibit the passage of fault logic up the tree. The gates show the relationships of events needed for the occurrence of a "higher" event. The "higher" event is the "output" of the gate; the "lower" events are the "inputs" to the gate. The type of gate denotes the relationship between the input events required for the output event.

The two basic gates in a fault tree are the "OR" gate(+), and the "AND" gate(*). The OR gate indicates that the occurrence of the output event will result from the occurrence of any of the input events. The output failure of an OR gate occurs if any of its inputs fail. The AND gate indicates that the output event will only occur if all of the input events occur.

An example of a simplified fault tree is shown in Figure 2-3. This figure shows that the system (SYS1) will fail if both components (COMP1 and COMP2) fail. COMP1 fails if either basic event (BE1 or BE2) occur, and COMP2 fails if both basic events (BE3 and BE4) occur. The basic event represents the failure modes of individual components.

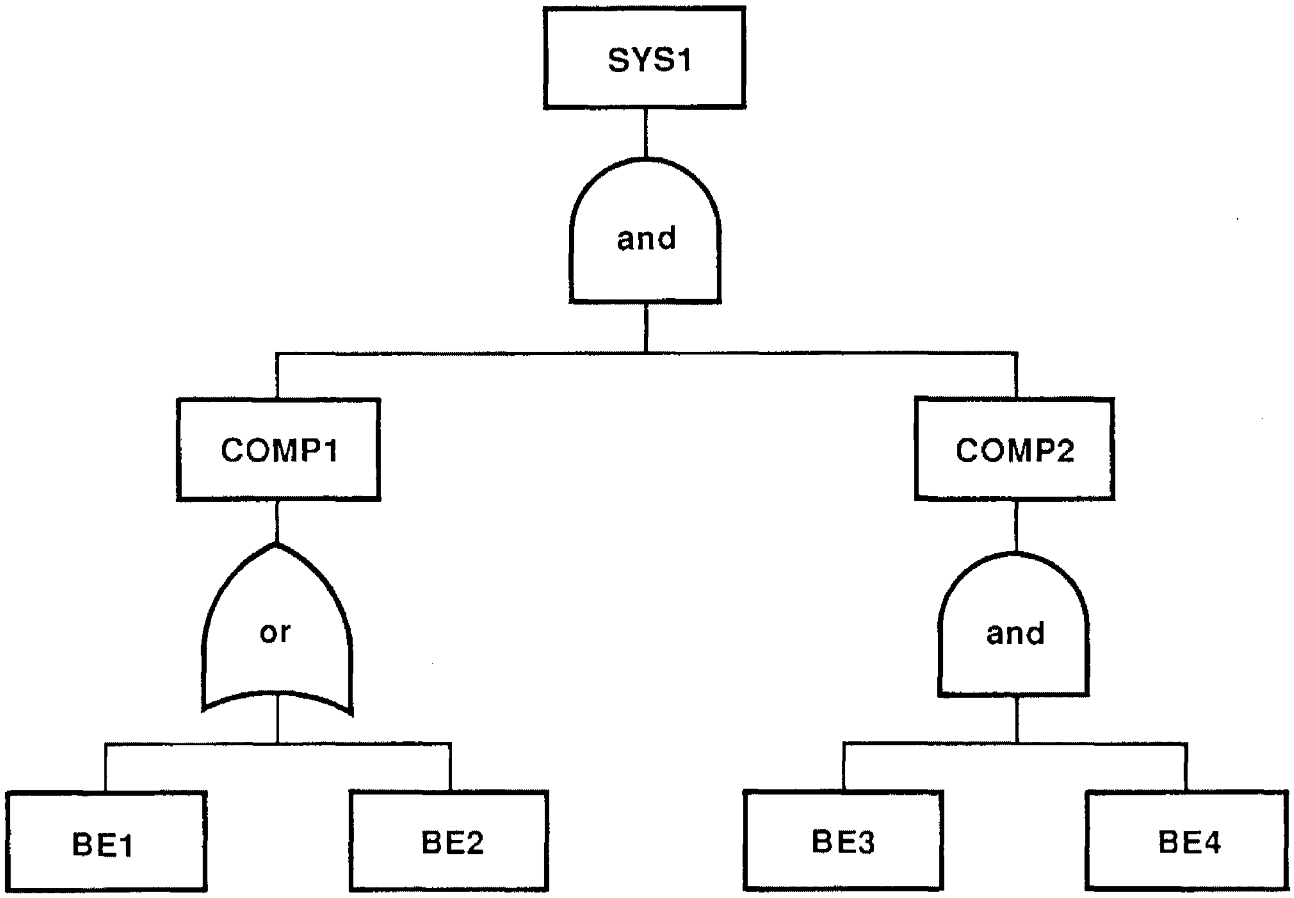
The Boolean expression for the system failure, SYS1, is:

$$SYS1 = COMP1 * COMP2, \tag{2}$$

where the mathematical representation of the logical AND operator is multiplication.

The logical expressions for the component failures, COMP1 and COMP2, are:

$$COMP1 = BE1 + BE2, \tag{3}$$



SYS1 = COMP1 * COMP2
COMP1 = BE1 + BE2
COMP2 = BE3 * BE4
SYS1 = (BE1 + BE2) * (BE3 * BE4)
SYS1 = BE1 * BE3 * BE4
+ BE2 * BE3 * BE4

Figure 2-3 Example of a simplified fault tree

$$COMP2 = BE3 * BE4, \tag{4}$$

where the mathematical representation of the OR operator is addition.

The system Boolean expression in terms of the basic events is then derived by replacing the component failures with their respective logical expressions. The system Boolean becomes:

$$SYS1 = (BE1 + BE2) * (BE3 * BE4), \tag{5}$$

or after expansion,

$$\begin{aligned} SYS1 = \; & (BE1 * BE3 * BE4) \\ & + (BE2 * BE3 * BE4). \end{aligned} \tag{6}$$

Each term in expression (6), separated by the addition operator, is called a "cut set." A minimal cut set represents the smallest number of events that will cause system failure if all the events fail.

The above simplified example is intended to show the basic concepts of fault trees and systems analysis. Normally, the analysis of fault trees is very complex and requires the use of computers. Logical and Boolean expressions are usually very large and can contain thousands of terms that need to be reduced using Boolean algebra.

A more detailed discussion of fault tree analysis is given in the Fault Tree Handbook [USNRC, 1981] along with [Carlson, 1983].

For the Maine Yankee review, fault trees were developed for the front-line and support system that perform the two important safety functions. For the RCS integrity event, a fault tree was developed that consisted of an OR gate containing three inputs, as shown in Figure 2-4. These inputs are small pipe ruptures, a failure in one of the power-operated relief valves (PORV) causing it to remain open, and a failure of a main coolant pump seal causing a small LOCA. During the review, it was possible to screen out the PORV failures and the pump seal LOCA because their components were estimated to have HCLPF capacities greater than the review earthquake level.

The small pipe ruptures, however, could not be screened out. This was due to radioactivity concerns, because components within the Maine Yankee containment structure were unaccessible for review during the walkdowns. In particular, instrument impulse lines that form part of the RCS pressure boundary could not be inspected or reviewed. Therefore, these lines could not be screened out and were assumed to be a source of a small LOCA at the Maine Yankee plant. This prompted the development of two event trees. One that considered a small LOCA concurrent with the LOSP and the other that considered the LOSP with no LOCA.

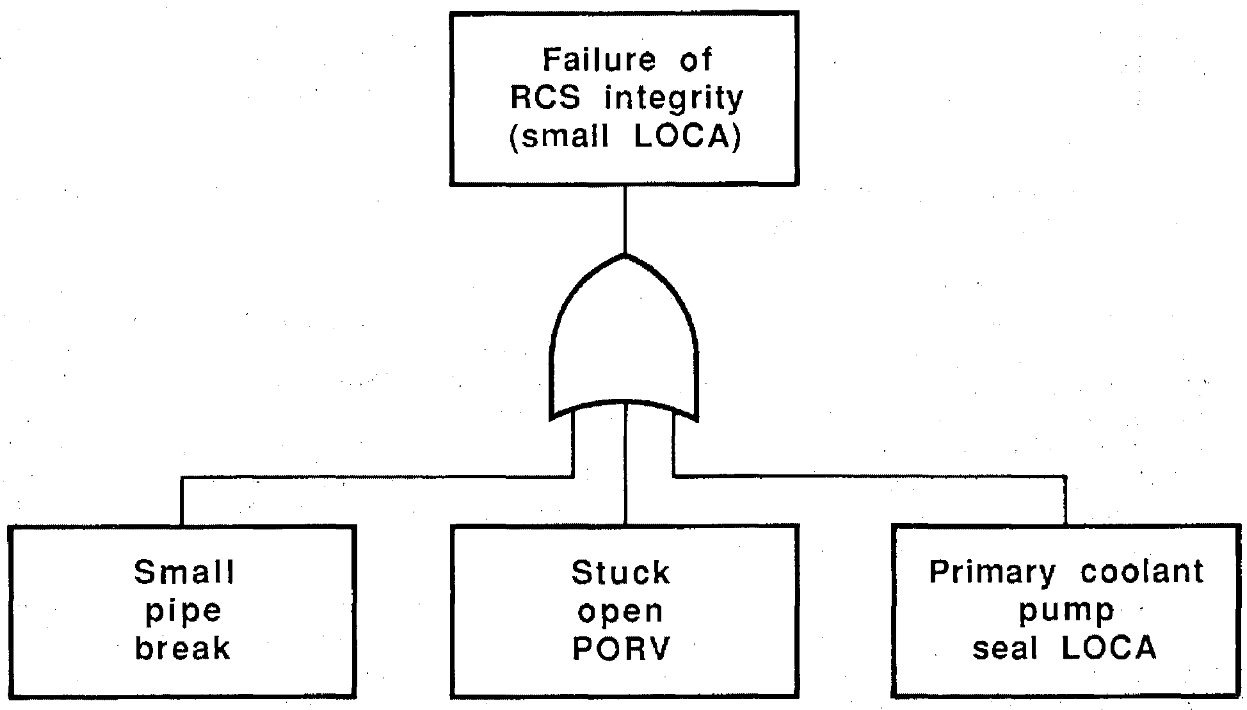


Figure 2-4 Fault tree for failure of RCS integrity.

At the completion of Step 5, the systems fault trees were "pruned" by removing those components that had been screened out in the previous steps. Care was taken when pruning the fault trees so that the paths from the remaining lower level components were left intact. These lower level components represented the possible failures for the systems under consideration.

A final plant walkdown is performed in Step 6. This walkdown is used to obtain additional specific information for determining the HCLPF capacities of the components that remain in the analysis. In addition, the systems models are verified for accuracy and any additional information needed to complete them is obtained.

During the second walkdown of the Maine Yankee plant, the fragility analysts collected detailed information about those components for which a complete evaluation was necessary. This information included detailed measurements of anchorages, equipment supports, and structural details along with gaining an understanding of the general physical condition of equipment, their mechanical layouts including the peripheral components, and collecting other information needed to assess capacities.

The systems analysts collected information needed to finalize their systems models. This included discussions with knowledgeable plant operations and maintenance personnel. The information collected included details concerning scheduled test and maintenance frequencies and durations for the components, and collecting plant data on equipment failures and repair times.

During Step 7, the systems models developed in Steps 3 and 5, and finalized in Step 6, are analyzed to determine the Boolean expressions for the seismic-induced core damage accident sequences. This step involves the analysis of the event trees to determine the accident sequences that lead to seismic core damage and the analysis of the fault trees to determine the Boolean expression for each system failure.

The development of accident sequence Boolean expressions follow in the same manner as the development of the systems Boolean expressions. For the accident sequence, the systems identifiers are replaced with their respective Boolean expressions. The sequence expression is reduced and the minimal cut sets are determined. The fragility curves and nonseismic failures for the basic events are then used to quantify the accident sequences.

A plant level Boolean expression can be derived by logically combining all the core damage accident sequence Boolean expressions. The initiating events for these plant level accident sequences are assumed mutually exclusive. Therefore, the occurrence probability for each initiating event has to be determined before the accident sequences can be combined.

This probability can be compared to the fraction of the time one initiating event occurs with respect to the occurrence of all the accident initiators. Since we consider a fraction for the occurrence of each accident sequence, we multiply accident sequences by a factor called a split fraction. The sum of all the split fractions is 1 since we assume that an initiating event always occurs. For example, during the SSMRP study of Zion, the small LOCA initiating event for a 0.3g earthquake was determined to occur about 1% of the time.

The plant Boolean expression is then derived by multiplying each accident sequence by the appropriate split fraction and logically combining all the sequences. For the Maine Yankee review, two plant level accident sequences were eventually developed. One for a LOSP initiator concurrent with small LOCA and the other for a LOSP initiator with no LOCA.

The final step in the margin review process, Step 8, is to calculate the HCLPF capacities for the important low-capacity components, important systems, accident sequences, and the plant. The HCLPF capacities are finalized for those components that appear in the single, double, and some low-capacity triple member cut set of the Boolean expression derived from the above systems analyses. These HCLPF calculations required detailed structural/mechanical analyses based on information gained in the previous steps. The fragility curves for the components are then used to quantify the Boolean expressions for the system failures, accident sequences and the plant.

There are two methods available to calculate the HCLPF capacity of components: the conservative deterministic failure method (CDFM) and the fragility analysis method (FA). For this trial review, the fragility method was used to calculate the HCLPF capacities for these components. This method was employed because the fragility analysis team has a detailed understanding of its application and use. In addition, the fragility method also allows the inclusion of nonseismic failures into the overall plant HCLPF and accident-sequence HCLPF calculations.

For the FA method, a component's fragility is represented by a simple model using three parameters: median capacity A_m, and logarithmic standard deviations β_R and β_U representing, respectively, randomness in the capacity and uncertainty in the median value. Using a double lognormal model, fragility curves like the one shown in Figure 2-5 are developed. The median, β_R and β_U are estimated using design-analysis information, test data, earthquake experience data, and engineering judgment. The median capacity may be estimated as a product of an overall median safety factor times the SSE, where the overall safety factor is a product of a number of factors representing the conservatisms at different stages of analysis and design. When the scaling of response is not appropriate (e.g., soil sites), the median capacity is evaluated using median structural and equipment response parameters, median material properties and ductility factors, median capacity predictions, and realistic structural modeling and methods of analysis.
The HCLPF capacity is expressed using this fragility model as:

$$\text{HCLPF} = A_m \exp\left[-1.64\,(\beta_R + \beta_U)\right] .$$

The FA method allows the combination of nonseismic and seismic failures for the determination of plant HCLPF capacities. A more detailed explanation of calculating component HCLPF capacities can be found in [Kennedy and Ravindra, 1984].

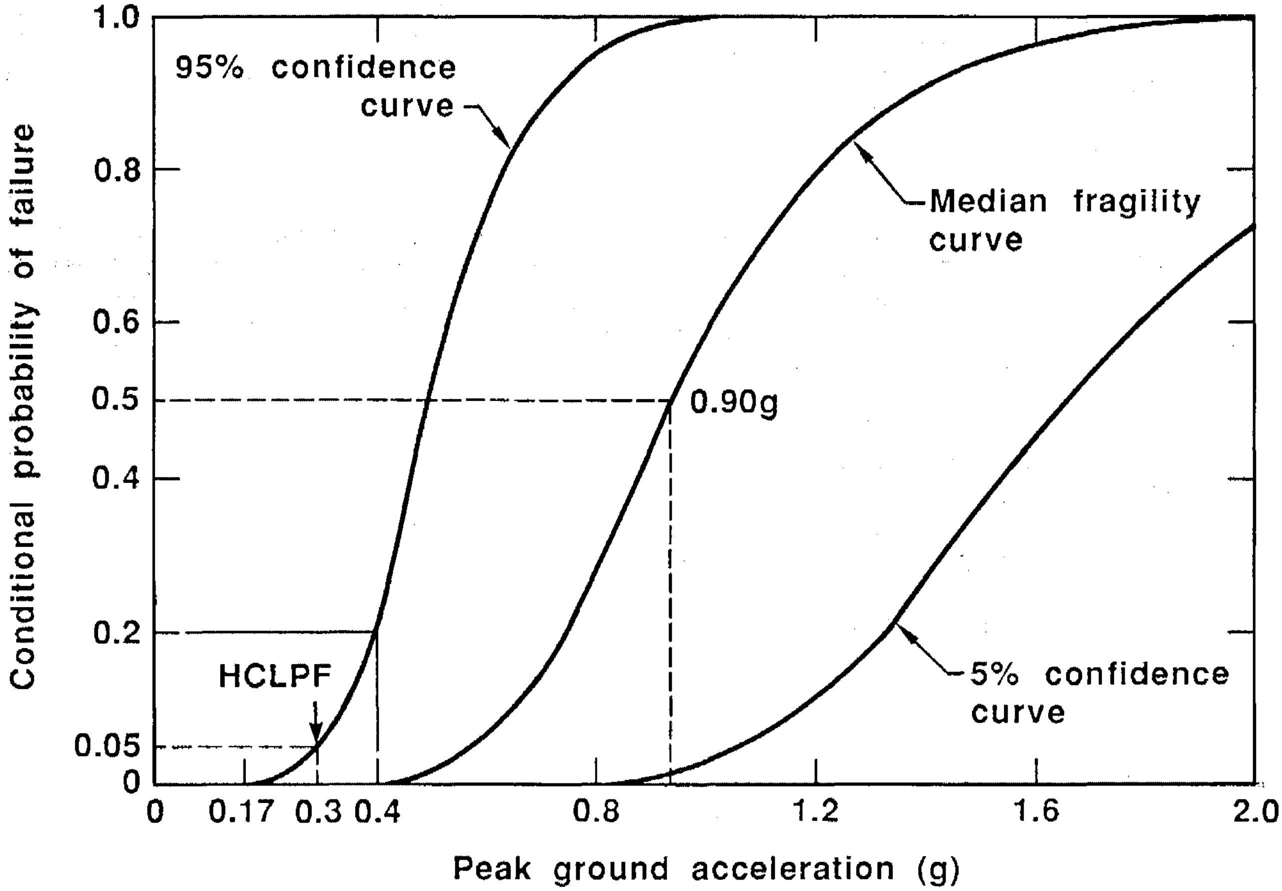


Figure 2-5 Example fragility curves for a structure.

The unavailability of the components (failure per demand) is determined by combining its random failure probability with its unavailability due to normal test and maintenance. Equipment unavailability also included both planned and unplanned maintenance and repair. The unavailability due to random failures considers the time between normal test and maintenance, or between scheduled plant outages. A component's unavailability due to human error is considered as a separate event.

The component failures are combined following the rules of Boolean algebra and a discrete probability distribution (DPD) numerical procedure described by [Kaplan, 1981]. For this purpose, the component fragilities are discretized into a family of fragility curves. Each fragility curve is assumed to be completely described by the median capacity and the value of β_R. The fragility curve is not truncated in either the lower or upper tail.

The "union" operation is performed for the first two components to obtain the composite fragility curves which are in turn combined with the next component fragility curves with either "union" or "intersection" operation in the Boolean expression. After each operation, the resulting fragility curves are condensed to keep the computation manageable.

The details of the analysis are given in Volumes 2 and 3.

CHAPTER 3

3. RESULTS OF SEISMIC MARGIN REVIEW

An objective of this trial seismic margins review is to assess the capability of the Maine Yankee plant to withstand a specified earthquake level greater than the SSE. The results of this review consist of:

- o Boolean expressions for each seismic-induced core-damage accident sequence.

- o The dominant component for plant seismic safety.

- o An assessment of the HCLPF capacities for important components, accident sequences, and the plant.

- o Insight into the seismic behavior of the plant systems required to fulfill the safety functions of subcriticality and early emergency core cooling injection.

The HCLPF capacities for the important components are logically combined as indicated by the Boolean expressions to estimate the HCLPF capacity for each core-damage accident sequence. Each of these accident-sequence HCLPF capacities represents a plant HCLPF capacity for the particular initiating event and plant systems response.

A plant level Boolean expression can be derived by logically combining the accident sequences after they have been multiplied by their respective split fraction. An overall plant HCLPF capacity can then be determined from the plant level Boolean expression.

Section 3.1 presents a summary of the overall result of the Maine Yankee review. A summary of the systems results, including a discussion of the accident sequences, is given in Section 3.2. Details of the systems analysis are in Volume 2. A summary of the component capacity assessment is given in Section 3.3. Details of the component capacity assessment are in Volume 3.

3.1 Overall Results

Following the review of plant information, a list of the components that make up the front-line and support systems required to perform the plant safety functions was developed. This list is given in Volume 3 and provided the basis for the remainder of the review. After the plant walkdowns and subsequent analyses by the systems and fragilities analysts, the list of components was reduced by screening out those components that had HCLPF values greater than 0.3g. For the remaining components, HCLPF capacities were calculated. The list of components for which a HCLPF capacity was calculated is given in Section 3.3.

These remaining components were used in the development of the event trees and fault trees for the seismic-induced core-damage accident sequences as described in Volume 2. The event trees and fault trees were analyzed to determine Boolean expressions for each accident sequence that could lead to core damage. The component failures that are significant to these Booleans are given in Tables 3-1 and 3-2. Table 3-1 gives the seismic induced failures along with the fragility parameters used to quantify their HCLPF capacities. Table 3-2 gives the nonseismic failures and their unavailabilities. Note that the component items and nonseismic failure events are numbered consecutively; the missing numbers represent the items that were screened out in the final pruning of the event and fault trees.

One component has been upgraded following the review during the two plant walkdowns. This component (number 4, Table 3-1) is a General Electric (GE) station service transformer (4160 V to 480 V) that supplies power to pumps and components needed to perform the seismic safety functions.

This transformer was originally installed with a "floating" bus bar to limit the amount of noise on the system. Several years after its installation, GE performed seismic qualification testing on these types of transformers and made modifications that essentially resulted in securing the "floating" bus bar and greatly increasing seismic capacity. Preliminary estimates of the transformer HCLPF capacity were approximately 0.1g. More detailed calculations could have been performed, however, the utility decided to upgrade this component during its March 1987 outage. An analysis of this planned upgrade has shown an increase in the estimated HCLPF capacity for the transformer to 0.30g.

Another component appearing in Table 3-1 with a HCLPF less than 0.3g is Number 20, the seismic failure of the primary water storage tank (PWST). This tank provides an alternate supply of water to the auxiliary feedwater system. The PWST has a HCLPF of 0.27g. This tank does not appear in either of the core damage Boolean expressions.

There is one type of component, not listed in Tables 3-1 or 3-2, for which there was insufficient data to determine the HCLPF capacity. This component is the lead-antimony type batteries used at the Maine Yankee plant. The four sets of Station Batteries (1, 2, 3, 4) have passed their electrical qualification testing. However, there is no seismic qualification or test data available on these types of batteries to estimate the seismic capacity of the aged lead-antimony plates within the battery casings.

Systems analysis indicated that the important batteries to plant seismic safety are Station Batteries 1 and 3. Maine Yankee had intended to change out one of the station batteries during their plant refueling outage scheduled for March 1987. This analysis prompted them to replace both Station Batteries 1 and 3. The remaining Station Batteries 2 and 4 have been scheduled for replacement during the 1988 outage.

Table 3-1 Component seismic fragility parameters.

Item No.	Item	A_m(g)	β_R	β_U	HCLPF Capacity (g)
4	Transformers (X507, X508)	0.84	0.30	0.32	0.30
7	RWST (TK-4)	0.45	0.20	0.25	0.21
8	DWST	0.36	0.20	0.26	0.17
20	Circulating Water Pumphouse	0.69	0.24	0.27	0.30
21*	PWST (TK-16)	0.57	0.20	0.26	0.27

* HCLPF less than 0.3g, but does not appear in the plant Boolean expressions.

Table 3-2 Probabilities for nonseismic failures.

Item No.	Description	Median Unavailability (per demand)	Error Factor*
10	Operator Failure to Close PCC Isol. Valves	8.0E-02	2
11	Random Failure of DG-1B	4.2E-02	5
12	Random Failure of DG-1A	4.2E-02	5
13	Operator Failure to Place AFW Pump Train B in Service Locally	1.5E-01	2
14	Nonseismic Common Cause Failure of DGS	1.6E-03	5
15	Nonseismic Common Cause Failure of AFW	1.2E-04	5
16	Operator Failure to Refill DG Fuel Tanks by Opening Valve or Running P-33A,B	8.0E-03	3
17	Operator Failure to Place AFW Pump Train B in Service from MCR	4.0E-02	3
22	Random Failure of the Turbine Driven Aux Feedwater Pump	3.0E-02	5

* Error factor equals (95% Confidence Value/Median Value).

Seismic qualification and test data on the new batteries indicates that they have a HCLPF capacity greater than the review earthquake level. Subsequently, the station batteries were screened out and eliminated from further consideration. The new station batteries and racks should be inspected for proper installation according to Expert Panel guidance after they are in place.

The capacity assessment of the new batteries being installed at the Maine Yankee plant assumes they possess the qualities of new equipment and are installed correctly. However, as with the analysis of all components during this seismic margin review, the effects of aging beyond this "snapshot" of the plant are not considered.

The analysis of the two event trees resulted in two Boolean expressions that lead to seismic-induced core damage. One of these expressions is the logical combination of three accident sequences that were initiated by the seismic-induced LOSP concurrent with a small LOCA. The other expression is the logical combination of two accident sequences initiated by the seismic-induced LOSP without a small LOCA. These two Boolean expressions are given below:

Small LOCA Core Damage

$$= (SL)\ [4 + 7 + 20].$$

No LOCA Core Damage

$$= (LOSP)\ [(4 + 20) * (8 + 13 + 15 + 17 + 22) + 8 * (14 + 16) + 15 * 7],$$

where the numbers in the expressions correspond to the failure of the components given in Tables 3-1 and 3-2. Entries with designators 4, 7, 8, 20, are seismic-induced failures and given in Table 3-1. Entries with designators from 10-17 and 22 are nonseismic failures and given in Table 3-2. The missing numbers in Tables 3-1 and 3-2 are component screened out during the final screening. The terms SL and LOSP represent the small LOCA and loss of offsite power initiating events, respectively. In the above expression, the "+" notation denotes probabilistic addition (union) and the "*" denotes probabilistic multiplication (intersection).

The small LOCA initiating event could be determined using the guidance given by the Expert Panel. This guidance indicates that for this review earthquake level, a walkdown of sample piping system should be conducted along with inspection of piping between buildings to look for possible problems such as weak spots, unanchored attached equipment, stiff sections between flexible termination points, and brittle connections.

During the walkdown of the Maine Yankee plant, none of these problems were identified and piping was found not to be a problem. Therefore, the small LOCA initiating event could be screened out thus eliminating the small LOCA Boolean expression from the seismic margins review. However, the analysis teams felt that eliminating the small LOCA from consideration was not conservative. The possibility of a small LOCA should not be screened out because the containment, which contains many small RCS pipes, could not be inspected and reviewed as previously discussed.

Impulse line failures were assumed to be the source of a small LOCA at Maine Yankee. This conservative assumption was required due to the tremendous number of hours which would be required to walk down each of these impulse lines and assess potential system interaction problems. These lines originate from the primary pressure boundary inside containment (i.e., RPV, steam generator, pressurizer, primary coolant loop piping, etc.) and are field routed to instrument racks inside containment. The amount of work required to demonstrate the seismic margin in each of these lines plus the fact that walkdown of these lines would have to take place during a plant outage necessitated the assumption of a small LOCA as an initiating event.

Inspection of the small LOCA core-damage Boolean expression indicates that the dominant components are the three singletons 4, 7, and 20. The singleton component with the lowest HCLPF capacity is the refueling water storage tank (RWST, number 7) with a HCLPF capacity of 0.21g. Failure of this tank results in no coolant being available for reactor vessel injection following a small LOCA.

The other singleton components have HCLPF capacities of 0.30g. The capacity of the transformer (number 4) is estimated based on the proposed upgraded condition. The other singleton is the circulating water pumphouse (Number 20).

For the no LOCA core-damage Boolean expression, the occurrence of the LOSP initiating event is assumed a certainty (probability = 1.0) due to the low capacity of switchyard components. Therefore, the LOSP initiating event could be removed from the no LOCA Boolean expression.

The HCLPF capacity for the no LOCA core damage Boolean expression is estimated to be greater than 0.30g. The higher capacity for this sequence, compared to the small LOCA sequence, is due to the absence of singletons and no low capacity doubletons in the expression. Although the DWST, i.e., component 8, with a HCLPF capacity of 0.17g appears in this sequence, its failure has to occur simultaneously with one of the higher capacity components, i.e., the transformer or the circulating water pumphouse.

The most important nonseismic failure is number 15, a common cause failure of the auxiliary feedwater system caused by steam binding. This failure results in the inability to cool down the reactor coolant systems using the steam generators. This nonseismic failure is the most important because it appears in a majority of the doubleton cut sets.

Figures 3-1 and 3-2 show plots of the small LOCA and LOSP core-damage fragility curves, respectively, in which the family of fragility curves is reduced to the 5%, 50%, and 95% confidence levels.

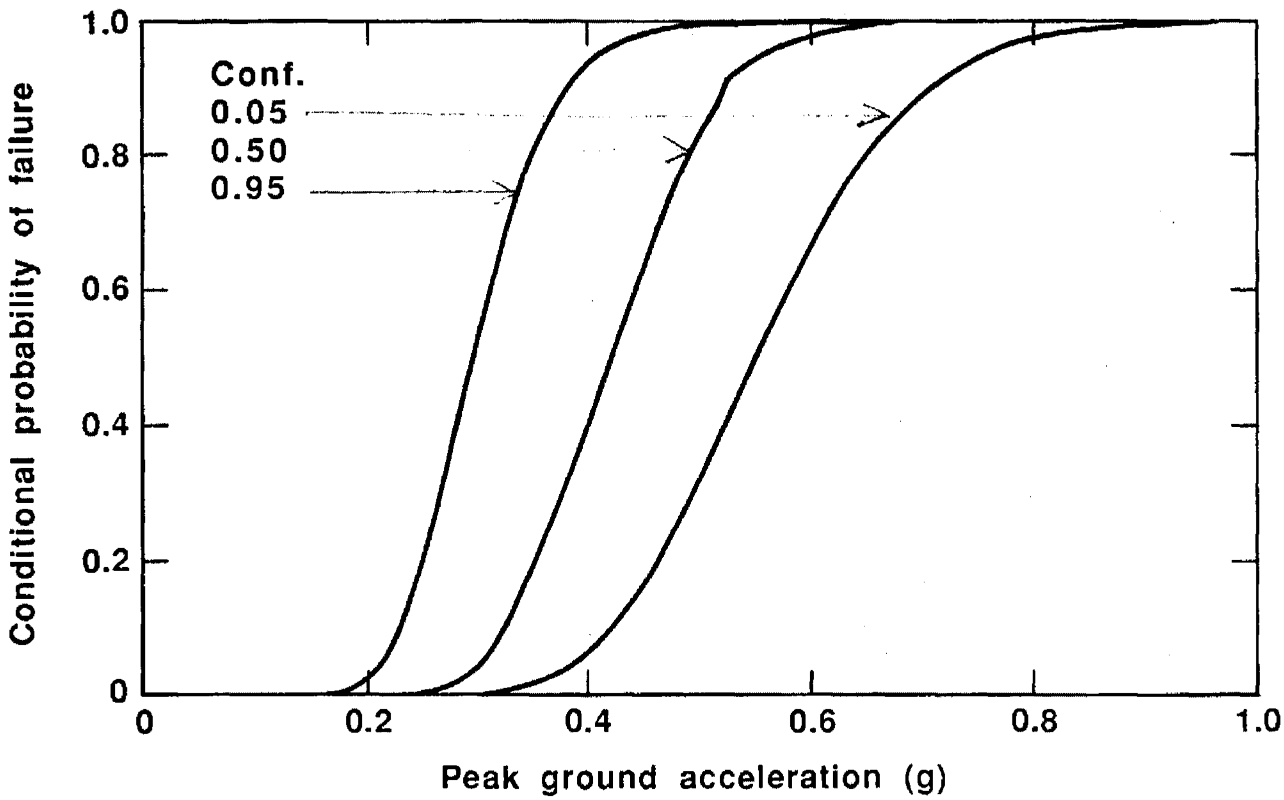


Figure 3-1 Fragility curves for small LOCA core damage

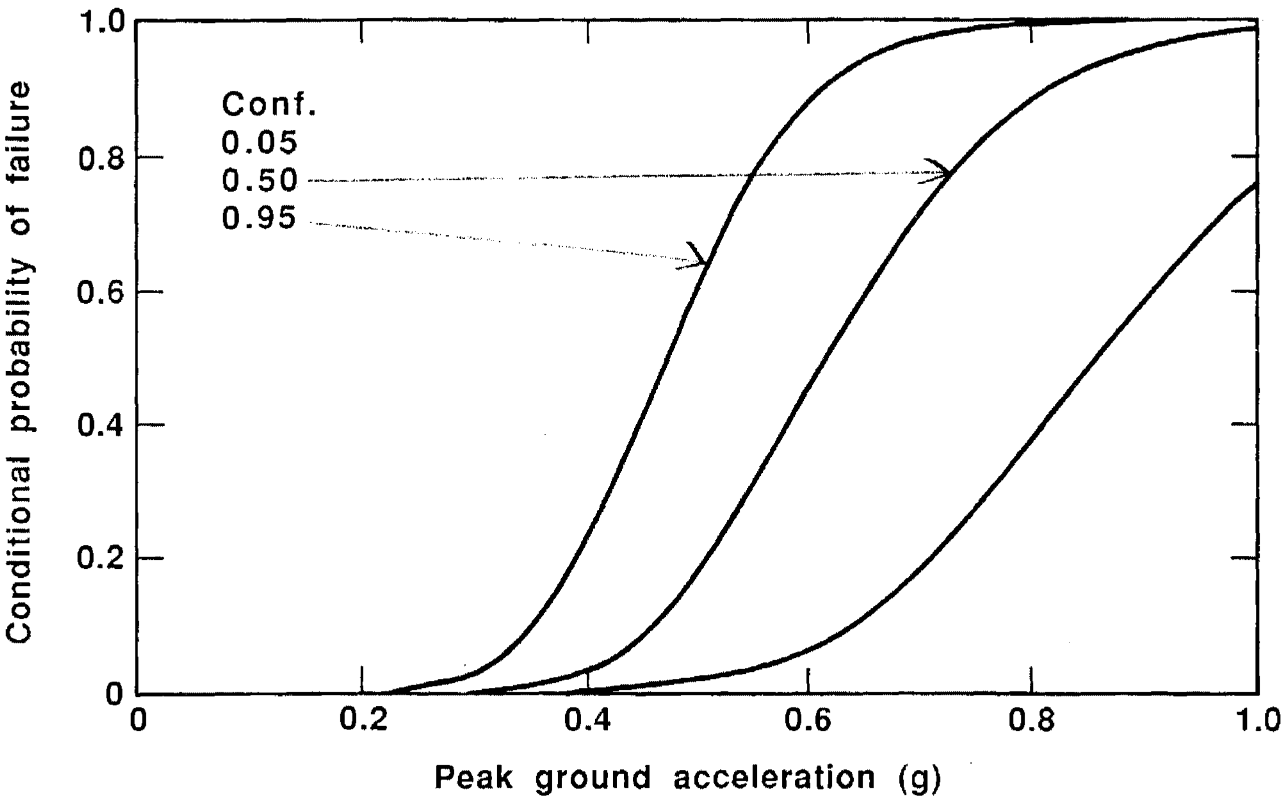


Figure 3-2 Seismic Fragility Curves for No-LOCA Core Damage.

To account for the fact that we could not quantify the small LOCA initiating event, we can develop an overall plant core-damage Boolean expression by logically combining the two Boolean expressions accounting for the split fraction p between the small LOCA and LOSP initiating events. This overall plant core-damage Boolean expression is given below:

Core Damage

= p [small LOCA core damage] +

(1 - p) [no LOCA core damage]

In this case, the split fraction accounts for the fraction of the time a small LOCA will occur along with the LOSP event.

When evaluating the HCLPF capacity for the small LOCA Boolean expression, there must be a consideration for the occurrence of the small LOCA initiating event. If the piping, whose failure will cause the small LOCA, has a HCLPF capacity lower than all the singleton components in this Boolean expression, the lowest capacity component (the RWST, number 7) will dominate the small LOCA core damage HCLPF capacity. If, however, the piping has a HCLPF capacity larger than the lowest capacity singleton, then the piping will dominate the small LOCA HCLPF capacity.

Overall Plant Core Damage HCLPF Capacity

As indicated above, to account for the two Boolean expressions in the overall plant core-damage HCLPF capacity, a sensitivity calculation can be performed accounting for a variation in the split fraction between the two accident-sequence initiating events (small LOCA and no LOCA). For different assumed split fraction values, the overall plant core-damage HCLPF capacities were obtained as shown in Table 3-3.

The conclusion regarding the dominance of RWST failure in the HCLPF capacity estimation (displayed in the small LOCA accident sequence) is a function of the split fraction assumed. If the plant HCLPF capacity needs to be increased, it is not necessary to concentrate only on RWST. A walkdown and review of small impulse lines within the containment may be performed to estimate their fragilities in order to assign a realistic HCLPF capacity or split fraction. By this procedure, the plant HCLPF capacity may be shown to be higher without the necessity of any upgrading of the components.

Effect of Nonseismic Failures

The overall plant HCLPF capacity was calculated using the Boolean expressions for the core-damage accident sequences which contained both seismic (Table 3-1) and nonseismic (Table 3-2) failures. Since this is a seismic margin review and the interest is only in the seismic capacity of the plant, one may choose to ignore the nonseismic failures in calculating the overall plant HCLPF capacity. In the small LOCA Boolean expression, there are no significant nonseismic failures. The effect of not including the nonseismic failures on the HCLPF capacity of the no LOCA Boolean had no effect on the plant HCLPF.

Table 3-3 Summary of plant level HCLPF capacities.

Case	Description		HCLPF Capacity (g)
1	Small LOCA		0.21
	- Independent Seismic Failures		
2	Small LOCA		0.21
	- Dependent Seismic Failures		
3	No LOCA		≥ 0.30
	- Independent Seismic Failures with Nonseismic Failures		
4	No LOCA		≥ 0.30
	- Independent Seismic Failures without Nonseismic Failures		
5	Core Damage		
	- Split Fraction	p = 0.01	≥ 0.30
		p = 0.10	0.28
		p = 0.50	0.23
6	Small LOCA		
	- RWST Reduced Fluid Level		0.26

Correlation Between Seismic Failures

The above calculations were performed assuming perfect independence between seismic failures of different components; i.e., the seismic capacities are assumed to be statistically independent both in randomness and uncertainty. This is a realistic assumption because the components involved in the core-damage Boolean expression are yard tanks (RWST, and DWST), transformer and circulating water pumphouse. These are dissimilar items, their locations in the plant are different, and their dynamic characteristics are different. Hence, correlation in the seismic responses and capacities of these components is judged to be minimal.

It is realistic, however, to expect some correlation in the component failures. The assumption of perfect dependence in both uncertainty and randomness is an extreme case. Assumption of perfect dependence in the uncertainties of different component fragilities means that the median ground acceleration capacity of a component is known if the median ground acceleration capacity of another component is known. Since uncertainty arises from an insufficient understanding of structural material properties, approximate modeling of the structure, inaccuracies in the representation of mass and stiffness, and the use of engineering judgment in lieu of plant specific data, it is expected that all components will be affected to some degree by these uncertainties. Therefore, some probabilistic dependence between component median capacities may be expected.

Dependence in the randomness arises from a common earthquake generating the responses in different components and common structural/material properties. Assumption of dependence in the randomness means that if the fragility (conditional probability of failure) of a component for a given peak ground acceleration is known, the probability of failure of the other components is somewhat modified by that knowledge.

The plant level fragility, and therefore, the HCLPF capacity depends on the degree of dependence in randomness and uncertainty between the component failures. However, the degree of dependence is difficult to estimate. One approach is to bound the core-damage HCLPF capacities by assuming perfect dependence as opposed to the case of perfect independence. This calculation was performed for the small LOCA core-damage Boolean expression given above. The small LOCA core-damage HCLPF capacity is estimated to be 0.21g, i.e., governed by the capacity of RWST. If the seismic failures were assumed to be perfectly dependent and the nonseismic failures were assumed to be perfectly independent. The plant level HCLPF is estimated to be 0.21g.

When the Boolean expression is dominated by singletons, the assumption of perfect independence is more severe than the assumption of perfect dependence between failures if the fragilities are approximately equal; if there is a single component with a very low capacity compared to the rest of the components in the Boolean expression consisting of singletons, both assumptions give about the same plant level HCLPF capacity.

The question of dependence between failures is important when there are similar components experiencing common seismic excitation. The case in point is the yard tanks (i.e., RWST, DWST, and PWST). By reviewing the cut sets, it was found that a tripleton cut set could lead to core damage. It is

$$DWST * PWST * RWST.$$

Although high dependence between the failures of tanks is possible and their seismic capacities are similar, the cut set should have a HCLPF capacity of 0.27g.

Deterministic Method

This approach is based on the assumption that the HCLPF capacities of components are true lower bound values. The HCLPF capacity of the plant is obtained directly by studying the Boolean expressions for small LOCA and no LOCA:

Small LOCA Core Damage

$$= 4 + 7 + 20$$

No LOCA Core Damage

$$= (4 + 20) * (8 + 15 + 17 + 22)$$

$$+ 8 * (14 + 16) + 7 * 15.$$

For calculating the plant level HCLPF capacity in this method, the nonseismic failures are ignored. Also, the cut sets that includes low probability nonseismic failures are also omitted from this estimation. Therefore, the simplified no LOCA Boolean expression becomes:

No LOCA Core Damage

$$= (4 + 20) * 8.$$

In the deterministic method, the HCLPF capacity of a "doubleton" cut set is represented by the higher of the two component HCLPF capacities. The HCLPF capacity of a "tripleton" cut set is represented by the highest of the three component HCLPF capacities. The HCLPF capacity of a union of singleton cut sets is estimated to be the lowest of all the component HCLPF capacities. Using this procedure, the HCLPF capacities of the core damage accident sequences are:

HCLPF Capacity for Small LOCA Core Damage

$$= \min [0.30, 0.21, 0.30]$$

$$= 0.21g.$$

HCLPF Capacity for No LOCA Core Damage

$$= \max [\min (0.30, 0.30), 0.17]$$

= max [0.30, 0.17]

= 0.30g.

Note that the failure of the DWST, with a HCLPF capacity of 0.17g, is not governing the plant level HCLPF capacity because the DWST has to fail simultaneously with one of two nonseismic failures to lead to core damage. Since these failures have median probabilities of 6×10^{-3} and 8×10^{-3}, respectively, it is appropriate to ignore the DWST failure in the plant level HCLPF capacity calculation.

Sensitivity Studies

Two sensitivity studies were performed to assess the effect of certain assumptions on the plant HCLPF capacity. These sensitivity studies addressed the following:

- o Effect of shearwall stiffness reduction on structure response and equipment seismic input.

- o Reduction of RWST fluid level.

For the first sensitivity, on-going scale model testing at Los Alamos National Laboratory being sponsored by the NRC has indicated potential reductions in the stiffness of concrete shear walls of up to a factor of 4, from elastically calculated values due to cracking. This would imply a reduction in the elastic structure frequencies of up to 50%. These variations were considered at the suggestion of the Peer Review Group, however the research results are still preliminary.

For no LOCA, the 4160/480-V transformer is the dominant contributor to the plant HCLPF capacity. The transformer has a 12-13 Hz fundamental frequency which corresponds to a spectral acceleration on the downward slope of the floor response spectra. A reduction of the building frequencies would result in a reduction of the seismic input to the transformer with a corresponding increase in its HCLPF capacity. As a further check on the effect of the building frequency shift, a similar evaluation was made on each of the components within the final Boolean expression. The potential building frequency reduction did not lower the HCLPF capacity for any of these components. Thus, for the purpose of the Maine Yankee seismic margin study, the shear wall stiffness reduction and resulting building frequency shift does not affect the plant seismic margin.

For the second sensitivity study, the effect of reducing the fluid level within the RWST was investigated. A reduction of the fluid level to an arbitrary height of 33 ft, which is approximately equal to 10% reduction from the current level of 37 ft, was assumed. This change leads to a reduction in the effective fluid weight, mass, centroid height, and overall tank seismic loads. The RWST HCLPF capacity is increased to 0.28g with a corresponding increase of the small LOCA HCLPF capacity to 0.26g.

3.2 Systems Results

A review of the Maine Yankee plant information resulted in identifying five front-line systems that are required to fulfill the safety functions of reactor subcriticality and early ECC injection. These systems are:

- The reactor protection system (RPS) including the control rod drive system used to shutdown the nuclear chain reaction in the core and the core internals through which the rods pass.

- The boric acid transfer system used to provide emergency boration to the reactor system and also shut down the nuclear chain reaction in the core.

- The high pressure safety injection (HPSI) system used to supply coolant to the primary system.

- The auxiliary feedwater system used to cool down and depressurize the primary system through the steam generators.

- The pressurizer power-operated relief valves (PORV) used in a feed and bleed mode to depressurize the primary system.

The support systems to these front-line systems are shown in matrix form in Table 3-4. The support system versus support system matrix is shown in Table 3-5. The components that make up the front-line and support systems are identified and listed in Volume 3.

During the review of the plant and as a result of discussions with Combustion Engineering (CE, the NSSS supplier), it was determined that the components that make up the RPS have HCLPF capacities larger than the review level earthquake level based on data and calculations performed by CE. These components included the control rod drive mechanisms, core internals, and the RPS actuation systems. Subsequently, the reactor subcriticality function was screened out and eliminated from further consideration. Screening out the reactor subcriticality function also eliminates the need for consideration of the boric acid transfer system.

Systemic event trees were developed for (1) seismic-induced loss-of-offsite power (LOSP) concurrent with a small LOCA and (2) LOSP only initiating events. Fault trees were developed for the front-line and support systems using the identified seismic components that were not initially screened out. The fault trees incorporated nonseismic failures including random failure, common cause failure, and operator error. A discussion of the success criteria, assumptions, and bases for the fault trees is given in Volume 2.

When modeling the primary component cooling system (PCCS), the support system which provides cooling to one train of required front-line equipment, it was discovered that a large portion of the system cools nonessential shutdown equipment which is not automatically isolated following an earthquake.

Table 3-4 Front-line system vs. support system dependency matrix.

		SUPPORT SYSTEMS								
		AC Power		DC Power		CCW		SIAS		IA
FRONT-LINE SYSTEMS		Bus 5 Bus 7	Bus 6 Bus 8	DC-1 DC-2	DC-3 DC-4	PCC	SCC	Train A	Train B	TK-25
HPSI/ RECIRC	P-14A to RCS	X		X		X		X		
	P-14B to RCS		X		X		X		X	
	FN-44A	X						X		
	FN-44B		X						X	
AFW	P-25A		X		X					
	P-25B									X
	P-25C	X		X						
PROVs	PR-S-14	X								
	PR-S-15		X							

NOTE: To determine the front-line system dependencies on the support systems, locate the front-line component in the first column and read across the row to find the support system dependencies.

CCW = component cooling water
SIAS = safety injection actuation system
IA = instrument air

Table 3-5 Support system vs. support system dependency matrix.

			AC Power										DC Power													
			4160V Bus		480V Bus		120V Vital Bus				DG		125V Bus				CCW		SW	SIAS		HVAC				1A
SUPPORT SYSTEM			5	6	7	8	1	2	3	4	1A	2A	1	2	3	4	PCC	SCC		A	B	FN-20A	FN-20B	FN-31	FN-32	TK-110
AC Power	4160V Bus	5									X		X											X	X	
		6										X			X									X	X	
	480V Bus	7	X										X											X	X	
		8		X											X									X	X	
	120V Vital Bus	1											X													
		2												X												
		3													X											
		4														X										
	Diesel Generator	1A											X				X					X				
		1B													X			X					X			
DC Power	125V Bus	1			X																			X	X	
		2			X																			X	X	
		3				X																		X	X	
		4				X																		X	X	
CCW		PCC	X																X	X	X					
		SCC		X															X							X
SWS					X	X																				
SIAS	Channel	A											X													
		B												X												
		C													X											
		D														X										
	Act.	SIS-A											X													
		SIS-B													X											
HVAC		FN-20A			X																					
		FN-20B				X																				
		FN-31			X																					
		FN-32				X																				

Note: To determine the support system dependencies on other support systems, locate the support system in the first column, and read across the row to determine dependencies on the other support systems.

Therefore, leakage in these components, if not isolated, could compromise the entire system. Much of this equipment, located in the PAB and containment, was not inspected during the walkdowns due to accessibility. This consideration prompted the development of postearthquake procedural guidance requiring the isolation of nonessential portions of the PCCS in the event of a major earthquake and combined plant trip. The nonessential portions of the system can be isolated using valves which are remotely operated from the control room and have HCLPF capacity greater than the review earthquake level. This isolation of nonessential components has no impact on seismic safety. Based on this procedure change, the only remaining failure to be considered is the operator error associated with not following the procedure and closing the valves (component No. 10 in Table 3-2).

The event tree analysis resulted in five accident sequences that led to seismic-induced core damage. Three of these are small LOCA sequences and two are no-LOCA sequences. The three LOCA sequences are designated S2D, S2LD, and S2LP2, and given below:

$$\begin{aligned} S2D &= HPSI \\ S2LD &= HPSI * AFW \\ S2LP2 &= PPS\text{-}LOCA * AFW, \end{aligned}$$

where HPSI, AFW and PPS-LOCA represent the high pressure safety injection system, the auxiliary feedwater system, and the plant pressure protection system operating for a LOCA condition, respectively.

For the seismic small LOCA, core damage will result if the HPSI system fails (S2D), or if both the HPSI and the AFW systems fail (S2LD), or if both the AFW system and power-operated relief valves (PORV) fail (S2LP2). For the LOCA case, both PORVs must fail since only one PORV is required to operate within approximately 30 minutes for feed and bleed. The block valve on each pressurizer relief line is included with its respective PORV.

The combined LOCA Boolean expression is the logical summation of these three accident sequences.

$$\text{LOCA Core Damage} = S2D + S2LD + S2LP2 = S2D + S2LP2$$

The two no-LOCA accident sequences are designated T1LD and T1LP1, and are given below:

$$\begin{aligned} T1LD &= HPSI * AFW \\ T1LP1 &= AFW * PPS, \end{aligned}$$

where the PPS represents the plant pressure protection system operating for the no-LOCA case.

For the no-LOCA case, core damage will result at Maine Yankee if both the AFW and HPSI systems fail (T1LD) or if both the AFW and one PORV fail (T1LP1). Based on PRA results, for the no-LOCA case, only one PORV must fail since both PORVs must open within approximately 30 minutes for feed and bleed.

The systems failures were determined by fault tree analysis. The systems fault trees were developed and analyzed following an 11-step process. A brief description of the 11 steps is given in Volume 2.

The analysis of the fault trees gives the minimal cut sets that lead to system failure. The single-failure (singleton) and double-failure (doubleton) cut sets for the systems given in the accident sequences are shown in Table 3-6. The cut sets shown contain either seismic-only failures or a combination of seismic and nonseismic failures.

The HPSI system contains three seismic singletons and no seismic doubletons that lead to system failure. The AFW system contains no seismic singletons and 10 doubletons. The PPS system contains two singletons and no doubletons operating for both the small LOCA (PPS-LOCA) and no-LOCA (PPS) sequences.

3.3 Fragility Results

The safety system components identified by the systems analysts were reviewed and subjected to three different types of screening.

1. Those components with a generic HCLPF value greater than the review earthquake level, as given in Table 2-1 in [Prassinos et al., 1986], were flagged for inspection during the first plant walkdown. The first plant walkdown confirmed that these components could be screened out based on their generic HCLPF values and the use of the screening table. Not all components with a generic HCLPF value greater than the review earthquake level were screened out after the first plant walkdown.

2. The remaining components were examined following a detailed walkdown review of their seismic capacity during the second plant walkdown. If the seismic capacity was judged to be greater than the review earthquake level, they were screened out.

3. The components remaining after the two screenings listed above were screened based on a calculated HCLPF capacity. If their calculated HCLPF capacity was greater than the review earthquake level, they were screened out. Data needed to calculate the component HCLPF capacities were collected during both plant walkdowns, but specifically during the second plant walkdown.

Both plant walkdowns were also performed to verify the configuration of the plant systems and to look for systems interaction and any plant unique features.

The components remaining after the first screening and those requiring support systems and operator action to perform the safety functions were used in the development of the system fault trees. As the subsequent screenings were performed, the system fault trees were pruned to reflect the elimination of these components.

Table 3-6 Minimal cut sets for important systems.

High Pressure Safety Injection System (HPSI)	
Singletons	1. 4160 V to 480 V station service transformer.
	2. Refueling water storage tank (RWST).
	3. Circulating water pumphouse.
Doubletons	4. No seismic double failures.
Auxiliary Feedwater System (AFW)	
Singletons	1. No seismic single failures
Doubletons	2. Primary water storage tank (PWST), and demineralized water storage tank (DWST).
	3. 4160 V to 480 V station service transformer, and the DWST.
	4. Circulating water pumphouse and the DWST.
	5. The DWST and human error, failure to open AFW pump trains A & B PWST isolation valves.
	6. 4160 V to 480 V station service transformer, and human error, failure to place AFW pump train B (P-25B) in service from the control room.
	7. 4160 V to 480 V station service transformer, and random failure of AFW turbine driven pump P-25B.
	8. Circulating water pumphouse and random failure of AFW turbine driven pump P-25B.
	9. Circulating water pumphouse and human error, failure to place AFW pump B (P-25B) in service from the control room.

Table 3-6 Minimal Cut sets for important systems. (Cont.)

		Auxiliary Feedwater System (AFW) (Cont.)
Doubletons (Cont.)	10.	DWST and human error, failure to refill diesel generator tanks by opening valves and/or running auxiliary fuel pumps.
	11.	DWST and nonseismic common cause failure of the diesel generators.
		Plant Pressure Protection System (PPS) (small LOCA and no-LOCA)
Singletons	1.	Circulating water pumphouse.
	2.	4160 V to 480 V station service transformer.
Doubletons	3.	No seismic double failures.

The screening processes resulted in a reduced list of components for which a HCLPF capacity was calculated. This reduced list has been divided into two lists, one consisting of structures and block walls, and the other consisting of equipment. These two lists are given in Tables 3-7 and 3-8, respectively, along with the calculated HCLPF capacities.

Table 3-7 Maine Yankee Structures and Block Walls

Structure	Construction	HCLPF (g) Capacity
Circulating Water Pumphouse, steel portion above El 21'-0"	Structural steel framing and diagonal bracing, concrete slab	0.3
Service Building, El 39'-0" floor	Structural steel framing and diagonal bracing, concrete slab	0.38
Main Steam Valve House, interior steel structure	Structural steel framing and diagonal bracing, metal grating	>0.3

Wall ID No.	Group A Components and Lifelines	HCLPF (g) Capacity
C 0.5-1 C 20-1	Pressurizer instrumentation (PT-104, LT-106) and tubing	>0.3
SB 21-17	Main control board (aux feedwater system panels)	>0.3
SB 21-18	Main control board (aux feedwater system panels), aux logic panels	>0.3
SB 35-1	Battery groups 3 and 4, safety-related cable trays	>0.3
SB 35-2	Battery groups 3 and 4, safety-related cable trays	>0.3
SB 35-3	Battery groups 3 and 4, safety-related cable trays	>0.3
SB 35-4	Battery groups 3 and 4 >0.3	
SB 35-7	PCC Surge line, PCC temperature controller	>0.3

TABLE 3-7 Maine Yankee structures and block walls (cont.)

Wall ID No.	Group A Components and Lifelines	HCLPF (g) Capacity
SB 45-1	125V DC distribution cabinets 1 to 4, battery group 2, inverters #1 and #2, Bus 8	0.3
SB 45-2	Battery group 1, MCC 8A, 480 V emergency switchgear	>0.3
SB 45-3	Battery group 1 >0.3	
VE 21-1, 2	Containment spray pumphouse Fans 44A and 44B, filter	>0.3
VE 21-3, 4	SCC line to penetration coolers	>0.3

Table 3-8 Maine Yankee equipment list for margins review.

Equipment Item	System	Building and Elevation	HCLPF (g) Capacity
<u>TANKS</u>			
Refueling Cavity Water Storage Tank TK-4	HPSI	Yd. + 20'	0.21
Primary Component Cooling Surge Tank TK-5	PCC	SB + 61'	>0.3
Primary Water Storage Tank TK-16	AFW	Yd. + 20'	0.27
Demineralized Water Storage Tank TK-21	AFW	Yd. + 20'	0.17
Spray Chemical Addition TK-54	HPSI	Yd. + 20'	>0.3
Secondary Component Cooling Surge Tank TK-59	SCC	SB + 70'	>0.3
Emergency Diesel Day Tank TK-62A, TK-62B	OF	AB + 21'	0.43
Diesel Compressed Air Tanks TK-76A1, TK-76A2, TK-76A3, TK-76B1, TK-76B2, TK-76B3	DG	AB + 21'	>0.3
Diesel Starting Air Receivers TK-76A4, TK-76A5, TK-76A6, TK-76B4, TK-76B5, TK-76B6	DG	AB + 21'	>0.3
<u>PUMPS</u>			
Service Water Pump P-29A, P-29B, P29C, P-29D	SW	CW + 7'	>0.3
Containment Spray Pumps P-61A, P-61B	CS	CS + 14'	>0.3

Table 3-8 Maine Yankee equipment list for margins review. (Cont.)

Equipment Item	System	Building and Elevation	HCLPF (g) Capacity
HEAT EXCHANGERS			
Residual Heat Removal Heat Exchanger E-3A*	PCC	CS + 14'	>0.3
Residual Heat Removal Heat Exchanger E-3B*	SCC	CS + 14'	>0.3
Primary Component Cooling Heat Exchangers E-4A, E-4B	PCC	TB + 21'	>0.3
Secondary Component Cooling Heat Exchangers E-5A, E-5B	SCC	TB + 21'	>0.3
Fuel Pool Heat Exchanger* E-25	PCC	TB + 31'	>0.3
CEDM Coolers* PCC E-53-1, E-53-2, E-53-3, E-53-4	RC + 46'	>0.3	
Reactor Containment* Air Recirculation Coolers E-54-1, E-54-2, E-54-3, E-54-4, E-54-5, E-54-6	PCC	RC + 46'	>0.3
Charging Pump Seal* Leakage Cooler E-92A	SCC	PAB + 11'	>0.3
Charging Pump Seal* Leakage cooler E-92B	PCC	PAB + 11'	>0.3

*Component whose failure may breach critical system pressure boundary.

Table 3-8 Maine Yankee equipment list for margins review. (Cont.)

Equipment Item	System	Building and Elevation	HCLPF (g) Capacity
<u>ELECTRICAL DISTRIBUTION SYSTEMS</u>			
4160-V Emergency Buses Bus 5, 6, 7, 8	Elec	SB + 46'	>0.3
480-V Emergency Motor Control Center MCC-7A, 8A	Elec	SB + 46'	>0.3
480-V Emergency Motor Control Center MCC-7B, 8B	Elec	RMC + 21'	>0.3
480-V Emergency Motor Control Center MCC-7B1, 8B1	Elec	CS + 20'	>0.3
Station Battery No. 1, 2 New Lead Calcium Batteries	Elec	SB + 46'	>0.3
Station Battery No. 3, 4 New Lead Calcium Batteries	Elec	SB + 35'	>0.3
Battery Chargers BC-1, BC-2, BC-3, BC-4	Elec	SB + 46'	>0.3
Inverters INVR-1, INVR-2, INVR-3, INVR-4	Elec	SB + 46'	0.82
Station Service Transformer X-507, X-608 (located adjacent to Bus 7 & 8)	Elec	SB + 46'	0.30
Diesel Generator Control Panel 1A, 1B	Elec	AB + 22'	>0.3
Main Control Board 120-V AC Vital Bus 1-4	Elec	SB + 21'	>0.3
Electrical Control Board DG-1A & 1B Start 1 & 2 Circuits and Control Power	Elec	SB + 21'	>0.3

Table 3-8 Maine Yankee equipment list for margins review. (Cont.)

Equipment Item	System	Building and Elevation	HCLPF (g) Capacity
ELECTRICAL DISTRIBUTION SYSTEMS (Cont.)			
Auxiliary Logic Cabinets	Elec	SB + 21'	>0.3
ESF Auxiliary Panels A & B	Elec	SB + 21'	>0.3
Air Condition Control Panel ACCP	Elec	SB + 21'	>0.3
Safety Parameter Display System Cabinets	Elec	SB + 21'	>0.3
HVAC			
Computer Room Air Conditioner* AC-1A	SCC	SB + 39'	0.38
Computer Room Air Conditioner* AC-1B	PCC	SB + 39'	0.38
Lab Air Conditioner AC-2*	SCC	Unknown	0.37
Containment Spray Fan* FN-44A, FN-44B	SCC	CS + 20'	0.3
VALVES			
Power-Operated Relief Valve PR-S-14, PR-S-15	PORV	RC + 66'	>0.3
Power-Operated Block Valve MOV PR-M-16, PR-M-17	PORV	RC + 64'	>0.3

*Component whose failure may breach critical system pressure boundary.

CHAPTER 4

4. INSIGHTS AND LESSONS LEARNED

During the trial seismic margin review of the Maine Yankee plant, several lessons were learned about the various aspects of performing the review. In addition, many insights have been gained about the entire process of performing seismic margin reviews of nuclear power plants.

Insights and lessons learned have been gained on:

- o Seismic margins approach and methodology including the quantification techniques.
- o The guidance for actually conducting seismic margins reviews including plant walkdowns.
- o The level of effort needed to perform such a review.
- o The identification and resolution of significant issues concerning the operation and licensing of the plant under review.

The purpose of this section is to highlight these insights and lessons learned.

4.1 Approach and Methodology (NUREG/CR-4334)

This section presents a brief discussion of the insights and lessons learned that pertain to the seismic margins approach and methodology. We gained a better understanding of the areas listed below:

1. The seismic margins review methodology should be able to be applied by knowledgeable engineers in the areas of earthquake engineering, component fragility analysis or systems analysis. They need to have some familiarity with these technologies.

   Guidance on performing seismic margin reviews needs to be prescriptive and descriptive enough to guide knowledgeable engineers so that an accurate determination of the plant's seismic margin can be made without allowing over-conservatism into the analysis.

   An independent review body should also be used to verify the application and accuracy of the review.

2. The fragility screening table needs to be strengthened. The actual meaning (representation) of the cut-offs between the table columns needs further explanation. In addition, more guidance in applying the table needs to be given.

3. More guidance is needed on selecting, establishing, and defining the review earthquake level (Step 1 of the review).

There are a number of ways of specifying this review earthquake level:

- o Uniform hazard spectrum at a specified annual probability of exceedance and confidence.
- o Site specific response spectrum evaluated for the specific earthquake magnitude and epicentral distance ranges.
- o Standard spectrum such as the NUREG/CR-0098.
- o Use of peak acceleration or mean acceleration.
- o Peak ground acceleration without specifying any spectrum.

Knowledge of the target spectrum allows the analysts to make a more definitive statement about the seismic capacity of the plant. The review earthquake must be completely specified prior to any plant review, walkdowns or analysis.

4. The seismic margin quantification techniques need to be better defined. There are two competing techniques for calculating the fragility of components, CDFM and FA. For this review, the FA technique was employed. However, the CDFM technique would be more easily applied if more prescriptive and better defined. More guidance is needed to use the CDFM technique. It is difficult to use as presently defined.

   More guidance is needed on the evaluation of plant level HCLPF capacity including the use of nonseismic failures and their significance to the results. More guidance is needed on methods for consideration of correlation between component failures.

5. More generic information is needed with respect to the safety function of reactor subcriticality. This function can be performed by either the insertion of the control rods, or by boron injection. Although the control rod drive mechanisms are screened out in Table 5-1 of NUREG/CR-4334 for review earthquake levels of 0.3g, the reactor internals are not screened out based on insufficient information. This meant that both the reactor internals and the boron injection system components were included in the information gathering process before and during the first walkdown. The boron injection system is fairly complex, with numerous components that are not initially screened out. Appreciable resources were expended in gathering information concerning the boron injection system. This was unnecessary after the seismic capacity review of the reactor internals. It would probably be more efficient in future seismic margin reviews if initial effort is placed in verifying that the reactor internals have high capacity, and only look at alternate means of subcriticality if this is not the case. If necessary, this examination of alternate systems could be accomplished during the second walkdown.

6. Guidance in NUREG/CR-4334 and 4482, based on evaluation of previous PRAs for PWR plants, states that the emergency core cooling (early) function is included in Group A, while the emergency core cooling (late) function is in Group Not-A, and therefore screened out of the analysis. This screening is conditional on not finding any extremely gross plant-specific differences.

   The systems analysis team therefore included the initial switchover phase from emergency core cooling injection (early) to emergency core cooling recirculation (late) as a screening verification step in the first plant walkdown. While the guidelines are ambiguous, this screening verification included long-term area cooling for the recirculation systems. It was determined that the containment spray pump area cooling fans FN-44A and B, and a block wall near the fans, VE-21-1, could not be screened out based on the first walkdown. Based on the plant Boolean equation for small LOCA, both of these items were single failures resulting in core damage in the long term.

   The utility will make changes to these items to increase their capacity so that they do not impact overall plant capacity. Based on these findings, guidance in NUREG/CR-4334 and 4482 should be revised to insure that potential failures such as these are explicitly evaluated during a seismic margins review.

7. For the no-LOCA case, the emergency core cooling (early) function is defined in NUREG/CR-4334 and 4482 as achievement of residual heat removal. The AFW or EFW system at Maine Yankee or other PWR plants will achieve this balance within the first hour. For most PWR plants, irrecoverable failure of the emergency ac power system (station blackout) will not prevent the turbine-driven AFW train from performing early residual heat removal, and therefore satisfy the emergency core cooling (early) function. However, in the longer term without ac power, the station batteries would be depleted, resulting in loss of instrumentation and AFW control power. Core damage could occur if dc power is not restored, and manual control of the turbine-driven AFW train or other feedwater source fails. Based on the guidelines, this long-term failure of AFW was screened out of the analysis. If it were assumed that battery depletion and loss of instrumentation and control power results in loss of AFW and other feedwater sources, then the Boolean expression for the no-LOCA core damage case would be dominated by the seismic failures that result in station blackout:

   - Failure of the SCC and PCC heat exchangers E-5A and 4B.
   - Failure of the station service transformers X-507 and 608.
   - Failure of the DG day tanks TK-62A and B.
   - Rupture of SCC and PCC because of chiller heat exchanger failure for the air conditioners AC-1A, 1B, and 2.
   - Structural failure of the circulating water pumphouse failing the SWS.

Explicit guidance on the treatment of these long-term battery depletion sequences would be helpful.

8. The Expert Panel reports have not explicitly discussed the seismic capacities of steel structures. At Maine Yankee, the steel structures have capacities in excess of 0.30g. At the same time, they could not be screened out based on a walkdown review because of unusual connection details and structural arrangement. More guidance is needed on the treatment of steel structures.

9. The availability of in-structure response spectra generated by current techniques can reduce the amount of effort necessary to quantify component fragilities since modification or regeneration of responses was not necessary. Accurate floor spectra may not always be available for older plants. In such cases, it may be necessary to develop new dynamic models and perform dynamic analysis to define the seismic input to equipment. This will increase the amount of time and funding necessary to perform seismic margin studies.

10. Detailed information on the Maine Yankee block walls was available. This data was very useful in conducting the trial plant review since it provided locations of all block walls in the plant and identified any safety-related components that could be affected by their failure. In particular, this latter set of information was valuable since it permitted quick identification of several Group A block walls.

11. From our review of Maine Yankee, we identified an additional plant unique feature. This unique feature is cast iron service water piping. This cast iron piping at Maine Yankee was adequately supported and judged to have an acceptable capacity.

4.2 Guidelines (NUREG/CR-4482)

This section provides a brief discussion of the lessons learned and insights gained concerning the guidelines for conducting seismic margins reviews.

1. Insights were gained about the planning and conduct of the physical plant walkdowns from our review of Maine Yankee.

   The most essential part of a plant walkdown is preparation and planning. All walkdown participants should help organize and plan the specifics of the plant visit which are dependent on the nature of the walkdown and the specific plant under review. The objectives of the walkdown should be clearly understood and arrangements for security, radiological safety, training, and plant operation should be considered.

   Walkdown groups consisting of three to five people with background and expertise in various engineering disciplines should utilize pre-developed inspection criteria and forms to collect information and data pertaining to the objectives of the plant review. The groups should be organized with respect to the information and data that needs to be collected. Each group should contain a utility person familiar with the plant layout, and the location and operation of the systems and components under review.

A clear understanding of what is going to be reviewed and inspected is also essential to the walkdown. A list of the systems, components and plant areas that need to be visited during the actual walkdown should be developed. An itinerary will also facilitate the actual plant walkdown.

During the plant visit, meetings should be arranged each day to organize the groups and prepare for the walkdowns. These meetings should allow time for discussion of findings at the end of each day and the preparation for the next day's effort.

Walkdowns can be performed in many ways, from a quick "walkby" of the plant to a thorough "crawl through," depending on the needs of each group, and the information and data collection requirements. The types of walkdowns and their progressions should be planned in advance. Tools for the collection and recording of the information and data will also help facilitate the walkdown.

2. The identification of the front-line system components was relatively straightforward because of the detailed nature of the available information, and the small number of components. However, identification of support system components was more difficult. This is because these systems are generally more complex, have many more components and branches, and are not generally documented as well. In addition, the references concerning interfaces between the front-line systems and support systems are often ambiguous. Finally, the actual physical nature and location of some items, such as distribution cabinets and panels, is not shown on plant drawings or documentation. Based on this experience, there are two recommendations. First, when reviewing the plant information, emphasize the interfaces with support systems such as ac and dc power, cooling water systems, HVAC systems, and instrument air systems. Document the ambiguities for later clarification. Second, plan to spend considerable effort tracing down these support system components during the first walkdown, and be prepared to make substantial revisions to the component list.

3. Another insight concerns documentation and information transfer between the Systems Analysis Team and the Fragilities Team. Many of the components identified by the Systems Analysis Team for HCLPF screening or evaluation were selected because of the potential for component rupture to cause flow diversion and consequent system failure. The component itself was not needed to fulfill a safety function, but the integrity of the component pressure boundary had to be assured for overall system success. The common example was heat exchangers for nonessential equipment whose seismic rupture would fail a necessary cooling water system. Since it can make a

difference to the HCLPF assessment, the systems team must make a clear differentiation between components that are required to function for system success, and components that are required only to maintain pressure boundary integrity.

4. There are a number of insights concerning the systems analysis and the fault tree pruning process which may be helpful to future seismic margin reviews.

   a. The procedure proposed to isolate the PCC lines and components inside containment, and the automatic rupture isolation system on the SCC greatly reduced the amount of components that had to be considered for HCLPF evaluation. Both the systems analysis and fragilities analysis efforts would have been larger, and eventually a containment walkdown might have been necessary.

   b. Early evaluation and screening out of potential recovery actions and alternate systems, such as the small positive displacement pump for core cooling injection, reduced the number of components that required systems and fragility evaluations.

   c. Being able to define all the components on one skid as one supercomponent, such as the DGs, reduces the systems analysis effort, but the evaluation for the fragility team may not be reduced.

   d. As the fault trees are developed, it is useful to keep a list of the failure modes which should be considered for each component. For example, the pump failure modes include fail to start, fail to run, and test or maintenance outage, as well as seismic failure, but the pump appears only once on the initial fault tree. This information is needed later for the quantification process.

   e. In the initial trees, it is necessary to include all the components that require support systems, including those components such as motor-operated or air-operated valves that will likely be screened out later because of their high HCLPF and low nonseismic unavailability. Otherwise, if they are pruned from the initial trees, their dependency on support systems may be overlooked in the rest of the analysis. Also, since physical interactions between the component and structures, such as block walls or restraints, must be checked, it is better to include the component in the initial fault trees.

   f. In this project, those components and structures that were assigned a HCLPF of ">0.3g" either because they fit the generic screening guidelines, or because a conservative fragility calculation was performed, were pruned from the fault trees before the Boolean equations were developed. The analysis team felt that their actual HCLPF would be well in excess of 0.3g, and that they would not affect the plant seismic capacity calculation. This pruning process resulted in very manageable fault trees, relatively small Boolean equations, and satisfactory calculations for the plant HCLPF. It also enabled easier communication, understanding, and insight into the results than if less pruning had been performed, but provided more information on seismic margin than a more severely pruned Boolean equation. The balance appears to be satisfactory.

g. Although a few seismic interactions, such as the possible impacts of potential seismically induced fires, were not evaluated, the potential for threaded firewater piping ruptures to damage equipment was reviewed. The DG control panels and distribution panels were located under firewater piping that could have considerable lateral movement. Upon investigation, however, it was determined that the piping was dry, and two signals would be required to fill the piping. The probability of inadvertent actuation was therefore negligible. The PCC and SCC pumps were also located under sprinkler nozzles, but the motor housings for these pumps were designed to prevent water from entering. Therefore, ruptures of firewater piping was not considered to impact seismic capacity.

5. Minimal cut sets were developed at four stages in this project. Front-line system level cut sets using partially pruned fault trees, including their support systems, were developed just before the second walkdown to provide some guidance to the fragility team. These pointed out some potentially important system minimal cut sets. Although plant or sequence level cut sets could have been of additional assistance, because the fault trees were still fairly large, the number of minimal cut sets would have been large as well. The additional effort to develop plant level cut sets before the second walkdown is not judged to be an effective allocation of resources, but the effort to develop system cut sets is effective.

6. The seismic systems models of the plant (event trees and fault trees) should be developed using best-estimate success criteria based on the FSAR and any relevant experience data. The FSAR analyses, in a sense, represents "high confidence" because of the regulations upon which they are based. A best-estimate realistic success criteria beyond the FSAR should be used if there are data and/or calculations to support their use.

7. Due to radioactivity concerns, critical components inside the Maine Yankee containment are accessible for review only at the time of a plant outage. Fortunately, Maine Yankee has maintained a relatively complete data file on components within the containment. Difficulties in reviewing plant components will be more severe for plants which do not maintain organized documentation on the components within containment.

More guidance is needed on the consideration of the seismic small LOCA-initiating event. It may be impossible to review and inspect the reactor coolant piping (primary pressure boundary) throughout the plant, in particular within the containment, in order to screen out this possible accident initiator. For the Maine Yankee review, the small LOCA-accident sequence was explicitly considered in the systems analysis and was the controlling event sequence with regard to plant capacity.

8. During the walkdowns, plant and utility personnel having expertise or specialized knowledge in particular fields were interviewed. Specific areas of expertise included:

   - Structural/mechanical.
   - Firewater systems.
   - Electrical.
   - HVAC.
   - Instrumentation and control.
   - Control room personnel.
   - Block walls.

   The insight gained from discussions with knowledgeable individuals proved to be very useful in assessing the overall state of the plant and resolving any particular seismic issues.

9. Walkdown data sheets, while not noted in the review guidelines, were developed prior to the first walkdown. These data sheets served as a checklist of items to review during the walkdown and facilitated the gathering of information necessary for HCLPF evaluation. While no set format need be established for future margin reviews, the review guidelines should indicate that the use of data sheets is encouraged.

10. Major distribution systems such as cable trays, piping, and ducting extend throughout the plant, there local configurations and support details can vary. Detailed walkdowns of these systems can be time-consuming. More guidance in the seismic margins methodology on the level of walkdown for these systems would be useful.

11. In the course of this study, five example components (i.e., refueling water storage tank, steel structure, diesel day tank, inverter and block wall) were analyzed using the two candidate methods (i.e., CDFM and FA). Our experience was that several judgmental decisions had to be made in arriving at the parameters of the CDFM method. In each case we were not sure whether we met the intent of the method, i.e., conservative estimation of the capacity, yet more liberal than the SRP requirements; in some cases, we may have been overly conservative as was pointed out by the Peer Review Group. The difficulties arise because of two factors:

    - The CDFM method has not been fully defined for all structures and equipment items.

    - The parameters of the CDFM method such as damping, material strength, static capacity equations, system ductility, and methods for floor spectra generation are not explicitly specified; even where they are specified they may be overly conservative. Also, the appropriate conservatism in the selection of the CDFM parameters needs to be determined using calibration methods.

We recommend that such a comparison study be performed.

12. This plant review examined a PWR on a rock site with a review earthquake level of 0.30g. The methodology, the review guidelines, and the staffing requirements have not been verified for other conditions including BWRs, soil conditions, and a higher review earthquake.

4.3 Effort to Perform the Review

An objective of the trial seismic margin review is to gain an understanding of the level of effort, and thus cost, of performing seismic margins reviews of nuclear power plants. The trial review involved effort by these personnel: the Systems Analyst Team, the Fragility Analyst Team, the utility or plant operator, and the management of the overall review.

NUREG/CR-4482 gives estimated staffing requirements for a seismic margin review; for a plant founded on rock with a review earthquake level of 0.30g pga, the estimate to perform the review is 2.5 staff-years. The actual effort expended in the present study is about 25% more than the Panel's estimate since this was the first review conducted.

Both analyst teams have divided their efforts into specific tasks. The tasks and the level of effort expended by both teams is given in Table 4-1. The Systems Analyst Team spent 1532 man-hours performing the review, and the Fragility Analyst Team spent 4176 man-hours. The combined effort was approximately 35 man-months.

A breakdown of the utility effort is given in Appendix C. They spent 1060 man-hours for providing data, answering question, organizing walkdowns, and reviewing results.

4.4 Findings and Their Resolution

A number of issues became apparent during the Maine Yankee review. Primarily, these were findings concerning the seismic capacity of various components at the plant. While many of these issues were discussed in the result section (Section 3.0), this section is intended to briefly discuss all the seismic margins related findings and their resolution concerning the Maine Yankee review. A listing of each issue and brief discussion of the resolution are given below:

1. Several types of analysis techniques can be employed to calculate the seismic capacity of free-standing tanks. For the Maine Yankee RWST these techniques resulted in the HCLPF capacity being greater than 0.20g. The lowest of these values is shown in Table 3-2.

The Maine Yankee DWST, which is not a singleton to the seismic safety of the plant, was found to have a HCLPF capacity of 0.17g.

Table 4-1 Cost breakdown items for seismic margins trial plant review.

Review Items		Actual Fragilities Team Staff-Hr	Actual Systems Team Staff-Hr
1.0	Collect Information on Design		
1.1	First Round Information	120	20
1.2	Additional Specific Information	80	34
1.3	Visit to Utility/AE/NSSS Vendor	96	8
2.0	Review of Plant Information		
2.1	Review of Review Earthquake Level	40	--
2.2	Initial Systems Review	--	60
2.3	Identify Components for Group A Functions	--	36
2.4	Perform Initial Screening of Components	176	50
2.5	Review of Design-Analysis and Seismic Reevaluation Reports	216	--
3.0	Plant Walkdowns		
3.1	Identify Target Areas for First Walkdown	240	50
3.2	Perform First Walkdown	184	84
3.3	Conduct Simplified Analysis	200	36
3.4	First Walkdown Documentation	320	34
3.5	Perform Second Walkdown	200	26
4.0	Systems Modeling		
4.1	Develop Event and Fault Trees	--	539
4.2	Derive Accident Sequences	--	70
4.3	Develop Boolean Expressions and Minimal Cut Sets	--	50

Table 4-1 Cost breakdown items for seismic margins trial plant review (Cont.).

Review Items	Actual Fragilities Team Staff-Hr	Actual Systems Team Staff-Hr
5.0 Seismic Margin Evaluation		
5.1 HCLPF Capacity of Components		
5.1.1 CDFM Method	280	--
5.1.2 Fragility Analysis Method	800	--
5.2 HCLPF Capacity of Plant	160	10
6.0 Reporting		
6.1 Internal Review	180	22
6.2 Letter and Final Reports	480	253
7.0 Meetings		
7.1 Project Teams	224	104
7.2 Peer Review Group	80	22
7.3 NRC/ACRS/Expert Panel	100	24
	4176	1532

2. There is no data or experience on the seismic behavior of aged lead-antimony station batteries at the Maine Yankee plant. Therefore, there was no way to estimate their seismic capacity and a HCLPF value.

   Maine Yankee will have these station batteries replaced with lead-calcium batteries during the next two refueling outages. Station Batteries 1 and 3 will be replaced during the next refueling outage (March 1987) and Station Batteries 2 and 4 will be replaced during the 1988 outage. The HCLPF capacity of the station batteries was evaluated for the replacement units.

3. Older General Electric station service transformers (4160 V to 480 V) were not tested or qualified for seismic loads and their seismic capacity is low. This component became the most dominant contributor to the seismic capacity of the plant prior to the upgrade.

   GE had recent seismic testing and qualification performed on its equipment transformer. This testing resulted in modifications. Base anchorage modifications are scheduled for the Maine Yankee transformers.

4. Anchorages on the containment spray pump area fans FN-44A and B will be strengthened. Failure of these fans could have led to long term heat-up and failure of the containment spray pumps, with subsequent failure of high pressure safety recirculation if recovery actions were not effective.

5. Block wall VE 21-1 will be strengthened to prevent its potential collapse from failing the containment spray area fans FN-44A and B discussed above.

6. The anchorages of the chillers for the computer room air conditioners AC-1A and B and the laboratory air conditioner AC-2 will be strengthened. Failure of the heat exchangers on these chillers could have failed the pressure boundary integrity of the SCC and PCC, and resulted in core damage upon loss of component cooling water.

7. A procedure is being developed to isolate nonessential PCC lines and heat exchangers following a large earthquake and receipt of a low level indication in the PCC surge tank. Although the PCC system was designed to seismic standards, the project team could not verify the capacity of all components. Isolating the PCC lines provides assurance that any potential small leakage of the pressure boundary will not fail the entire PCC system.

8. An unanchored monitor in the main control room panel was anchored. Its impact on other components and the seismic capacity of the plant therefore did not have to be evaluated.

9. The emergency lights in the control room and throughout the plant were strapped and anchored.

10. A missing bolt on the anchorage of some level transmitters for the RWST was replaced.

11. Loose pressurized gas cylinders, a welding machine, and some heavy parts near the containment spray pump area fans were moved or tied securely.

12. Additional anchorage was added to both diesel generator day tanks.

CHAPTER 5

5. CONCLUSIONS

The trial seismic margins review of the Maine Yankee plant was conducted with the concerted effort of all parties involved. The analysis teams worked together closely and followed the guidance on performing the review to estimate the overall plant HCLPF and the HCLPF capacities for the accident sequences that lead to seismic-induced core damage. The Maine Yankee utility and Yankee Atomic Electric Company provided invaluable assistance in performing the review. Without their efforts, this review would have been much more difficult. The Expert Panel provided an initial review of the approach early in the project. The Peer Review Group provided guidance and a critical examination of the process and interim results at each stage of the review. The NRC assisted in the definition of the scope of the review and licensing issues involved.

The conclusions for this trial review are divided into two sections: conclusions concerning the application of the seismic margins methodology, and guidelines including the numerical results and insights gained from this study are given in Section 5.1. The conclusions regarding Maine Yankee's licensing issue are given in Section 5.2.

This seismic margin review has been performed with the following assumptions and limitations:

- The review earthquake level was specified by the NRC as the NUREG/CR-0098 median spectrum anchored to 0.3g.

- The structural models and the in-structure response spectra generated by Maine Yankee have been judged to be adequate for the purposes of this margin review.

- Since the Analysis Team could not perform the walkdown inside the containment, the seismic capacity of components inside the containment could not be determined. We could not confirm the absence of potential system interaction effects that may make the impulse lines inside the containment vulnerable to earthquakes and lead to a small LOCA.

- In keeping with the Expert Panel's philosophy, the screening of components was performed using conservative procedures. For the screened-in components, the seismic capacities have been calculated using conservative methods. In all cases, the factors contributing to the seismic margin and their variabilities are identified and quantified using procedures normally used within the state-of-the-art.

- The HCLPF capacity of the plant has been determined based on the seismic capacities of components in their existing or proposed modified conditions. Maine Yankee has proposed that certain modifications or replacements would be made for station batteries, transformer internal core/coil assembly anchorage, vibration-isolation supports for containment spray fans and air conditioners, anchorage of diesel day tank, and block wall near the containment spray fans.

- o The results of this seismic margins review represents the best estimate analysis of the components and the plant following the proposed modification as a result of this review. No effort was made to account for the effects of future aging.

5.1 Conclusions Regarding the Methodology, Guidelines, and Insights

The conclusions from the trial seismic margin review include:

- o The plant HCLPF capacity was determined to be 0.21g. This capacity is dominated by the small LOCA-accident sequence with the RWST being the dominant component. There was no effect on the plant HCLPF capacity when we considered the dependence between component failures.

  An arbitrary 10% reduction in RWST fluid level results in an increase in HCLPF capacity to 0.26g.

  Assuming an arbritrary ten percent (10%) probability of occurrence of the small LOCA-initiating event as compared to the occurrence of all the other possible initiating events in the plant, HCLPF capacity increases to 0.28g.

- o We found that careful plant walkdowns are essential to successful seismic margin reviews.

- o Insights and lessons learned (discussed in Chapter 4) concerning:

  - The selection of the review earthquake level.
  - The methodology and use of the review guidelines.
  - Component qualification data.
  - Plant walkdown procedures.
  - Guidance on the CDFM HCLPF calculation procedure.

- o The maintenance of hot shutdown following a seismic-induced initiator was considered by performing a thorough walkdown review and analysis of the components needed to perform this function. This led to upgrading of Fans FN-44A, B and the adjacent block wall.

- o Important components and failure modes discovered during this review are:

- Station service transformers (4160 V to 480 V) require a review of their internal configuration.

- Lead-antimony station batteries may fail due to the failure of the plates within the battery casing. No data are available to estimate a HCLPF capacity for lead-antimony batteries.

- Consideration must be given to the location and possible systems interaction from threaded fire water piping.

- Consideration must be given to modifications and upgrades identified during the seismic margins review process.

- The small LOCA-initiating event had to be considered for this review because of the difficulty in performing a walkdown inside the containment building.

- The walkdown review of closed-loop component-cooling systems may require considerable effort.

- The components that affect the reactor subcriticality function, in particular the control element drive mechanisms and the reactor internals, need to be considered early in the review.

5.2 Conclusion Regarding the Maine Yankee Licensing Issue

The Maine Yankee plant was found to be clean and well-maintained. There has been a concerted effort to seismically upgrade the plant consistent with newer plants. The Maine Yankee utility and its contractor, Yankee Atomic Electric Company, were very cooperative with this effort and responded on their own initiative to increase the seismic margin of the plant.

A number of components were identified as having a low seismic capacity. A letter indicating that Maine Yankee will upgrade these components by the end of the next two refueling outages in March 1987 and 1988 is included in Appendix C. These components are listed below:

- Important station service transformers (4160 V to 480 V)

- A block wall near the HVAC equipment (Fan 44A & B) needed to cool the containment spray pump enclosure. This enclosure houses the long-term cooling equipment.

- The PCC and SCC heat exchangers for both the computer room air conditioning compressors (chillers).

- Station batteries 1 and 3 will be replaced during the March 1987 outage. Batteries 2 and 4 will be replaced in the 1988 outage.

- Upgrading the anchorage of both diesel generator day tanks.

The plant HCLPF capacity after the planned upgrades was estimated to be 0.21g. This HCLPF capacity is governed by the RWST and represents a conservative estimate of the seismic capacity of the plant. That is, given an earthquake producing this ground acceleration and specified spectral shape, there is high confidence (95%) that there is a low probability of core damage (occurring only approximately 5% of the time).

It is recommended that an inspection and review of Maine Yankee be performed after the March 1987 outage to assure that the proposed modifications have been carried out and that their HCLPF capacity calculated in the present study are still applicable.

CHAPTER 6

6. RECOMMENDATIONS TO IMPROVE METHODOLOGY

During the development of the seismic margins review methodology and guidelines, and as a result of performing this first trial seismic margins review, a number of recommendations can be made for further analysis and research that will improve the applicability and usability of this method for future reviews. These recommendations are listed below:

- Need to revise and clarify the seismic margins methodology and guidelines based on lessons learned so they are more usable.
- Need to make a comparison of this methodology and its application with a similar effort being performed by EPRI.
- Need a comparison study that addresses the CDFM and FA methods for calculating HCLPF capacities.
- Need a study of the available methodologies for calculating the capacity of older tanks.
- Need a trial seismic margins review of a BWR plant to test enhancement of the methodology presently underway that will address this type of plant.
- Need testing of aged batteries to understand their seismic capacity. Additionally, to understand the internal plate failure mode for batteries in general.

REFERENCES

American Society of Civil Engineers (ASCE), "The Effects of Earthquakes on Power and Industrial Facilities and Implications for Nuclear Power Plant Design," Draft UCRL-93320 (1986). To be published by the ASCE.

Memorandum from Newton Anderson, NRR Co-Chairman Seismic Margins Working Group, to Mr. James Richardson, RES Co-Chairman Seismic Margins Working Group, "Comments on the Seismic Margins Program and the Trial Plant Review," July 7, 1986.

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

Campbell, R. D., M. K. Ravindra, A. Bhatia, and R. C. Murray, "Compilation of Fragility Information from Available Probabilistic Assessments," Lawrence Livermore National Laboratory, Livermore, CA, UCID-20571 (September 1985).

Carlson, D. D., "Interim Reliability Evaluation Program Procedure Guide," NUREG/CR-2728, SAND82-1100 (January 1983).

Memorandum from Dennis M. Crutchfield, Assistant Director for Technical Support, Division of PWR Licensing-B, to Patrick M. Sears, Project Manager, Project Directorate No. 8, Division of PWR Licensing-B, "Recommended Spectrum for Maine Yankee Seismic Design Program," May 7, 1986.

Cummings, G. E., J. J. Johnson, and R. J. Budnitz, "NRC Seismic Design Margins Program Plan," UCID-20247 (October 1984).

Memorandum from: D. Guzy, MSEB, DET, to NRC Seismic Design Margins Working Group, "Maine Yankee Seismic Margins Review," May 12, 1986.

Kennedy, R. P. and M. K. Ravindra, "Seismic Fragilities for Nuclear Power Plant Risk Studies," Nuclear Engineering and Design, Vol. 79, No. 1, pp. 47-68 (May 1984).

Maine Yankee Atomic Power Company, Final Safety Analysis Report (FSAR), "Section 2.5, Seismology."

Letter from Frank J. Miraglia, Director, PWR Project Directorate No. 8, Division of PWR Licensing-B, to Mr. J. B. Randazza, Executive Vice President, Maine Yankee Atomic Power Company, "Maine Yankee Participation in Seismic Design Margins Program," March 31, 1986.

Newmark, N. M. and W. J. Hall, "Development of Criteria for Review of Selected Nuclear Power Plants," NUREG/CR-0098 (May 1978).

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

Oral Communication, Richardson, J. E., Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, on the clarification of the seismic margin review earthquake level for the Maine Yankee trial review (December 11, 1986).

U.S. Nuclear Regulatory Commission, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Regulatory Guide 1.60 (December 1973).

U.S. Nuclear Regulatory Commission, "Fault Tree Handbook," NUREG-0492 (January 1981).

U.S. Nuclear Regulatory Commission, "PRA Procedures Guide," NUREG-2300 (January 1983).

APPENDIX A

EXPERT PANEL CORRESPONDENCE

INTERFACE BETWEEN THE EXPERT PANEL ON SEISMIC MARGINS AND THE TRIAL MARGIN REVIEW OF MAINE YANKEE

R. J. Budnitz, Chairman

The NRC's Expert Panel on Seismic Margins developed the original approach to performing seismic margin reviews (NUREG/CR-4334) and oversaw the development of interim guidance on how such a review should be carried out (NUREG/CR-4482). It is important that the Expert Panel have confidence that the trial margin review is accomplished in a technically sound manner; this confidence will be assured by the Peer Review Group that is overseeing the trial margin study at Maine Yankee.

The Expert Panel's role during the trial review will be very limited. Specifically, it is expected that the Expert Panel will review the approach being taken at an early stage of the Maine Yankee trial review, to assure itself that the methods and techniques being employed are consistent with the Panel's guidance and are relevant for performing a seismic margin review.

This will involve a review of procedures prepared for the Maine Yankee review, discussion of these procedures on a conference call set up by the Chairman of the Expert Panel, and a follow-up letter stating the Panel members' opinions concerning the Maine Yankee review.

After the early interaction, the Expert Panel will not be involved again until the trial review has been completed. At that time, the Panel will be convened so that it can study the results, examine how the trial review was implemented, interview the study team, and then evaluate the overall effort. The Expert Panel will be expected to re-examine its interim guidance and to revise it, if necessary, prior to issuance in final form. The Expert Panel will also be expected to evaluate the overall usefulness of the seismic margins review approach, any limitations whether previously recognized or not, and provide any other relevant comments to LLNL and the NRC.

Future Resources Associates, Inc.

2000 Center Street Suite 418 Berkeley, CA 94704 415-526-5111

11 July 1986

TO: Expert Panel on Seismic Design Margins (P. Amico, C.A. Cornell, J.Reed, M. Shinozuka)

R.C. Murray, Lawrence Livermore National Laboratory

FROM: R.J. Budnitz, Chairman of the Panel

REF: Minutes of the Telephone Conference Call Meeting of June 27, 1986

This memorandum consists of the minutes of meeting of June 27, 1986 of the NRC's "Expert Panel on Seismic Design Margins", whose membership consists of the individuals listed above. The 'meeting' was actually a conference telephone call arranged by Bob Murray of LLNL, who is the NRC/LLNL liaison for the Panel. In 'attendance' were all five members of the Panel; Murray and P.G. Prassinos of LLNL; David L. Moore of Energy, Incorporated; and M.K.Ravindra, P. Hashimoto, G. Hardy, and M. Griffin of EQE, Incorporated. The conference call began at 10:00 AM and ended at 12:40 PM, Pacific time.

The objective of the meeting was to provide a forum for the Expert Panel to discuss progress to date on the trial seismic margin review that is now underway under NRC/LLNL sponsorship using the Maine Yankee plant as the trial plant. The review has been underway for 1½ months, and this was an ideal time to obtain Panel comments and input as to the progress being made. As it turned out, there were several technical issues that had arisen in the course of the start-up phase of the trial review which benefitted from input from the Panel.

The 'bottom line' outcome of the meeting was that the Panel was satisfied with the way the trial review was being undertaken so far, with the exception of a few (important) comments that will be covered in the remainder of these minutes. It is NRC's intention that the Expert Panel be involved next at the stage when the trial review has been completed, at which time the Panel will study the results, interact with the team of analysts, and then reevaluate the guidance that has been provided in interim form in the Panel's NUREG publications on this subject, NUREG/CR-4334 and NUREG/CR-4482. (During the course of the conference call, the Panel estimated that about two person-weeks of effort would probably be required of each Panelist at this later time to carry out the review effectively, and this estimate does not count effort that may be needed to revise the interim guidelines.)

1) The meeting began with Bob Murray's discussion of the progress to date of the analysis team, whose key participants were all on the telephone line: Pete Prassinos and Dave Moore representing the systems analysis team, and Bob Murray, Ravi Ravindra, and Ravindra's EQE colleagues representing the fragilities team. Murray discussed the sechedule for the review, whose major

upcoming milestone will be the first plant walkdown on July 21-26, at the Maine Yankee plant. He discussed the interactions with the plant team, which interactions have been entirely favorable and cooperative so far; and the arrangements made for a 'peer review group' to follow the progress of the trial review. Murray's introductory discussion set the stage for the rest of the conference call discussion.

2) The Panel discussed the fact that the trial margins review at Maine Yankee is intimately tied up with an NRC licensing action. The Panel expressed regrets that this linkage was necessary, since the Panel believes that this trial review should be viewed in an important way as a research project whose outcome is by no means well understood until it has been accomplished. The Panel expressed a strong desire that NRC find a plant (expected to be a BWR) for the planned second trial review which does not have a licensing action tied to the seismic margin review.

3) The Panel discussed the peer review arrangement for the trial review, and agreed that the set-up was consistent with the guidance provided in its earlier NUREG reports. The peer review group's charter is to assure the technical competence of the review, and to report on its findings to NRC and Maine Yankee.

4) The Panel discussed at length the selection of the 'review level earthquake' (henceforth abbreviated RLEQ) by NRC, which selection is at the 0.30 g level, with a spectrum using the 50th percentile amplification factors from NUREG/CR-0098. A question arose as to how to cope with motions at higher frequencies than those in the more-or-less standard spectrum chosen. The Expert Panel reaffirmed its earlier position that a separate research project is needed to provide information to the Panel before the Panel can develop margin-review guidance on this issue, and that higher-frequency motions are explicitly not included in the guidance already developed in the earlier NUREG reports.

5) Regarding the choice of the RLEQ, the Panel agreed that especially for a trial review like the one being undertaken at Maine Yankee, it would be preferable if the RLEQ were selected to be at a high enough level that the 'plant HCLPF value' can be affirmatively determined through the review, rather than having the review determine only that the plant HCLPF value is "at least as high as the RLEQ level." The choice of 0.30 g for Maine Yankee's RLEQ does not obviously meet that criterion, although it may (after the fact) turn out that way. The Panel agreed to reconsider at a later date whether it is necessary to provide more detailed guidance on 'how to select the RLEQ'.

6) The Expert Panel, after much discussion, agreed that it would be unfortunate if the fact that a given plant's HCLPF turned out to be less than the RLEQ were considered a 'negative' outcome for the review. Considering the conservatisms embedded in the HCLPF idea, a plant's median capacity would be expected to be considerably greater than its plant-level HCLPF value.

7) The Panel discussed the meaning of its categorization in NUREG/CR-4334 and NUREG/CR-4482 of fragilities into three groupings, "below 0.30 g", "0.30 to 0.50g", and "above 0.50g". In its earlier reports, the Panel pointed out that these boundaries were not to be taken as being too precise: specifically, the Panel stated that the 0.30g boundary might just as easily have been a rough range from about 0.25g to about 0.35g. While this Panel judgment still stands,

the calculated HCLPF values are not to be interpreted in the same way....that is, if a plant-level HCLPF value is determined by the analysis to be, say, 0.30g the Panel believes that it is not correct to believe that the HCLPF value might just as easily have been anywhere in the range of, say, about 0.25g to about 0.35g. The conservatisms embedded in the "HCLPF" concept, through the concept of 'high confidence' and the concept of 'low probability of failure', should allow the regulatory decision-maker to cope with any technical decision without the additional 'smearing' of placing an analyzed HCLPF value in a broad range like that just mentioned. The Panel was explicit and strong about its insistence on this point.

8) The Expert Panel provided guidance to the margins review team that, even though categories or specific items of equipment are thought to possess quite high capacities compared to the RLEQ, it is still necessary to do at least some type of review of these items to confirm that they are not 'outliers'. The criteria in the Panel's chapter 5 discussion (NUREG-CR-4334) are considered by the Panel to be 'generally conservative' but not 'absolutely conservative in every case.' Specifically, every screening decision must be confirmed somehow.

9) The Panel expressed its strong disappointment that there has been no start to date on the separate research project that was recommended to compare the CDFM (conservative deterministic failure margin) method and the FA (fragility analysis) method for calculating capacity values. The Panel was told that in the Maine Yankee review Ravindra's fragility team will do such a comparison on about a half-dozen items. This outcome, in which all of the capacities are to be determined using the FA method and the CDFM approach used only as a 'trial' or 'check', is certainly not what the Panel had envisioned when the guidelines were being developed. The trial review approach is <u>not a substitute</u>, in the Panel's opinion, for a separate study to confirm the adequacy of the CDFM method. The Panel expressed a strong need to go on record that <u>the CDFM method needs to be confirmed</u>. Although the Panel believes that the CDFM approach is ultimately the preferable method, it also believes that CDFM should not be used and relied on until confirmed. The Panel discussed possible dangers of two kinds: first, the CDFM method might be used prior to its being studied and confirmed; and/or second, it might never be used because it is not a confirmed approach, thereby leaving margins analysts with no choice but to use the much more expensive and less desirable FA method.

10) The Panel was told that it is very unlikely that Maine Yankee's reactor <u>internals</u> and control rod drives can be reviewed to obtain useful fragility information, because they are inaccessible and the vendor design/test information is unavailable. The Panel's guidance was that in such a case it is preferable to do a detailed study of the back-up reactivity control sysems such as the borated water tanks and plumbing, rather than to leave the issue open, even though there is high confidence on a generic basis that the capacities of the reactor internals and CRDs are above the RLEQ selected for Maine Yankee.

11) There was extensive discussion on the safe-shutdown end-point used by the Panel. The Panel reiterated that it used the traditional PRA approach: for transients a stable state is typically achieving hot shutdown plus holding it for 24 hours (sometimes 36 or 48), while for LOCAs a stable state is usually cold shutdown. The Panel did not explicitly use 'hot shutdown plus 72 hours'. The Panel reiterates its finding, based on the extensive PRA literature <u>for PWRs</u>, that it is very unlikely in a probabilistic sense that the systems being screened in the seismic margin method proposed would have a high HCLPF value while the systems needed for post-hot-shutdown heat removal would have a quite low HCLPF

value. Therefore, based on its study of the PRA literature the Panel has concluded that it is not necessary to do a detailed margin review that includes the systems supporting the function of post-hot-shutdown heat removal.

12) Notwithstanding the above, the Panel reaffirms its earlier guidance that it is necessary to do a modest review of these and other systems to assure that there are not 'plant unique features' in the sense of the discussion in NUREG/CR-4334 and 4482.

13) The Panel also reaffirms its earlier approach to non-seismic-induced failures or unavailabilities, which approach is that these are to be combined with seismic-induced failures using an approximate probability-based cut-off criterion that relies on expert judgment (see NUREG/CR-4334). There is no precise numerical guidance on how to apply this probabilistic cut-off.

14) Regarding seismic-initiated failures, the Panel was told that it would not be possible to enter the Maine Yankee containment during the upcoming review. Therefore it will not be possible to rule out small LOCAs inside containment caused by earthquakes like the RLEQ. The Panel agreed with Dave Moore (leader of the systems analysis subcontractor team) that an appropriate approach is to do the analysis two ways ---- one way assuming that a small LOCA inside containment does occur, and the other way assuming that it does not occur. Mitigating systems (injection pumps, etc.) will be studied to gain engineering insights as to the plant's possible vulnerabilities, if any.

15) The Panel agreed that, at the RLEQ selected, the traditional large LOCA pipe breaks were not a problem, on a generic basis. The Panel was told that Maine Yankee has large loop valves in the primary piping which were placed there in the original design to isolate one of the main loops. These might be more vulnerable to the RLEQ than the primary piping itself. The Panel's guidance was to treat these as 'plant unique features' and to study them through design information if that was the only available approach.

16) Panel discussed at length the difference between the Panel's and EPRI's approaches to the systems analysis. In a nutshell, EPRI's approach is to search for one 'success path' at the RLEQ, while the Panel's approach is to search for cut sets representing combinations of failures. The Panel reaffirmed its earlier conclusion that the cut-set approach is preferable since it provides more extensive engineering insights, but agreed to revisit this issue after both the EPRI trial review of Catawba and NRC's trial review of Maine Yankee are completed.

17) Bob Murray reported to the Panel about the excellent cooperation being received from the EPRI margin study effort, which is doing a parallel review of Catawba. The Panel wishes to encourage the maximum cooperation, since both groups have already learned from each other and this will likely continue to be true.

18) The Panel was informed that the BWR systems study is now underway, which should enable the Panel to develop BWR guidance sometime next fiscal year. The Panel was urged by Bob Murray to give thought now to possible nominees of a BWR candidate for the second trial margin review.

In conclusion, the Panel's main finding based on the conference call meeting of June 27 is that the trial margin review at Maine Yankee is proceeding appropriately, and that the Panel's overall guidance is being followed, with the exception of our comment on the CDFM method (see item 9 above). The Panel is very pleases otherwise with the progress so far.

Robert Budnitz

Robert J. Budnitz
Chairman, Expert Panel

Applied Risk Technology Corporation

P.O. Box 175, Columbia, MD 21045 (301) 992-0525

Date: 28 July 1986

To: R. C. Murray, L-197, Lawrence Livermore National Laboratory

From: P. J. Amico, Applied Risk Technology Corporation

Subj: P.O. 9225705, Comments on Maine Yankee Trial Plant Review for Seismic Design Margins Program

This memo constitutes the final deliverable for the subject purchase order, and is assumed to be sufficient documentation for the tasks performed. As you are aware, I reviewed the procedures developed by the NRC for the Maine Yankee trial plant review and partcipated in the conference call held on 27 June 1986. In general, my comments on the trial plant review are fully recorded on the tape recording of that meeting and adequately represented in the minutes of the meeting produced by R. J. Budnitz and dated 11 July 1986. Insofar as this is the case, my comments will not be repeated in this report. However, I would like to make an addition/clarification to my position which was not included in the recording or minutes.

First, regarding the cursory review of systems not part of the Group A functions (not-A systems), it should be clearly understood that the purpose of reviewing these systems is to identify any plant unique features which indicate that these systems are significantly more fragile with respect to earthquake than the Group-A systems. Under no circumstances is this to be construed as demonstrating that these not-A systems have a HCLPF greater than the RLEQ. By way of illustration, if a tank is located such that its rupture would cause failure of only not-A systems, this would be included in the review. Similarly, if for some reason there is similar equipment in both Group-A and not-A systems, but the not-A equipment has anchorages which are noticably inferior to the Group-A anchorages (easily recognizable during a walkthrough), this would be included in the review. However, if the above deficiencies existed for both the Group-A and not-A systems, the effect on the not-A systems would not be considered. Again, the purpose of the review is to determine only that the susceptibilities of the not-A systems to seismic events are generally of the same order as the susceptibilities of the Group-A systems, not whether they meet the HCLPF requirement. The review is very cursory in nature, and anything which cannot be clearly identified visually during the initial plant walkthrough is not to be pursued further.

Second, I wish to strongly express my disappointment with the decision that the expert panel would not constitute the peer review team, as was originally intended. In my opinion, the expert panel as a whole is much better suited to oversee the implementation of its review method and to assess its affectiveness and the required modifications than the group which has been selected as a peer review team. I wish to make clear that I will not "rubber stamp" the conclusions of the peer review team with regard to their assessment of the trial review process and guidelines, or blindly approve their suggested changes to the methodology based solely on their opinion of what changes should be made. However, it will obviously be extremely difficult for me to formulate an informed opinion on the validity of their peer review comments while not having any involvement in the trial plant review or its peer review process. The resolution to this dilemma is not clear. In the extreme case, it may be necessary for me to dissent from the peer review team's conclusions regarding the utility of the trial review guidelines.

R. Budnitz

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 16 1986

Dr. R. Murray
LLNL (L-197)
P.O. Box 808
Livermore, CA 94550

References: 1. 7/11/86 letter from R. Budnitz "Minutes of the Telephone Conference Call Meeting of June 27, 1986".

2. 6/18/86 memo from D. Guzy "Seismic Design Margins Program".

Dear Dr. Murray:

Reference 1 (enclosed) discusses the review of the Maine Yankee seismic margins review procedures by the Expert Panel on Seismic Design Margins. The Panel also discussed topics brought up by the NRC staff at our June 10, 1986 meeting held here at the Nicholson Lane Building. I would like to have clarification of the following items from Reference 1.

General: In Reference 2, I noted that I felt the review team's written procedures lacked depth, but that from the June 10th presentation it was clear to me that the team had definite ideas that would be finalized (and should be documented) as their review progresses.

Please document whether or not the Expert Panel felt the review procedures had sufficient detail. If not, how did the panel make their judgment on the adequacy of the procedures (on subsequent verbal discussions?) and what should be added to future written procedures?

Item 4

This discussion seems to back away from a previous Expert Panel position that for high frequency, high acceleration, low to moderate magnitude earthquakes, their guidelines were conservative. Please clarify if the Panel's position has changed.

Item 7

Although this discussion addresses an issue raised at the June 10th meeting, I'm not sure it directly answers a specific question. That is, in performing a margins review, can we use the values in the first column of Table 2-1 of NUREG/CR-4482 and make a statement about the ability of the plant to withstand earthquake up to the .3g level?

Item 9

I recommended that we delay the start of a separate study on CDFM vs. PRA fragility for the following reasons:

1. The end date for the proposed study was such that it could not be used in the Maine Yankee review.

2. EQE's proposal presumably was going to use both methods and thus there would be redundancy if a separate study was run concurrently.

3. FY 86 funding limitations wouldn't permit this study along with the other proposed activities.

It's a particular sore point with me that after repeated requests for more detailed 189 work statements and review procedures, the extent of EQE's CDFM/PRA fragility comparison was not stated until I asked the question at our June 10 meeting. Now I see that the Panel feels EQE's use of the PRA fragility approach is "less desirable" and that the number of CDFM examples is presumably inadequate to make any kind of a comparision. In light of these comments, I wonder if the scope of EQE's proposed use of the CDFM and FA methods was known and was a factor during LLNL's subcontractor selection process.

It would be useful to know what kind of study the Panel had in mind for "confirming the CDFM method". It seems to me that building on the current Maine Yankee review would be the most efficient way to go.

Dan Guzy

Dan Guzy
Mechanical/Structural Engineering Branch
Division of Engineering Technology

Enclosures: As stated

cc: R. Budnitz
J. E. Richardson
N. Anderson

Future Resources Associates, Inc.

2000 Center Street Suite 418 Berkeley, CA 94704 415-526-5111

September 26, 1986

Mr. Daniel J. Guzy
Engineering Branch
Division of Engineering Safety
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dan:

This letter is a reply to your memorandum of July 16, 1986 in which you raise several questions about the seismic margins program, both in general and in particular regarding the trial review at Maine Yankee.

I will address the points one-by-one, going down your memorandum. Therefore, a copy of your two-page memo is attached to this letter for convenience.

1. Your General Comment: The documentation of the NRC/LLNL review team's procedures has been insisted on by LLNL, and my understanding is that this documentation will be an integral part of the "deliverable" package that the review contractors will provide. The Expert Panel did not comment in detail about whether the procedures had sufficient detail, since at the time of the Panel's conference call of June 27 these procedures were still being developed. However, I am certain that the Expert Panel will comment on them later, because these procedures are intended to be a model of what should be followed by a review team implementing the "final" procedures to be finalized next year.

2. Your "Item 4" Comment: The Expert Panel believes that a separate study is required of the issue of high-frequency, high-acceleration, low-to-moderate-magnitude earthquakes. The Panel believes that its current guidelines are conservative, but that a separate study is needed to reaffirm this, to clarify areas where the conservatism may be inadequate (or may be excessive), and to understand this issue better.

3. Your "Item 7" Comment: This issue should be clear already, but I can clarify it again, by restating what the Expert Panel believes. The Panel believes that the first column should be used for "review level earthquakes" up to 0.30 g; the second column if the review level earthquake is between 0.30 and 0.50 g; and the third column for above 0.50 g. Suppose that the review-level earthquake is selected as, say, 0.29 g, so that the first column should be used. If, using the first column, a plant is found to have no Booleans (loosely, "accident sequences") that are

vulnerable, then the conclusion is that the plant-level HCLPF is at least as large as the review level earthquake being reviewed against. If on the other hand, a plant-level Boolean is identified, then the plant-level HCLPF can be quantified, for example at, say, 0.26 g (or whatever).

4. Your "Item 9" Comment: The Expert Panel has continued to believe that a separate study is needed to examine the relative merits of the CDFM and the PRA-fragility approaches to determining HCLPF values. The sooner this is begun, the sooner it will be completed. What is needed is a comparison of several different classes of equipment with several examples selected for each class. The first task is to establish a well-defined working definition of the CDFM method. Following this, for each specific equipment item being studied, the analysts should compute the HCLPF value using the CDFM method and compute a separate and independent HCLPF value using the fragility method. It should also be valuable, if possible, to use more than one group in the study, so as to illustrate any analyst-to-analyst differences in interpretation.

   The key aspect of the study is assuring that all assumptions made in each analysis for each equipment item are documented, so that differences between the HCLPF results may be traced and understood rather than merely stated.

   After this part of the study is completed, the analysts should be expected to make recommendations for any modifications in either the CDFM or the fragilities method, as needed to help resolve any differences found.

   The overall goal is to develop a method sufficiently "cut and dried" (that is, prescriptive) that it can be used routinely by the general engineering community for performing HCLPF calculations for seismic margins reviews.

Your final comment is correct--building on the current Maine Yankee review may be the most efficient way to proceed on the CDFM comparison study at this stage. But that probably means starting as soon as we can, since getting a study in place probably will take a few months.

I hope that this letter has answered your comments. If not, please let me know.

Sincerely yours,

Robert Budnitz

Robert J. Budnitz
Chairman
Expert Panel

RJB/sa

Attachment

cc: Expert Panel
R. Murray, LLNL

APPENDIX B

PEER REVIEW GROUP CORRESPONDENCE

PEER REVIEW GROUP CHARTER

R. J. Budnitz, Chairman

The objective of the Peer Review Group is to assure that the trial seismic margins review is executed in a fully competent and professional manner, uses methods that are at the state-of-the-art, follows the guidance established in NUREG/CR-4334 and NUREG/CR-4482, and takes cognizance of all relevant information. The sponsors of the study (Lawrence Livermore National Laboratory for the NRC and Maine Yankee as the plant owner) desire to utilize the results of the study, and require the Peer Review Group's assurance that the study is technically sound.

To accomplish its objective, the Peer Review Group will be provided full access to all materials, information, and methodologies that are inputs to and used by the study team. Access to the study team itself will occur through scheduled meetings to follow the study's progress. The Peer Review Group will also review draft reports and participate in walkdowns of the plant. Formal reporting and interface for the Peer Review Group will be through the Group's chairman to LLNL.

Future Resources Associates, Inc.

2000 Center Street Suite 418 Berkeley, CA 94704 415-526-5111

4 June 1986

Mr. J. Thomas
Duke Power Company
Design Engineering Department
422 South Church Street
P.O. Box 33189
Charlotte, North Carolina 28242

Dear Mr. Thomas:

This letter is being written on behalf of Lawrence Livermore National Laboratory's 'Seismic Margins Program". As you may know, I am the chairman of a newly-constituted "Peer Review Group" that will be assisting LLNL in carrying out a trial 'seismic margins review' of the Maine Yankee plant in the coming months. I am grateful that you have agreed to participate as a member of this Peer Review Group, and I look forward to interacting with you. I know that Bob Murray of LLNL has given you more details.

The purpose of this letter is to confirm the tentative dates of the first Peer Review Group meeting, to provide you with a tentative schedule for the rest of the Peer Review Group's activities, and to provide you with a copy of the Group's charter. (The charter is attached to this letter.)

The tentative schedule is as follows:

1) The first Peer Review Group meeting will be held at the Maine Yankee site on July 21-22-23. The first day will be used for a Group meeting, and the next two days will involve a walkdown of the plant along with the margins review team and the utility personnel. Bob Murray of LLNL should provide you with travel details soon.

2) In September, it will be necessary to devote 1 day (perhaps 2 days) to a review of the progress to that date; the study team will have provided their tentative findings and issues to us. There will not be a Peer Review Group meeting but we will discuss this by telephone.

3) In November, there will be a 1-day Peer Review Group meeting to go over what has been accomplished to date. This meeting could go to a second day but I hope it will not. The site of this meeting and its exact date is not yet known.

4) There will be a 1-2 day commitment in February, 1987 to review the final study report in draft form. This meeting will be at a site to be determined, but it will probably be held in Washington so that the Peer Review Group can interact with the relevant NRC staff members.

The Peer Review Group reports formally to LLNL, and specifically to Dr. Robert C. Murray, who is the designated LLNL contact. However, our report will be a public document that will also be of use to the Maine Yankee group, other utilities, as well as EPRI and the NRC. The report format will be a letter from me as Chairman, reporting on the Group's findings. It has been agreed that there will be an opportunity for any Peer Review Group member to write a separate minority report if needed, but the expectation is that the Chairman's letter can capture all of the review comments.

I hope that this letter and the attached Charter can clarify your role as a member of the Peer Review Group. If you have any questions, I will be happy to discuss them with you. I look forward to meeting you at Maine Yankee on July 21.

With warmest regards,

Sincerely yours,

Robert J. Budnitz

Robert J. Budnitz

cc: R.C.Murray

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 7 1986

MEMORANDUM FOR: Mr. James Richardson, RES Co-Chairman
Seismic Margins Working Group

FROM: Newton Anderson, NRR Co-Chairman
Seismic Margins Working Group

SUBJECT: COMMENTS ON THE NRC SEISMIC MARGINS PROGRAM AND TRIAL PLANT REVIEW

I have recently received a copy of a letter from Dr. Budnitz, Chairman of the Seismic Margins Peer Review Group to members of that group. The letter transmitted a tentative schedule for Peer Review Group meetings and a copy of the Peer Review Group Charter for the Maine Yankee Trial Review.

I have some concerns about both the charter and the schedule. I believe that the Charter does not reflect what my understanding is of the Peer Review Group function. It should be clearly articulated that we expect the Peer Review Group to not only provide "... assurance that the study is technically sound", but to endorse the technical results. At the conclusion of the Maine Yankee review they should be expected to provide their best judgment with regard to both the review procedure and the seismic capability of Maine Yankee, based on their collective expert opinions and backed by their professional reputations.

The other problem I have with the Charter is the statement on formal reporting. The Peer Review Group should not report through LLNL. They should report as an independent group, providing their results to both LLNL and NRC without any intermediaries. LLNL should provide administrative support to the group but I feel strongly they should maintain their independence.

If my comments with regard to the Charter are accepted, then I feel that the proposed schedule for Peer Review Group meetings is inadequate. It appears to me that the limited number of meetings and allotted review time is not sufficient to allow the group to "stake their professional opinions" on their assessment of the study results. The Peer Review Group members should decide what level of involvement they need based on what we expect of them.

I would also like to express my concern about some problems I see with the relationship between the NRC/LLNL Seismic Margins Program and the EPRI Seismic Margins Program. My basic concern is that there is a significant difference in the approach and in the scope of review. We need to work hard to bring these programs together. My understanding is that EPRI sees the NRC program as a study to "develop a requirement" to do a margins review and that the EPRI program is to provide the utilities with a tool for performing plant specific reviews to meet that requirement. If this is a reasonable way to proceed, then we certainly should work with EPRI to ensure the two programs are compatible. I do not mean to imply that we are not coordinating with EPRI. I know we are; but I do not see these differences being resolved.

I suggest that we hold a meeting with the NRC seismics margins working group in early August for the purpose of discussing these issues and developing a plan for resolving them.

Newton R. Anderson, NRR Co-Chariman
Seismic Margins Working Group

cc: Seismic Margins Working Group Members
Dr. Robert Budnitz
Dr. Garth Cummings, LLNL
T. Speis
B. Sheron
R. Bosnak

Future Resources Associates, Inc.

2000 Center Street Suite 418 Berkeley, CA 94704 415-526-5111

1 August 1986

TO: U.S. Nuclear Regulatory Commission
D. Guzy, Office of Nuclear Regulatory Research
P. Sears, Office of Nuclear Reactor Regulation

Maine Yankee Atomic Power Company
D. Whittier, Manager, Nuclear Engineering and Licensing

FROM: Robert J. Budnitz, Chairman of Peer Review Group for the Maine Yankee Seismic Margin Review Study

REF: MINUTES, FIRST MEETING OF THE PEER REVIEW GROUP

These are minutes of the first meeting of the "Peer Review Group" that is reviewing the technical competence of the "Seismic Margin Review Study" that is being undertaken on the Maine Yankee reactor plant under sponsorship of the U.S. Nuclear Regulatory Commission.

The meeting was held on Monday, July 21, 1986 and continued on Tuesday and Wednesday, July 22 and 23. Attending besides the chairman were Dr. John Reed; Mr. Loring Wyllie; and Mr. James Thomas. Mr. Thomas' attendance was limited to only the morning of the first day. Absent was Dr. Michael Bohn, the fifth member of the Peer Review Group.

Also attending were representatives of the NRC; the Maine Yankee staff and their Yankee Atomic Electric associates; Lawrence Livermore National Laboratory staff; LLNL's subcontractors performing the review, Energy Incorporated and EQE Inc.; and R. Kennedy, a consultant to Maine Yankee. The signup sheet for the opening session is attached as Attachment A; however, not all of these individuals participated in all of the meeting sessions.

Attachment B shows the agenda for the first day's session. On the second day, a plant walkdown was done by most of the attendees in teams of 4-6 each, preceded by a radiological protection briefing and followed by a de-briefing session in which technical issues raised during the walkdown were discussed. On the third day, another walkdown in the morning was followed by still another session in which technical issues were discussed. The Peer Review Group then adjourned its session just after noon, and the study team continued its discussions and its walkdown, which would last for the remainder of the week.

The minutes of the meeting will be presented as numerically ordered topics, as follows:

1) The first session began with a briefing by Robert Murray of LLNL, who presented the background of the project. Murray discussed the purpose of the trial margin review, the development of the methodology by the NRC's "Expert Panel of Seismic Margins", the schedule of the review, its structure, and the relationships among the parties. During the course of Murray's presentation an important comment by A. Thadani of NRC was made in which Thadani pointed out that any licensing action that might result from this study would only occur after its completion. NRC, according to Thadani, was awaiting the results because there are methodological issues as well as technical issues about the Maine Yankee plant that are being studied in this trial review.

2) Robert Budnitz discussed the interactions that have occurred so far between this study effort and a parallel effort being undertaken by the Electric Power Research Institute, which is studying the seismic margin at Duke Power's Catawba station. Budnitz pointed out that recent discussions have assisted both study teams to understand in what ways their respective methodologies are similar or different. Continuing interactions will be encouraged.

3) James Thomas discussed the importance of interactions between this study and related technical work being done under the NRC's A-46 program and under the industry's SQUG (Seismic Qualifications Utility Group) effort. Taking cognizance of these other efforts will be important for the study team.

4) William Henries of Yankee Atomic then discussed the current status of the Maine Yankee plant in terms of its seismic capacity. He provided background on the history of the seismic design of the plant, recent utility-sponsored studies that provided information to assist them in understanding their plant, and recent actions taken to modify the plant's seismic performance.

5) David Moore of Energy Incorporated, who is leading the systems-analysis team performing this review under LLNL subcontract, made a presentation that provided information on the approach being taken. Moore's viewgraphs are attached to these minutes as Attachment C. He pointed out that there are a few key issues that must be addressed, and asked for discussion and guidance from the group on how best to approach them. For example, because a walkdown inside the containment will not be possible, it will be necessary to make some assumptions about the presence of small LOCAs, and the approach being taken to this issue was covered. Also, the handling of the possible sequences in which control rod function might be compromised was discussed, and Moore's review approach to reactor internals and boric-acid safety injection was covered.

6) A lunch-time break was taken so that the Peer Review Group could go into executive session. After that executive session, the Peer Review Group suggested that minor modifications to the draft PRG charter should be made to reflect more accurately the actual approach being taken. The revised charter which was discussed by the PRG with those present is shown as Attachment D. The differences between this version and the earlier draft are a more explicit

statement of the fact that the Peer Review Group is an independent group reporting to NRC and Maine Yankee, and a clarification of the Group's role vis-a-vis assuring that the methodology is being followed.

7) M.K. Ravindra of EQE Inc. made a presentation in which he discussed his group's approach, as the fragilities subcontractor for this effort, in carrying out their work. His viewgraphs are attached as Attachment E. Technical topics covered included Ravindra's approach to HCLPF determinations using the fragilities method and the CDFM (conservative deterministic fragilities method) for analyzing capacities. The handling of seismic capacities for those numerous components (valves, etc.) which can only be studies in a sampling way rather than in a 100 %-analysis way was covered. Also, a discussion took place on how the study will examine those 'Group B' components that require study to assure that there are not unusual issues involved in their capacities.

8) The structure of the Final Report was covered, so that all parties present could understand how it will be structured, who will write which sections, and its schedule. The process whereby the Peer Review Group will be able to review a draft version of this report toward the end of the project was discussed. It is anticipated that there will be a meeting of the participants to enable the Peer Reviewers to interact.

9) The PRG's reporting was discussed. It was agreed here, as in earlier meetings, that a letter from the PRG Chairman would provide the Group's final comments on the study. Concurrence by the other members, with the opportunity for any individual PRG member to provide individual comments as a minority report if desired, would be the method used.

10) An extensive discussion took place on how the systems analysis team would handle 'non-seismic induced failures' in their systems analysis, when a potential accident sequence might involve both these and seismic-induced failures. The Energy Incorporated study team will do an analysis which will incorporate these non-siesmic-induced failures where their presence will make a significant difference in the overall risk profile of the plant.

11) The Monday session ended just after 5:00 PM.

12) On Tuesday (July 22) a radiological briefing and a plant walkdwon in small groups took most of the day. A debriefing session from 4:30 to 7:00 PM ended the day. At that session, various technical issues uncovered during the day's walkdown were discussed, to allow the next day's walkdown to be more effective. An extensive disussion of the capacity of the DC batteries took place, along with discussions of the capacities and functions of several tanks. Some of the walkdown teams had not completed their entire first-pass tours, so this discussion was partly of the character of assuring that these teams looked at and studied plant features that some of the other groups had highlighted.

13) During this session late on Tuesday afternoon the role of the Peer Review Group in interacting with the study participants was discussed. The independence of the PRG must be assured, but it is also important that technical issues of concern to the PRG be provided to the study team rather than 'kept to the end'. The handling of this aspect of the project was covered.

14) On Wednesday morning (July 23)the plant walkdown continued, until 11:00 when the meeting of the Peer Review Group continued for just over one hour. In this final session, PRG members discussed again the issue of assuring that independence for their work would be a fact as well as a perception. Also, the interactions between the NRC-sponsored study team and the utility staff were discussed, to assure that there would not be improper influence by the utility over the project outcome.

15) The Peer Review Group's final discussion and comments prior to adjournment were of the character that the study seems to be 'on track' in a technical sesne so far, although of course this is a very preliminary observation, and was made by only three of the PRG members (Budnitz, Reed, Wyllie), the other two (Thomas, Bohn) being absent.

16) The PRG meeting adjourned just after noon on Wednesday, 23 July.

17) The next meeting of the Peer Review Group will be held on September 30, at a location in the San Francisco Bay area that will be identified by the LLNL team soon. The subsequent meeting of the PRG will be held in November in conjunction with the second plant walkdown. It will be held at the Maine Yankee plant site. While dates have not been firmly set, the dates of November 17-18-19 were written down tentatively, and are being held by all participants pending further developments.

18) These minutes, written by the PRG Chairman (Budnitz), are being sent to the other members for their review and comment, after which they will be made final. The draft version was not circulated outside the Peer Review Group.

Robert J. Budnitz

Robert J. Budnitz, Chairman

Attachment A: Attendance signup sheet, 7/21/86
Attachment B: Agenda for first day's session, as prepared by R. Murray of LLNL
Attachment C: Viewgraphs for D. Moore's presentation (Energy Incprporated)
Attachment D: Revised charter for Peer Review Group
Attachment E: Viewgraphs for M. K. Ravindra's presentation (EQE Inc.)

ADDITIONAL ITEM:

(This item was not in Budnitz's notes for the meeting, but Reid and the others remembered that it was discussed.....probably while Budnitz was out of the room. It is presented here because it is deemed important additional guidance from the Peer Review Group):

19) The Peer Review Group's interpretation of NRC's requirement on the input spectrum to be used in this review is that a NUREG/CR-0098 spectrum, using 50th percentile amplification factors,and anchored to 0.30 g zero period acceleration, is the proper ground motion to use. No variability in this input spectrum is to be used. Reference for NRC's guidance is the memorandum from D.M. Crutchfield to P.M.Sears, dated 7 May 1986.

ATTACHMENT A

JULY 21, 1986 at MAINE YANKEE

NAME	INSTITUTION	TELEPHONE
Robert Budnitz	Future Resources Associates	(415) 526 [illegible]
ASHOK THADANI	NRC	(301) 492-4553
GARTH CUMMINGS	LLNL	(415) 422-4949
Bob Murray	LLNL	(415) 422-0308
Bob Kennedy	SMC	(714) 777-2163
Jim Thomas	Duke Power Co.	(704) 373-4612
Loring Wyllie	Degenkolb Assoc	(415) 392-6952
John Reed	JBA	(415) 969-8212
David Moore	Energy Inc	(206) 854-0080
MARC QUILICI	ENERGY INC	(206) 85[illegible]-00[illegible]
M.K Ravindra	EQE Inc	(714) 852-9299
Dan Guzy	NRC	301-443-7710
JON YOUNG	ENERGY INC	(206) 8540080
Bill HENRIES	YANKEE ATOMIC	(617) 8728100
Nilesh Chokshi	NRC	301-492-8347
PETE PRASSINOS	LLNL	(415) 423-6758
Greg Hardy	EQE Inc.	(714) 852-9299
STEPHEN EVANS	MAINE YANKEE	207-623-3521
Sam Swan	EQE, Inc.	(415) 495-5500
Wm J. METEVIA	YANKEE ATOMIC	617-[illegible]
PATRICK SEARS	NRC/NRR	301 492 8006
Joe McCumber	Yankee Atomic	617-872-8100
Cornelius Holden	NRC/Resident	207-882-7519
Phil Hashimoto	EQE Inc.	714-852-9299

ATTACHMENT B

Monday Meeting Agenda

Maine Yankee Atomic Power Station

Staff Building

July 21, 1986

9:00	Project Overview/R. C. Murray
9:15	Yankee Atomic/Maine Yankee Plant Overview
9:45	Systems Status/Energy Inc.
10:30	Break
10:45	Fragility Status/EQE, Inc.
11:30	Open Discussion (collect Health Physic paperwork)
12:00	Lunch (at plant)
1:00	Peer Review Group Discussion/R. J. Budnitz
5:00	Adjourn

ATTACHMENT C

SYSTEMS ANALYSIS

ENERGY INCORPORATED

TEAM:

DAVID MOORE
JON YOUNG
MARC QUILICI

SYSTEMS ANALYSIS PROCEDURES

STEP 2 - Initial Systems Review

STEP 4 - First Plant Walkdown

STEP 5 - System Modeling

STEP 6 - Second Plant Walkdown

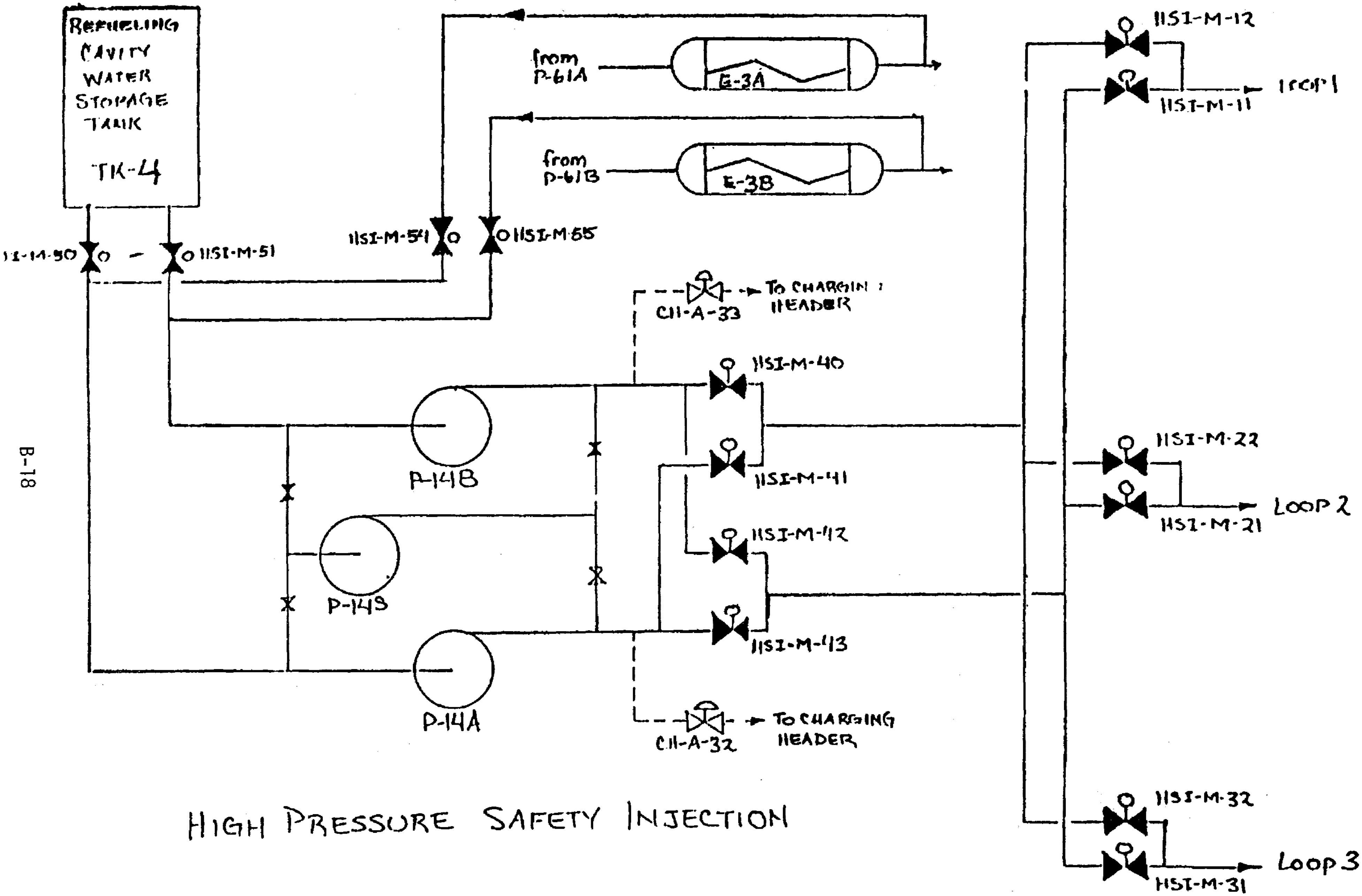
STEP 7 - System Modeling Analysis

STEP 8 - Margin Evaluation of Components and Plant

GROUP A SYSTEMS

HPSI	HIGH PRESSURE SAFETY INJECTION
AFW	AUXILIARY FEEDWATER(includes EMERGENCY FEEDWATER)
ASDHR	ALTERNATE SHUTDOWN DECAY HEAT REMOVAL
BAT	BORIC ACID TRANSFER
PPC	PRIMARY PRESSURE CONTROL
SPC	SECONDARY PRESSURE CONTROL
ACP	AC POWER
DG	DIESEL GENERATORS
DCP	DC POWER
PCC	PRIMARY COMPONENT COOLING WATER
SCC	SECONDARY COMPONENT COOLING WATER
SWS	SERVICE WATER SYSTEM
	ACTUATION (includes RPS, SIAS, and maybe RAS, CSAS, CIS)

SYSTEM:	HIGH PRESSURE SAFETY INJECTION (HPSI)	
SAFETY FUNCTION:	Inject borated water into the reactor vessel immediately after a LOCA. Also for feed and bleed, post-accident core cooling and additional shutdown capability during rapid cooldown of RCS.	
SYSTEM COMPONENTS:		
Tanks:	TK-4	Refueling Cavity Water Storage Tank
Pumps:	P-14A (N.O.)	Charging (HPSI) Pump
	P-14B (S)	Charging (HPSI) Pump
	P-14S (Spare)	Charging (HPSI) Pump
Heat Exchangers:	E-3A	Residual Heat Exchanger
	E-3B	Residual Heat Exchanger
SUPPORT SYSTEMS:		
AC Power:	4160V Emergency Bus 5	
	4160V Emergency Bus 6	
DC Power:		
Air:		
HVAC:		
Pump Cooling:	PCC	P-14A-2, 3 Lube Oil Pumps
	SCC	P-14B-2, 3
		P-14S-2, 3
Actuation:	SIAS	Low Pressurizer Pressure
		High Containment Pressure



HIGH PRESSURE SAFETY INJECTION

VALVE TABLE

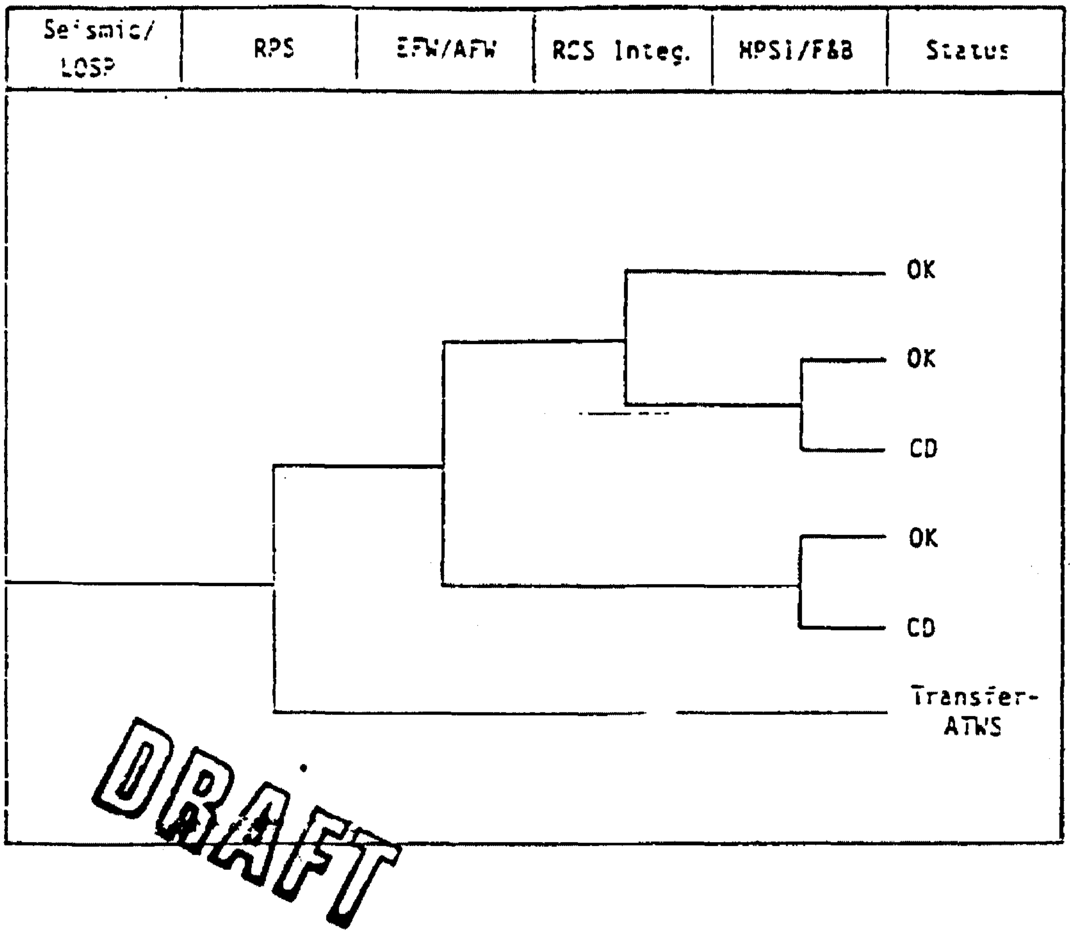
Valve	Description	Power (SOV)	Normal Position	Operating Position (Actuation)	Fail Position
AFW-A-101	Flow control to SG E-1-1	120VAC 1A (1201A1)	O	C(SLR)	O
AFW-A-201	Flow control to SG E-1-2	120VAC 1A (1201B1)	O	C(SLR)	O
AFW-A-301	Flow control to SG E-1-3	120VAC 1A (1201C1)	O	C(SLR)	O
AFW-A-338	Flow control isolation valve (AFW-A-101)	120VAC 3A (1205A)	O	C(SLR)	
AFW-A-339	Flow control isolation valve (AFW-A-201)	120VAC 3A (1205B)	O	C(SLR)	
AFW-A-340	Flow control isolation valve (AFW-A-301)	120VAC 3A (1205C)	O	C(SLR)	
BA-A-32	Boric acid VCT isolation valve	(210Z)	C	C(SIAS)	C
BA-A-80	Boric acid VCT (bottom) isolation valve		O	C	
BA-F-30	Boric acid flow control valve	(210Y)	C	C(SIAS)	C
BA-M-36	Emergency boration isolation valve	MCC 8A	C	O(MAN)	AI
BA-M-37	Emergency boration isolation valve	MCC 7A	C	O(MAN)	AI
CH-A-32	HPSI pump B discharge to charging header	BATT-1 (255)	O	C(SIAS)	
CH-A-33	HPSI pump A discharge to charging header	BATT-1 (254)	O	C(SIAS)	

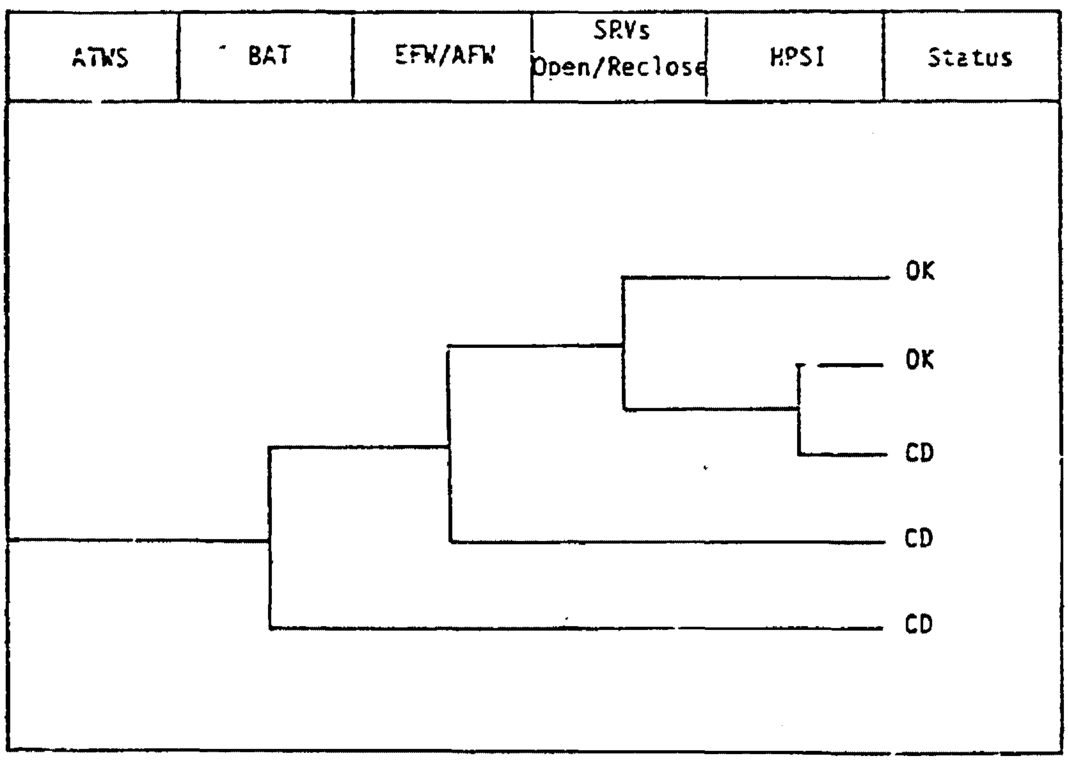
Front Line Systems \ Support Systems		AC Power		DC Power		CCW		SIAS		Inst Air		HVAC i/o Pump Cooling	
		4160 V Bus 5 480 V Bus 7 VB 1,2	4160 V Bus 6 480 V Bus 8 VB 3,4	Dist Cab 1,2	Dist Cab 3,4	PCC	SCC	Train A	Train B	TK-111	TK123		
HPSI	P14A to RCS	X		X		X		X					
	P14B to RCS		X		X		X		X				
	From RWST to P14S	X		X				X					
	P14S	*	*	+	+	*	*	*	*				
	From P14S to RCS	X		X				X					
AFW	P25A		X		X								
	P25B	*	*							*	*		
	P25C	X		X									
PORVs	PR-S-14	X											
	PR-S-15		X										
BAT	P6A	X		X									
	P6B	X		X									
	P6C		X		X								
	To Ch Pump P14A	X											
	To Ch Pump P14B		X										

DRAFT

Support Systems \ Supported Systems			AC Power 4160V Bus 5	AC Power 4160V Bus 6	AC Power 480V Bus 7	AC Power 480V Bus 8	AC Power Vital 120V VB 1	AC Power Vital 120V VB 2	AC Power Vital 120V VB 3	AC Power Vital 120V VB 4	AC Power Diesel Gen DG-1A	AC Power Diesel Gen DG-1B	125V DC Power DC 1	125V DC Power DC 2	125V DC Power DC 3	125V DC Power DC 4	CCW PCC	CCW SCC	SWS	SIAS SISA	SIAS SISB	HVAC
AC Power	4160 V	Bus 5									X		X									
		Bus 6										X			X							
	480 V	Bus 7	X										X									
		Bus 8		X											X							
	Vital 120 V	VB 1											X									
		VB 2												X								
		VB 3													X							
		VB 4														X						
	Diesel Gen.	DG-1A											X				X					
		DG-1B													X			X				
125 V DC Power		DC 1			X																	
		DC 2			X																	
		DC 3				X																
		DC 4				X																
CCW		PCC	X																X			
		SCC		X															X			
SWS					X	X																
SIAS	Inst	A											X									
		B												X								
		C													X							
		D														X						
	Act.	SISA											X									
		SISB													X							
HVAC																						

DRAFT





Seismic Margin Event Trees (Preliminary)

PEER REVIEW GROUP CHARTER

(21 July 1986)

The objective of the Peer Review Group is to assure that the trial seismic margins review, following the guidance established in NUREG/CR-4334 and NUREG/CR-4482, is executed in a fully competent and professional manner, uses methods that are at the state-of-the-art, and takes cognizance of all relevant information. The sponsors of the study (Lawrence Livermore National Laboratory for the NRC and Maine Yankee as the plant owner) desire to utilize the results of the study, and require the Peer Review Group's assurance that the study is technically sound.

To accomplish its objective, the Peer Review Group will be provided full access to all materials, information, and methodologies that are inputs to and used by the study team. Access to the study team itself will occur through scheduled meetings to follow the study's progress. The Peer Review Group will also review draft reports and participate in walkdowns of the plant. A formal report for the Peer Review Group will be made by the Group's chairman to NRC and Maine Yankee. It is understood that the Peer Review Group's report will be a public document.

ATTACHMENT E

NRC SEISMIC MARGINS PROGRAM

TRIAL PLANT REVIEW

FRAGILITY ASPECTS

STATUS REPORT

PRESENTED BY

M.K. Ravindra
G.S. Hardy
P.S. Hashimoto
S.W. Swan

EQE Incorporated
Newport Beach, CA

PRESENTED TO

PEER REVIEW GROUP

JULY 21, 1986

OUTLINE

- Initial Screening
- Identify Target Areas for Walkdown
- Walkdown Procedures
- Documentation
- Outstanding Issues

INITIAL SCREENING

- Categorize Maine Yankee Components Into Generic Component Categories Identified by Panel

- Pre-screen Components <u>In</u> or <u>Out</u> Based on Panels Recommended Guidelines

- Identify Areas of Concentrated Effort for Each Generic Component Class

MAINE YANKEE SEISMIC MARGINS REVIEW

SYSTEM: HIGH PRESSURE SAFETY INJECTION (HPSI)

COMPONENT		INITIAL SCREENING IN	INITIAL SCREENING OUT
TK-4	Refueling Cavity Water Storage Tank	X	
P-14A	Charging (HPSI) Pump		X 1
P-14B	Charging (HPSI) Pump		X 1
P-14S	Charging (HPSI) Pump		X 1
E-3A	Residual Heat Exchanger	X	
E-3B	Residual Heat Exchanger	X	
Bus 5	4160V Emergency Bus	X 2	
Bus 6	4160V Emergency Bus	X 2	
P-14A-2,3	Lube Oil Pumps (Pump cooling)		X 1
P-14B-2,3	Lube Oil Pumps (Pump cooling)		X 1
P-14S-2,3	Lube Oil Pumps (Pump cooling)		X 1
Actuations		X 3	

1 Anchorage must be inspected during walkdown and verified adequate.

2 Cabinet anchorage & attached component anchorage must be inspected during walkdown and verified adequate.

3 Actual components yet to be identified.

MAINE YANKEE SEISMIC MARGINS REVIEW

SYSTEM: Auxiliary Feed Water (AFW)

COMPONENT		INITIAL SCREENING IN	INITIAL SCREENING OUT
TK-21	Demineralized Water Storage Tank	X	
P-25A	Emergency Feed Pump		X 1
P-25B	Emergency Feed Pump		X 1
P-25C	Emergency Feed Pump		X 1
T-1	Turbine for P-25B (Powered from Main Steam)	X	
Bus 5	4160V Emergency Bus	X 2	
Bus 6	4160V Emergency Bus	X 2	
E-86A	Oil Cooler	X	
E-86B	Oil Cooler	X	
E-86C	Oil Cooler	X	
Actuations		X 3	
Instrumentation		X 3	

1 Anchorage must be inspected during walkdown and verified adequate.

2 Cabinet anchorage & attached component anchorage must be inspected during walkdown and verified adequate.

3 Actual components yet to be identified.

IDENTIFY TARGET AREAS FOR WALKDOWN

- Locate All Components for Walkdown on Plant Layout Drawings

- Identify Buildings and Areas Requiring Access for Walkdown

- Based on Initial Screening-Identify Specific Component Areas for Concentrated Review, ie (Anchorage, Lateral Restraints, etc.)

- Develop walkdown Data Sheets for Each Generic Component Class Defining Areas of Concentrated Walkdown Effort

WALKDOWN PROCEDURES

- Perform Walkdown for Identified Components

- Address Areas Identified in Initial Screening Requiring Concentrated Review

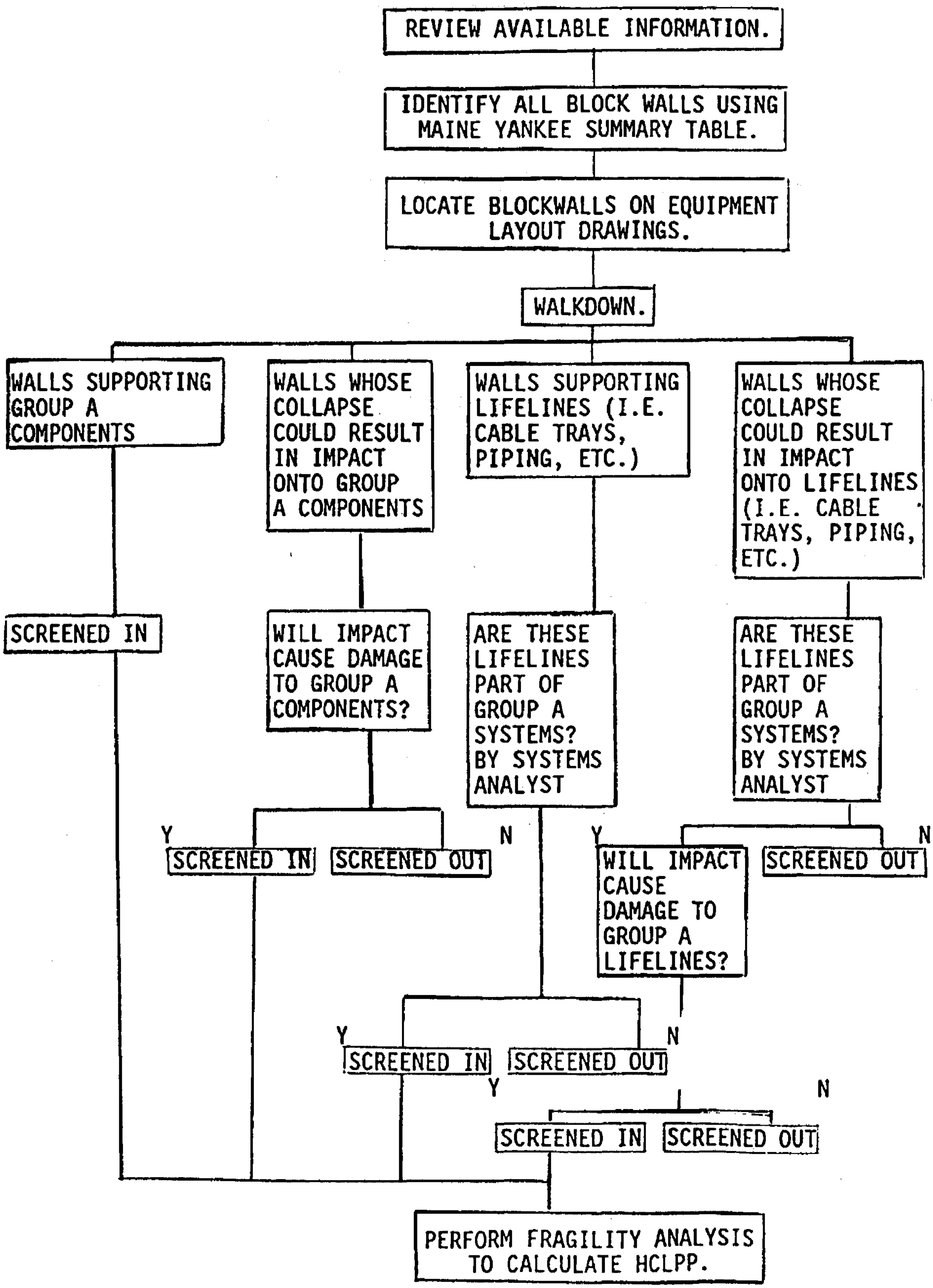
- Confirm Screening Criteria of Panel is Satisfied for Each Component

- Identify Areas That May Require Additional Review

STRUCTURES SCREENING PROCEDURE

Component	Screening	Comments
Structures	Out	Structures housing Group A components are typically categorized as Class I in the FSAR. Class I structures were designed for the 0.1g hypothetical earthquake using the ACI 318-63 and AISC codes. Any gross structural deficiencies will be identified by review of the design drawings and walkdown.
Structure Impact	Out	Class I structures are either cast integral with each other or are separated by three inch gaps. This will be confirmed by walkdown.
Block Walls	In	A comprehensive screening procedure has been developed.
Yard tanks	In	Information from drawings will be supplemented by walkdown.
Soil Liquefaction	Out	Structures and yard tanks are founded on rock.
Control and Battery Room Ceilings	In	Presence and adequacy of safety wiring will be confirmed by walkdown
Dams, Levees, and Dikes	In	Nearby dike will be reviewed in walkdown

BLOCK WALL EVALUATION PROCEDURE



DOCUMENTATION

- Review of Design Calculation/Qualification Test Reports
- Conclusions of Previous Studies
- Walkdown Data Sheet
- Walkdown Photographs
- HCLPF Calculations

OUTSTANDING ISSUES

- Lack of Qualification Data

- Extent of Review for Components Identified as C

- CDFM Method Requires Site Specific Spectrum

- Reactor Internals

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 1/

JOB NO. 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ______ DATE ______

FACILITY: Maine Yankee Atomic Power Station

COMPONENT: ______________________

Line Number & Diameter ______________________

LOCATION: Building = ______________________
Elevation = ______________________
Room = ______________________

1.0 CONCENTRATED AREAS OF REVIEW:

1.1 Pipe Flexibility:

- Piping run = ______________________
- Piping to equipment = ______________________
- Building penetrations = ______________________

Note: Provide details on building penetrations if low capacity is observed. Indicate dimensions and details in Section 4 sheet 2.

Photograph Roll No. ______ Frame No.s ______

1.2 Pipe Condition:

- Corrosion = ______________________
- Brittle connections = ______________________
- Cast iron = ______________________

1.3 Support Details:

- Type of anchor point
  Directional = ______________________
  Full restraint = ______________________
- Spans between supports = ______________________
- Support anchorage details = ______________________

Photograph Roll No. ______ Frame No.s ______

1.4 Joints and Connections:

- Threaded = ______________________
- Socket welded = ______________________

Photograph Roll No. ______ Frame No.s ______

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 2/

JOB NO. 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ______ DATE ______

3.0 ADDITIONAL COMMENTS OR OBSERVATIONS:

4.0 Sketch details of building penetrations if low capacity is observed.

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 1/

JOB NO. 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ______ DATE ______

FACILITY: Maine Yankee Atomic Power Station

COMPONENT: ______

LOCATION: Building = ______
Elevation = ______

1.0 COMPONENT DATA:

Plant ID Number = ______
Manufacturer = ______
Model = ______
Function = ______

Photograph (overall) Roll No. ______ Frame No.s ______

2.0 AREAS REQUIRING DETAILED REVIEW:

2.1 Battery Rack Anchorage:

Number and size of anchor bolts = ______
Type of anchor bolts = ______
Description of foundation = ______

Photograph Roll No. ______ Frame No.s ______

Note: Provide a sketch of anchorage plan with dimensions and indicate any foundation deficiencies observed in space provided under Section 4 sheet 2.

2.2 Battery Rack Support:

Type of support = ______
Overall dimensions = ______
Lateral Restraints = ______
Type of member connections = ______

Photograph Roll No. ______ Frame No.s ______

Note: Provide a sketch of the rack indicating lateral restraints and dimensions in Section 5 sheet 3.

2.3 Batteries:

Type of batteries = ______
Description of battery spacers = ______

Photograph Roll No. ______ Frame No.s

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 2/

JOB NO 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ______ DATE ______

3.0 ADDITIONAL COMMENTS OR OBSERVATIONS:

4.0 Sketch anchorage plan with dimensions and note any foundation deficiencies observed.

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 3/

JOB NO. 8227-09 JOB LLNL Seismic Margin Review BY ____ DATE ____

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ____ DATE ____

5.0 Sketch battery rack indicating lateral restraints and dimensions.

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 1/

JOB NO. 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT ~~Plant Walkdown Data Sheet~~ CHK'D ______ DATE ______

FACILITY: Maine Yankee Atomic Power Station

COMPONENT: ____________________

LOCATION: Building = ____________________
Elevation = ____________________

1.0 COMPONENT DATA:

Plant ID Number = ____________________
Manufacturer = ____________________
Model = ____________________
Function = ____________________

Photograph (overall) Roll No. ______ Frame No.s ______

2.0 AREAS REQUIRING DETAILED REVIEW:

2.1 Anchorage: Number and size of anchor bolts = ______
Type of anchor bolts = ______
Description of foundation = ______

Photograph Roll No. ______ Frame No.s ______

Note: Provide a sketch of anchorage plan with dimensions and indicate any foundation deficiencies observed in space provided below.

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 2/

JOB NO. 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ______ DATE ______

3.0 ADDITIONAL COMMENTS OR OBSERVATIONS:

Note any system interactions.

Future Resources Associates, Inc.

2000 Center Street Suite 418 Berkeley, CA 94704 415-526-5111

14 October 1986

TO: U.S. Nuclear Regulatory Commission
D. Guzy, Office of Nuclear Regulatory Research
P. Sears, Office of Nuclear Reactor Regulation

Maine Yankee Atomic Power Plant
D. Whittier, Manager, Nuclear Engineering and Licensing

FROM: Robert J. Budnitz, Chairman of Peer Review Group for the Maine Yankee Seismic Margin Review Study

REF: MINUTES AND REPORT, SECOND PEER REVIEW GROUP MEETING

This is the report of the second meeting of the "Peer Review Group" that is reviewing the technical competence of the "Seismic Margin Review Study" that is being undertaken on the Maine Yankee reactor plant under sponsorship of the U.S. Nuclear Regulatory Commission.

The meeting was held at the San Francisco Airport Clarion Hotel on Tuesday, September 30, 1986. Attending besides the chairman were all four of the other Peer Review Group members: Michael Bohn; John Reed; James Thomas; and Loring Wyllie.

Also attending were representatives of the NRC; the Maine Yankee staff and their Yankee Atomic Electric associates; Lawrence Livermore National Laboratory staff; LLNL's subcontractors performing the review (EI International and EQE, Inc.); and R. Kennedy, a consultant to Maine Yankee. The sign-up sheet for the day's session is attached as Attachment A. All attendees attended essentially the entire meeting.

Attachment B shows the agenda for the day's session. The meeting began at 9:00 AM and ended at 6:00 PM. The order of the agenda was followed quite closely, although some of the times were different. All agenda topics and presentations were given, and most of the day was used for technical discussion of aspects of the on-going margins review study.

The minutes of the meeting will be presented as numerically ordered topics, as follows, with commentary included:

1) R. Murray of LLNL led the introductory session, which consisted mainly of a discussion concerning the future schedule for the review effort. The tentative schedule, as distributed by Murray, is shown as Attachment C. The schedule is judged to be satisfactory, in the sense that the project is currently 'on schedule' and the participants believe that it can remain on schedule according to the schedule shown in Attachment C. The following specific dates have been decided on for the schedule:

January 15, 1987: first full draft report due from contractors, for limited distribution only to Peer Review Group, Maine Yankee, LLNL, 2 or 3 NRC staff

January 22, 1987: next meeting of Peer Review Group, in San Francisco; attendance by invitation of PRG only

February 12, 1987: second draft due, to be distributed to wider distribution including broadly in NRC

February 19, 1987: meeting in Washington with NRC in-house "Working Group on Seismic Margins".

It was also decided that another Peer Review Group meeting would be held sometime after the February time period, to study the final version of the report to be prepared after the February comments are in.

2) Daniel Guzy of NRC gave a brief summary of the July meeting of the NRC "Working Group on Seismic Margins". He discussed the Working Group's current thinking, including the selection of a BWR plant as the subject of a possible second trial margins review study. Newton Anderson of NRC, who is co-chairman of the NRC Working Group, provided additional comments. This part of the meeting was mainly for information purposes and elicited very little discussion.

3) M. Ravindra of EQE, Inc. gave an overview presentation of the work and preliminary findings of his group, who are the subcontractors doing the fragilities analysis of Maine Yankee. He discussed a list of key items that will be examined in detail, and another list of items that the fragilities team has already

analyzed. This presentation was the subject of much technical discussion on several items of equipment, although a few key items were put off until more detailed discussions scheduled for the afternoon. Ravindra discussed various data that the utility had furnished to assist the fragilities evaluation, and also discussed his group's use of various experience data in their analyses. It was emphasized that the Peer Review Group will need access to all data that will be relied on in these evaluations, in order to review its applicability.

Among the technical topics covered in Ravindra's presentation were service water piping; heat exchanger supports; valves with extended operators; HVAC fans and blower supports; cable tray motion and a possible interaction with valve operators; cable trays themselves, including Maine Yankee pull tests; steel frame buildings (specifically the pump house); and several others. Maine Yankee agreed to provide the summary report to the Peer Review Group which presents the results of the pull tests conducted on concrete inserts.

The main thrust of this part of the session was to familiarize the Peer Review Group with the approach being taken by the EQE team.

4) David Moore of EI International next presented an overview of the systems analysis work that his firm is doing. He discussed the first walkdown, and a few tentative 'lessons learned' that may assist others in preparing for a first walkdown, such as doing more detailed preparatory study of HVAC and actuation systems. He emphasized that the first round of systems analysis is well under way, with seven fault trees complete and a few more under development. His group is now going back to answer specific questions that have arisen, in preparation for the second walkdown. The data sources being used were discussed, and it was again emphasized by the Peer Review Group that access to those data sources will be needed. Also, the method of combining seismic and non-seismic failures was covered.

Moore's group will be doing some system cut sets soon, to obtain early guidance for EQE on what items seem to be more important. However, they do not plan to do sequence cut sets until after the second walkdown, as is called for in the NUREG/CR-4334 and 4482 guidance.

There was discussion of certain specific items, and the Peer Review Group was provided with some detailed fault trees for their study.

5) Robert Kassawara of EPRI next gave a presentation covering EPRI's on-going seismic margins review project at Catawba. This review effort is on a schedule not very different from the schedule for NRC's review at Maine Yankee: initial report due early in 1987, final report due a few months later.

While this presentation was mainly intended for information, it generated significant discussion concerning the differences between EPRI's and NRC's approaches. EPRI's approach is emphasizing the CDFM method for HCLPF determinations, while NRC's approach has not yet settled on one or another method, although it favors the CDFM method if it can be studied enough. EPRI's approach also uses a 'success path' method for the systems analysis rather than a fault-tree/event-tree method. There was much discussion about how to cope with small LOCAs inside containment, which perhaps cannot be walked down and analyzed well. EPRI's analysis at Catawba is studying some relay chatter, but their analysis is limited to a very few relays on their chosen success path; NRC's approach is not considering relay chatter for the time being, because further research is needed.

The general flavor of this discussion was that both the EPRI review at Catawba and NRC's review at Maine Yankee will be able to learn much from each other, and continuing cooperation and coordination will be encouraged.

6) Philip Hashimoto of EQE, Inc. gave a long and detailed presentation about three specific technical items that EQE is analyzing: the RWST, the pump house, and one block wall. He presented both CDFM and fragilities analyses of HCLPF values, and his presentation provided a vehicle for extensive discussion about the methodology used, the data relied on, approximations introduced, and whether EQE's analytical approaches were generic or only specific to the item being analyzed.

It was pointed out in the discussion that in the analysis approach used for the RWST, the deflection compatibility between anchor bolt and water resistance modes is not provided. This needs to be carefully investigated before the capacities from these two modes are combined.

In the course of this discussion, much detail was covered that will not be discussed here. The principal thrust of the Peer Review Group's comments were that there is a need for careful study of both the CDFM and fragilities methods, since the 'results' for HCLPF seem to depend greatly on assumptions made and data chosen. One key aspect was the Peer Review Group's observation that experience data, if available, play a key part of the underlying approach to HCLPF analysis. It was requested that realistic resistance modes be analyzed rather than using design assumptions which are often overly conservative and do not represent the realistic response.

7) Gregory Hardy of EQE, Inc. gave a detailed presentation about two other items being analyzed, the diesel day tank and one inverter located in the switchgear room. This discussion followed the same tenor as the discussion of Hashimoto's work just earlier. There was extensive interaction between the Peer Review Group and the analyst team, and much detailed discussion of data bases and assumptions. Again, discussion of experience data played a key role.

8) Greg Hardy then gave a briefer presentation, for information purposes only, about progress in tracking down fragilities information on three key issues: the lead-antimony batteries, the reactor internals and CRDMs, and the fire water threaded pipe. In the battery case, he has been unsuccessful so far in identifying relevant test or experience data. Therefore, it is the Peer Review Group's understanding that analysis will be performed with and without these batteries. For the other two cases, the information now in hand, either due to configuration or fragility aspects, should be adequate for the purposes of this margins analysis. Concerning the internals and CRDMs, the comment was made that Combustion Engineering's cooperation has been outstanding.

9) David Moore of EI, in reply to Peer Review Group questions, discussed how his analysis group is coping with Group B functions, especially those items needed to support long-term heat removal. He provided a rationale for his approach, which will be the subject of later review by the Peer Review Group. He then discussed in more detail the treatment of non-seismic failures, with an emphasis on those that might compromise both redundant trains of some function. The question of how to combine these failures with seismic-induced failures in a HCLPF analysis was covered, but not resolved. What approach to take remains an open question, and the NRC Expert Panel's guidance on this subject is inadequate.

The Peer Review Group discussed a problem with incorporating non-seismic failures after an earthquake, if their incorporation can change the HCLPF level found. The problem arises if the way they affect post-earthquake plant response at the SSE level is identical to the way they affect plant response near the HCLPF level or near the margin-review-earthquake level; in this case, their actual effect on 'plant seismic margin' is minor. This issue will require more discussion in the future.

10) There was further discussion of how the comparison between CDFM and fragilities methods for HCLPF determination will be accomplished. It is recognized that the EQE effort in this project will not extend to CDFM-fragilities methodological comparisons beyond those few presented to the Peer Review Group at this meeting. There is a continuing need for a more thorough way to address this issue, which the Peer Review Group will undoubtedly comment on further later on. It is likely that the NRC Expert Panel may need to give this issue much more careful thought after this Maine Yankee review has been completed.

11) John Reed asked a question of the NRC staff whose answer was not fully covered, since the NRC staff hazard experts were not present. The question concerned exactly what was intended by NRC staff in their choice of the margin review earthquake being used in the Maine Yankee analysis. After some discussion, it was decided that it would be assumed that the median NUREG/CR-0098 spectrum anchored to 0.30 g ZPA represented a uniform hazard spectrum at the 84 % confidence level. NRC staff will attempt to obtain some clarification on this issue soon.

12) James Thomas raised a general question of how systems interactions aspects should be dealt with in the analysis, such as when a valve may hit a support during strong earthquake motion. Some generic guidance is needed to provide analysts with an acceptable approach, which should be usable in most cases without doing a detailed and expensive calculation. This aspect was not resolved in the meeting, but will be given further thought.

OVERVIEW COMMENT:

It is the broad consensus of the Peer Review Group that the Maine Yankee trial margins review project is being accomplished so far with acceptable technical competence. In the course of its review, the Review Group discussed several technical issues that have been subjects of continuing difficulty in the analysis, and in some cases these difficulties may not be resolved during this trial margins review project. The Peer Review Group recognizes that one key objective of this trial review is to uncover such issues, especially methodological issues or issues of inadequate data. Its overview comment is that the analysis team, including

LLNL, EI International, and EQE experts, is carrying out a fully acceptable analysis so far, within the constraints of the project scope and subject to the comments in the detailed discussion above.

The Peer Review Group looks forward to further interactions as the analysis proceeds through a second walkdown and then to final analysis and documentation of the project results.

That completes this report and minutes.

Robert J. Budnitz
Chairman, Peer Review Group

9/30/86

ATTACHMENT A

PEER REVIEW MTG

NAME	POSITION	PHONE
P. SEARS	USNRC/NRR/PD8	301-492-8006
Don Guzy	NRC/RES	301-443-7894
Michael Bohn	Sandia Labs	505 846-4927
Sam Swan	EQE	415 495-5320
David Moore	EI International	206/854-0080
Philip Hashimoto	EQE Inc.	714-852-9299
M.K. Ravindra	EQE Inc	714-852-9299
R.C. Murray	LLNL	415-422-0308
R.P. KASSAWARA	EPRI	415 855-2775
J.E. Thomas	Duke Power	704 373-4612
AVTAR SINGH	EPRI	415-855-2384
Robert P. Kennedy	Struct. Mech. Consult.	714-777-2163
Stephen D. Evans	Maine Yankee Atomic Power	207-623-3521
Joe McCumber	Yankee Atomic Electric Co	617-872-8100
Bill Henries	"	"
John W Reed	JBA	(415) 969-8212
LORING A. WYLLIE	Degenkolb Assoc	415-392-6952
Greg Hardy	EQE Inc.	(714) 852-[illegible]
Robert Budnitz	FRA Inc	(415) 526-[illegible]
Garth Cummings	LLNL	(415) 422-4949
Newton Anderson	NRC	(301) 492-7306

AGENDA

Seismic Margins Program
Peer Review Group Meeting
San Francisco Airport Clarion Hotel
September 30, 1986

9:00 a.m.	Introductions	Budnitz/Murray
9:10 a.m.	Briefing on NRC Working Group Meeting	Guzy/Anderson
9:30 a.m.	Walkdown Summary	Ravindra
10:00 a.m.	Systems Summary	Moore
11:00 a.m.	Briefing on EPRI Program	Kassawara
12:00 noon	Lunch	
1:00 p.m.	Fragility Summary	
	HCLPF Comparisons	Hardy/Hashimoto
	Tank Methodology	Hashimoto
3:00 p.m.	Important Open Issues	
	Batteries	Hardy
	Internals/CRDM	Hardy
	Threaded Fire Water Piping	Hardy
5:00 p.m.	Report Outline	Murray
	Second Plant Walkdown	
	Schedule	
6:00 p.m.	Adjourn	

SEISMIC MARGINS PROGRAM

Tentative Schedule

September 23, 1986	Federal Express mailing to Peer Review Group
September 30, 1986	Peer Review Group meeting at San Francisco Airport Clarion Hotel
October 1, 1986	Analysis Team meeting at San Francisco Airport Clarion Hotel
October 22, 1986	Analysis Team meeting at Energy Incorporated, Kent, Washington
November 10, 1986	Second pre-walkdown status summary Federal Express package
November 17-19, 1986	Analysis Team second plant walkdown at Maine Yankee.
November 18-19, 1986	Peer Review Group meeting and additional walkdown at Maine Yankee
December 8, 1986	Preliminary plant HCLPF available
December 10-12, 1986	Possible exchange meeting with EPRI and Analysis Team meeting (most participants will be at the Symposium on Current Issues Related to Nuclear Power Plant Structures, Equipment and Piping to be held at North Carolina State, December 10-12, 1986).
~ January 1987	Briefing at the NRC on program and results
~ February 1987	Draft final report submitted to the NRC

Future Resources Associates, Inc.

2000 Center Street Suite 418 Berkeley, CA 94704 415-526-5111

5 December 1986

TO: U.S. Nuclear Regulatory Commission
D. Guzy, Office of Nuclear Regulatory Research
P. Sears, Office of Nuclear Reactor Regulation

Maine Yankee Atomic Power Plant
D. Whittier, Manager, Nuclear Engineering and Licensing

FROM: Robert J. Budnitz, Chairman of Peer Review Group for the Maine Yankee Seismic Margin Review Study

REF: MINUTES AND REPORT, THIRD PEER REVIEW GROUP MEETING

This is the report of the third meeting of the "Peer Review Group" that is reviewing the technical competence of the "Seismic Margin Review Study" that is being undertaken on the Maine Yankee reactor plant under sponsorship of the U.S. Nuclear Regulatory Commission.

The meeting was held at the Maine Yankee site on Tuesday and Wednesday, November 18-19, 1986. Attending besides the chairman were all four of the other Peer Review Group members: Michael Bohn; John Reed; James Thomas; and Loring Wyllie.

Also attending were representatives of the NRC; the Maine Yankee staff and their Yankee Atomic Electric associates; Lawrence Livermore National Laboratory staff; LLNL's subcontractors performing the review (EI International and EQE, Inc.); and R. Kennedy, a consultant to Maine Yankee. The sign-up sheet for the first day's session is attached as Attachment A. All attendees attended essentially the entire meeting. D. Whittier of Maine Yankee, whose name is not on the sign-up sheet, attended the second day's session.

Attachment B shows the agenda for the meeting, which lasted all day Tuesday and half of Wednesday. The Tuesday meeting began at 9:00 AM and ended at 5:30 PM. Wednesday's meeting began at 8:30 AM and ended about noon. The order of the agenda was followed closely, although some of the times were different. All agenda topics and presentations were given. Most of the meeting time was used for technical discussion of aspects of the on-going margins review study. In addition, on Tuesday afternoon all attendees participated in a walk-down of the plant to study particular aspects of the design relevant to its seismic margin.

The minutes of the meeting will be presented as numerically ordered topics, as follows, with commentary included:

1) R. Murray of LLNL led the introductory session, which consisted mainly of a discussion concerning the future schedule for the review effort. The schedule is judged to be satisfactory, in the sense that the project is currently 'on schedule' and the participants believe that it can remain on schedule. The following specific dates have been decided on for the schedule:

December 11, 1986: meeting between NRC margins team and EPRI margins study team at Raleigh, NC

January 15, 1987: first full draft report due from contractors, for limited distribution only to Peer Review Group, Maine Yankee, LLNL, 2 or 3 NRC staff

January 22, 1987: next meeting of Peer Review Group, in San Francisco; attendance by invitation of PRG only

February 12, 1987: second draft due, to be distributed to wider distribution including broadly in NRC

February 19, 1987: meeting in Washington with NRC in-house "Working Group on Seismic Margins".

Another Peer Review Group meeting is planned for sometime after the February meeting, to study the final version of the report to be prepared after the February comments are in.

Members of the Peer Review Group requested permission to visit the offices of the EQE and EI subcontractors toward the end of the current phase of the project, in order to review ongoing work prior to the completion of the subcontractors' reports. It was agreed that a visit to EI could occur any time after early December, but that visits to EQE should not be held until early January. A meeting on the fragilities is scheduled for January 6 at

EQE. The Peer Review Group wishes to emphasize the importance of obtaining access to various analyses and calculations as early as is feasible, to allow for optimum opportunity for review.

2) M. Ravindra of EQE, Inc. gave an overview presentation of the work of his group, who are the subcontractors doing the fragilities analysis of Maine Yankee. He presented slides that included preliminary tables on HCLPF values for various components and structures, which elicited much discussion from the Peer Review Group. The question of how best to document and present these results was covered in some detail.

The issue of how to treat dependencies was discussed, especially in regards to the CDFM approach to determining HCLPF values. (For the fragilities approach, the methodology for treating dependencies is more straightforward.)

Discussion about how to standardize the CDFM approach was lively. It was agreed that this issue, including comparison of the CDFM and FA approaches, would be a major topic in the Peer Review Group's final review work for the Maine Yankee project.

Some of this discussion covered specific items of equipment at Maine Yankee, which the Peer Review Group agreed would be concentrated on during their afternoon walkdown of the plant.

It was emphasized that the Peer Review Group will need access to all data that will be relied on in these evaluations, in order to review its applicability. This especially includes various aspects of the earthquake experience data base. In this regard, G. Hardy of EQE agreed to send the recent SQUG report on about 20 classes of equipment to J. Reed.

3) David Moore of EI International next presented an overview of the systems analysis work that his firm is doing. He discussed the preliminary system 'min cut sets' that have been developed, including a few interesting failure modes that are still being investigated in detail.

Specific technical topics covered in Moore's presentation included load sequencing, starting the turbine-driven emergency feedwater pump after an earthquake, and the reactivity-control systems.

The group discussed EI's process for pruning the trees during their development. Moore agreed that his team would publish the entire un-pruned trees, for later use, along with the pruned trees that will be used in the quantification process.

An extensive discussion occurred about how to combine non-seismic-induced failures with seismic-induced failures in the systems analysis. This discussion covered many aspects of this issue. The guidance given to the analysis team is as follows:

> The systems analysis team should identify all non-seismic failures that, in combination with a seismic failure, produce the undesired end-point (core damage) with a frequency above a certain cutoff. (The cutoff that EI is now using will include non-seismic failures whose contingent failure probability is above about 0.01 for failure of a single train, and above about 0.001 for failure of two trains or of a full safety function.) The key seismic contributors thus identified will then require determination of HCLPF values by the fragilities team. However, the analysis will <u>not</u> quantify an overall HCLPF plant value that includes these non-seismic failures. Instead, the combinations of seismic failures and non-seismic failures thus identified will be documented and discussed, so that decision-makers can be aware of their existence and the HCLPF values associated with the seismic failures identified. This documentation should be done in such a fashion that a complete HCLPF re-evaluation that includes the non-seismic failures is possible.

The EI group's main work in recent weeks has been development of the system fault trees, which were distributed in preliminary form to the Peer Review Group. EI's main work in the coming period will be to finalize these, based in part of the new information being gathered at the current walkdown.

4) An extensive discussion took place about how to analyze Maine Yankee's <u>batteries</u>. The issue came up because the fragilities team has been unable to develop any defensible technical basis for calculating fragilities for the current lead-antimony batteries. The MY plant will be changing out 2 of the 4 battery racks during their upcoming shutdown in March, 1987, and may change out the other 2 battery racks at a later shutdown in a future year. Therefore, the question arose as to which configuration should be analyzed in this study. The Peer Review Group's guidance is that the study team should analyze both the configuration in which only 2 of the 4 racks will be changed, and the ultimate configuration in which all 4 will be changed. If the

technical basis for developing fragilities for the existing batteries cannot be developed, this fact should be stated in the report.

5) Daniel Guzy of NRC gave a brief summary of the meeting the previous week of the NRC "Working Group on Seismic Margins". He discussed the Working Group's current thinking, including the selection of a BWR plant as the subject of a possible second trial margins review study. The NRC's current thinking is that they may decide to perform the planned BWR trial margin review in full collaboration with EPRI (same plant, same study).

6) The topic of how the margin review earthquake was selected by NRC was then discussed. John Reed agreed to draft a paragraph for these minutes on this subject, which paragraph is as follows:

> Dan Guzy discussed the question of what confidence and probability level the margin earthquake represented (i.e., NUREG-0098 rock median spectrum anchored to 0.3g peak ground acceleration). It was explained by Guzy that the NRC did not use probability concepts to establish the margin earthquake requirements, but rather adopted the NUREG-0098 spectrum shape to be consistent with a previous licensing decision made for Maine Yankee. John Reed requested as a minimum that the NRC agree that the margin earthquake input represents a uniform hazard spectrum over the entire frequency range. This understanding is consistent with the fragility method being used by the analysis team in developing HCLPF values for structures and equipment. In addition, it allows the NRC to make confidence and probability statements in the future after the margin analysis is completed, if they so desire.

7) The Peer Review Group went on an extensive walkdown of the plant on the afternoon of Tuesday, November 18. Items walked down were those that were of special interest to the reviewers, based on either their importance to the Maine Yankee margins analysis or their intrinsic methodological interest. After this walkdown, the Peer Reviewers and the analysis team reconvened to discuss what was observed. Interaction with the Maine Yankee staff and their consultant (R. Kennedy) was important during this session.

OVERVIEW COMMENTS:

1) The main overview comment here is identical to that made after the last Peer Review Group meeting. Specifically, it is the broad consensus of the Peer Review Group that the Maine Yankee trial margins review project is being accomplished so far with acceptable technical competence. Numerous technical and methodological issues have come up, and all have been coped with well by the analysis team. The cooperation of Maine Yankee and Yankee Atomic personnel has been excellent so far.

2) Although PRA-type information is being developed during the trial margins review, the Peer Review Group wishes to emphasize that it is probably not usable for full-scope PRA analysis per se without significant additional effort.

3) The Peer Review Group also wishes to emphasize that it cannot be expected to endorse the results of the trial Maine Yankee review. If all goes well, the Peer Review Group should find itself able to endorse the methodology used, and should also be able to comment on some specific technical aspects of the study. However, an endorsement of the study results would require a level of review effort much greater than can be expected from our PRG.

To summarize, the Peer Review Group looks forward to further interactions as the project proceeds to final analysis and documentation of the project results.

That completes this report and minutes.

Robert J. Budnitz
Chairman, Peer Review Group

Attachments: A = sign-up sheet
B = meeting agenda

ATTENDANCE, SIGN-UP SHEET 11/18/86

AT MAINE YANKEE

NAME	AFFILIATION	TELEPHONE
Robert Budnitz	FRA, Inc.	(415) 526-5111
GARTH CUMMINGS	LLNL	(415) 422-4949
Michael P. Bohn	Sandia Nat'l Lab	(505) 846-4927
Pete Prassinos	LLNL	(415) 423-6758
M.K. Ravindra	EQE	(714) 852-9299
Don Guzy	NRC	301 443-7894
Andrew Murphy	NRC	301-443-7710
S D EVANS	MAINE YANKEE	207-623-3521
Joe McCumber	Yankee Atomic	617-872-8100
Robert Kennedy	Struct. Mech. Consult.	714-777-2163
John W Reed	Jack R. Benjamin & Assoc.	(415) 969-8212
Loring A. Wyllie	HJ Degenkolb Assoc	415-392-6952
William Henries	Yankee Atomic	617-872-8100
David Moore	Energy Inc	206/854-0090
Jim Thomas	Duke Power Co.	704-373-4612
Bob MURRAY	LLNL	415-422-0308
PATRICK SEARS	NRC/NRR	301-492-8006
NILESH C. CHOKSHI	NRC/NRR	301-492-8347
Greg Hardy	EQE	714-852-9299
MICHAEL J. GRIFFIN	EQE	714 852-9299
Phil Hashimoto	EQE	714 852-9299
DAN MAGNARELLI	YANKEE ATOMIC	617 872-8100

AGENDA

Peer Review Group Meeting
November 18-19, 1986
Maine Yankee

<u>Tuesday, November 18, 1986</u>

9:00 a.m.	Introduction
9:15 a.m.	Status of Fragility Evaluation
10:00 a.m.	Status of Systems Analysis
10:45 a.m.	Open Items
11:00 a.m.	Peer Review Group Discussion
12:00 noon	Lunch
1:00 p.m.	Plant Tours and Discussions with Plant Personnel
5:00 p.m.	Adjourn

<u>Wednesday, November 19, 1986</u>

8:00 a.m.	Plant Tours and Discussions
10:00 a.m.	Peer Review Group Wrap-up Meeting
12:00 noon	Adjourn

Future Resources Associates, Inc.

2000 Center Street Suite 418 Berkeley, CA 94704 415-526-5111

27 February 1987

TO: U.S. Nuclear Regulatory Commission
D. Guzy, Office of Nuclear Regulatory Research
P. Sears, Office of Nuclear Reactor Regulation

Maine Yankee Atomic Power Plant
D. Whittier, Manager, Nuclear Engineering and Licensing

FROM: Robert J. Budnitz, Chairman of the Peer Review Group

REF: FINAL REPORT: PEER REVIEW GROUP for the MAINE YANKEE TRIAL REVIEW OF THE NRC SEISMIC MARGINS METHODOLOGY

This is the final report of the "Peer Review Group" that is reviewing the technical competence of the "Seismic Margin Review Study" that is being undertaken on the Maine Yankee reactor plant under sponsorship of the U.S. Nuclear Regulatory Commission.

The Peer Review Group has five members:

Robert J. Budnitz (chairman)
Michael P. Bohn
John W. Reed
James Thomas
Loring Wyllie, Jr.

The PRG is pleased to report that the five members are all in agreement with this report, and all endorse it fully.

I. Peer Review Group Charter

At its first meeting on July 21, 1986 the Peer Review Group adopted the following charter, which was also agreed to by all project participants, including NRC and the utility participants:

> The objective of the Peer Review Group is to assure that the trial seismic margins review, following the guidance established in NUREG/CR-4334 and NUREG/CR-4482, is executed in a fully competent and professional manner, uses methods that are at the state-of-the-art, and takes cognizance of all relevant information. The sponsors of the study (Lawrence Livermore National Laboratory for the NRC and Maine Yankee as the plant owner) desire to utilize the results of this study, and require the Peer Review Group's assurance that the study is technically sound.
>
> To accomplish its objective, the Peer Review Group will be provided full access to all materials, information, and methodologies that are inputs to and used by the study team. Access to the study team itself will occur through scheduled meetings to follow the study's progress. The Peer Review Group will also review draft reports and participate in walkdowns of the plant. A formal report for the Peer Review Group will be made by the Group's chairman to NRC and Maine Yankee. It is understood that the Peer Review Group's report will be a public document.

II. How the Review Was Accomplished

The Peer Review Group met on five occasions during the course of this project. Attendance was perfect except that at the first meeting one member could not attend and another's attendance was cut short. The five meetings were:

1. July 21-22-23, 1986, at the Maine Yankee site. During this meeting the PRG had the opportunity to walk down the plant, and to meet with all project participants. During this meeting the PRG's charter was discussed and clarified.

2. September 30, 1986, near San Francisco Airport. At this meeting the PRG received presentations and documentation from the project participants, and asked questions to clarify technical issues.

3. November 18-19, 1986, at the Maine Yankee site. During this meeting the PRG had a second opportunity to walk down the plant, to meet with all project participants, to review documentation, and to ask questions.

4. January 22, 1987, at Loring Wyllie's office in San Francisco. During this meeting the PRG met alone with no other attendees, discussed its conclusions about the study, discussed an early draft version of the

project report, formulated verbal comments on this early draft that were then relayed to the project team, and laid out the work needed for the PRG to arrive at its final conclusions.

5. February 19, 1987, at NRC's Bethesda offices. The PRG met alone briefly in the morning, spent the main part of this day in a large meeting with the entire team of project participants as well as NRC and utility representatives, and then met again at the end of the day to finalize its comments as embodied in this letter.

Besides these meetings, each of the PRG members had access to considerable information during the course of the project. Technical material was distributed at each of the first three meetings by the project participants (including material from Lawrence Livermore National Laboratory, from their subcontractors at Energy Incorporated and EQE Incorporated, and from the utility). This material enabled the PRG to remain fully informed and up-to-date on project progress, technical problems encountered, and proposed resolutions of various issues.

In addition, PRG members met individually on a few occasions with project participants to allow interaction on technical issues, and of course the participation of PRG members in both walkdowns of Maine Yankee provided hands-on interactions with the project.

Copies of the first version of the project's draft final report were distributed to the PRG only six days prior to its fourth meeting. About a month later, copies of the second version of the final report were distributed to the PRG only 3 to 6 days prior to its fifth and final meeting. In both cases, there was not enough time for the PRG to undertake as thorough a review as would have been possible if more time had been available. However, the PRG believes that there was enough time to allow the PRG to reach conclusions on each subject in its charter. This is primarily due to the opportunity for PRG interaction with the project team extensively throughout the project, including in the last several weeks by telephone, personal interaction, and study of preliminary material.

III. PRG Findings

The PRG charter requires findings on several topics. These findings form the substance of our review, and are presented here through discussion of the following five topics:

1) The PRG's access to technical information and to the project team been fully adequate to allow it to reach useful conclusions on each topic in the charter. The PRG members believe that their interaction with the project team has been significantly more extensive than is typical of reviews of this kind. Access to technical information has been complete and timely, within the obvious constraints of a very tight schedule toward the end of the project.

2) <u>The project team followed the guidance established in NUREG/CR-4334 and NUREG/CR-4482 as fully as could be expected.</u> In several areas, the guidance provided in the two NUREG reports was found to be incomplete or inadequate, and in each case the PRG believes that the project team successfully overcame the specific issues involved. We are grateful that the PRG was consulted on the major issues, a few of which are discussed below.

3) <u>The project team executed the study in a fully competent and professional manner.</u> The PRG believes that the professional competence of the project team is outstanding, and that the execution of the project has also been outstanding. This is all the more remarkable when one considers the various constraints on the project (such as the fact that this has been a research project which was unfortunately carried out as an NRC licensing action, which led to the problem that various uncertain researchable questions took on a licensing and therefore a financial aspect).

There were numerous issues within the project about which professional disagreements occurred between various participants, or between participants and the PRG. All were resolved in a fully professional manner, with open discussion and honest effort to find the best approach.

4) <u>The project team used state-of-the-art methods.</u> In fact, during the project a few advances were made in the state-of-the-art. Among them were the approach taken to combining seismic and non-seismic failures in the systems analysis; the issue of how to reconcile the CDFM and fragilities approaches to calculating the HCLPF values of components; and the way the systems-analysis team was able to use pruned fault trees to develop PRA-type systems cut sets with considerable savings of effort.

5) <u>The project team seems to have taken cognizance of all relevant information.</u> In fact, much information not previously published, or available only in draft or incomplete form, was used by both the systems analysts and the fragilities analysts. Information from the Maine Yankee and Yankee Atomic Electric contacts was well used.

<u>IV. Other PRG Comments</u>

A number of technical issues arose in the course of the project that deserve special comment by the Peer Review Group. These topics are the following, discussed in the next paragraphs:

A. selection of the review level earthquake
B. combining seismic and non-seismic failures
C. CDFM method vs. fragilities method for HCLPF analysis
D. level of expertise required by a margins review team
E. earthquake experience and test data base
F. treatment of relay chatter
G. correlations among earthquake-induced failures
H. comments on the screening methods used.

A. Review level earthquake: Throughout the Maine Yankee trial review, there has been confusion as to the precise interpretation of the review-level earthquake. The PRG believes that the confusion has now been cleared up (in large part through PRG interaction with the project team and with NRC), and that the analysis performed for Maine Yankee has correctly utilized a consistent review-level earthquake. However, the PRG believes it essential that more explicit guidance be provided as to what technical features must be established for the review level earthquake, and how it is to be used. (Presumably, the provision of this explicit guidance should be a task for the NRC Expert Panel, after interaction with the NRC staff and after seeking advice from industry representatives.) A second comment on this subject, which has been given verbally to the study team, is that the write-up in the draft report version we have studied in mid-February is still inadequate in its description of both how the review level earthquake for Maine Yankee is characterized and how it was used.

B. Combining seismic and non-seismic failures. The methodology used in the Maine Yankee review for combining seismic and non-seismic failures is not a rigorous methodology, in the sense that the final plant-level HCLPF value for Maine Yankee depends on factors that are not completely specified in the guidance. For the Maine Yankee review project, the final HCLPF values for both LOCA and non-LOCA cases have essentially no dependence on this inconsistency, but in a general case there can be differences. The PRG urges that the methodology for this aspect of seismic margins reviews be developed fully (presumably, again, by the NRC Expert Panel) and documented for general use. It is important that the way NRC will use the results of a margins review be explicitly considered in developing guidance on this issue. It is also important in this regard to note that there needs to be clarification of how a "seismic margin" is to be interpreted in terms of being greater than the SSE level, especially if non-seismic failures affect the HCLPF value but, of course, do not affect the SSE level to which the HCLPF value could in some minds be compared.

C. CDFM vs. fragilities method for HCLPF analysis. The PRG endorses the recommendation in the Maine Yankee report that further research is needed to develop the CDFM (conservative deterministic failure method) approach to analyzing HCLPF values, so that it can be applied routinely. We have learned much from the comparisons in the current project between HCLPF values calculated for a few components by both the CDFM and fragilities approaches. In particular, we have learned that the CDFM method as currently set down has too much latitude for the analyst and therefore does not provide robust HCLPF values. We urge that this situation be resolved by further research.

D. Level of expertise needed. Based on our interactions with the Maine Yankee trial review, the PRG believes that a review team must have certain attributes in order to carry out a successful seismic margins review. A utility whose choice of a review team does not take these attributes into account will be less than fully successful in applying the methodology. The main attribute for the fragilities analysis team is that it must have experience in plant walkdowns, in how to focus on the critical components, and in realistically analyzing seismic capacities; otherwise the margins analysis will be burdened by unnecessarily conservative results beyond those already embodied in the HCLPF approach. The systems analysis team must be familiar with PRA methods, and will be substantially strengthened by prior experience in PRA analysis of external initiators. Of course, it is not necessary for the analysis team to possess expertise at the level that was needed to develop the margins methodology in the first place.

E. Earthquake experience and test data base. In the current Maine Yankee review, considerable reliance has been placed on the earthquake experience and test data base for various components. The draft report we have reviewed contains citations of much of this data base. Even though some members of the review team are among those who have developed the data base, its interpretation for the purposes of this study has been problematical in a few situations because some of the data base is not yet publicly available. For other analysis teams in the future, this issue will be also exist. Therefore, it is essential in future studies that the data base relied on be thoroughly documented, and that its interpretation by the analysis team be set down in detail in future written reports. Guidance to future analysts along these lines must be strengthened (presumably, again, by the Expert Panel) to assure that future reviews are carried out acceptably.
In addition to the documentation aspect, it is important that the guidance be more explicit regarding the type and configuration details for components covered by the screening guidance, because some analysts in future routine applications may not be as familiar with the underlying raw experience data as were the analysts performing this Maine Yankee trial review.

F. Treatment of relay chatter. Following the guidance of NUREG/CR-4334, the issue of relay chatter and its recovery by operating crews has not been covered in the Maine Yankee review. The PRG wishes to point out that considerable progress in understanding relay chatter has recently occurred, and that the guidance for future analysts may be able to be strengthened in this aspect. (This is presumably another task for the NRC Expert Panel, with input from electrical circuitry and systems analysis experts).

G. Correlations among earthquake-induced failures. The handling of correlations among earthquake-induced failures is a methodological problem. These correlations can exist in several areas, including the fragilities, the responses, and the systems aspects. Although there are methodological weaknesses in the approach that was used, they do not affect the plant-level HCLPF results at Maine Yankee, and the review team's treatment is therefore acceptable. However, there remains a general methodological question as to how these correlations are to be treated, which the current guidance in NUREG/CR-4334 and 4482 does not cover well. The problem is that, while there are exact methods for handling these correlations in a full-scope and realistic seismic PRA study, their treatment in the approximate HCLPF analysis requires some explicit guidance not yet available. This area is in need of methodological development before the HCLPF methodology can be considered fully acceptable. In addition, the guidance needs to be more explicit concerning when correlations should be assumed to be 100 % --- and hence treated implicitly in the basic event definitions in the fault trees--- and when they should be treated explicitly in the numerical evaluations of the cut sets.

H. Comments on the screening methods used. The PRG has carefully studied the methods used by the Maine Yankee analysis team in its several screening tasks during the study. (This has been one of the most important reasons why PRG interaction during this trial review has been so valuable.) The approach taken, which employed an iterative process between systems-based screening and fragility-based screening, seems to have been very successful in reducing the amount of analysis work required while retaining the key items for final study. The PRG wishes to affirm its endorsement of this iterative and interactive approach. It also wishes to endorse the original guidance (in NUREG/CR-4334 and 4482) on how important a walkdown-type screening step is even for those components that are thought be screenable out based on generic guidance in the NUREGs. Only by such walkdown work (which is not necessarily a 100 % walkdown of all components) can there be high assurance that specific components do in fact fit within the generic categories for which the generic guidance is applicable.

I. Additional comments on the screening guidance: There are a few other areas where the Peer Review Group believes that the screening guidance should be improved. Among these are the approach to screening or analyzing batteries, especially aged batteries; the approach to walking down and analyzing fragilities for small LOCAs, in recognition of the fact that a thorough walkdown of all small lines in the primary pressure boundary may be prohibitively difficult, so that a screening approach may be needed; the treatment and documentation of system successes in the event trees; and how to treat component interfaces and 'super-components' such as multiple items on a single skid.

V. Level of Peer Review Group Effort

The PRG has been asked to comment on whether the level of review effort expended by the Group has been adequate; and more generally on how much effort should be devoted to peer review of seismic margins studies in the future.

1) On the first issue, the PRG is unanimous in believing that the peer review effort expended on this trial review has been a necessary part of accomplishing the project's objectives effectively. This project was exploring a new methodology, determining what are its strengths and weaknesses, and performing the research work in the context of a licensing action. This combination of ingredients would have been much more difficult to accomplish well without the continuing interactions that the project team had with our

Peer Review Group. These continuing interactions allowed for the timely resolution of several technical issues that arose during the project. If there had been no PRG, or if the peer review activity had only been a review of the final report once completed, we believe that the success of this project in accomplishing all of its objectives would probably not have been as outstanding. We also believe that all of the project participants agree with this conclusion.

2) On the second issue (concerning peer review of seismic margin studies in the future), our PRG members are divided. Here we are considering the situation in which the margin review methodology has become relatively "standardized" and a review is being undertaken by a utility, on either its own initiative or NRC's initiative.

(i) Some PRG members believe that carrying out a successful seismic margin study requires the interaction of a group of outside reviewers, experts in the disciplines involved, to provide technical review as a way of assuring the validity of the study results. The rationale for this position is that it will be difficult for a utility to perform such a review competently, especially for routine applications of the methodology in which review teams are composed of engineers who may not be truly expert already in the technical aspects of these subjects.

(ii) Other members of the PRG believe that, once the methodology has been developed into a more-or-less standard methodology, the necessity for peer review will be no different than it is for any other engineering study performed by utilities on their plants. In this view, the internal review procedures that utilities already use to assure the technical validity of their internally-supported analyses should be just as adequate in this arena as they are in the numerous other engineering arenas already part of reactor safety analysis.

(iii) There is yet a third view among the PRG members that can be summarized as follows: outside peer review is not an essential attribute of any routine utility-supported seismic margins study, but its presence will surely enhance the study's credibility with NRC, and hence the NRC's need for its own separate review may be less. Therefore, outside peer review should be considered carefully by any sponsoring utility, especially if the margins review team itself is not fully familiar with all of the technical aspects of performing these studies.

This division of views among the PRG members on the need for routine outside peer review does not conflict with the unanimity of views on the usefulness of the peer review exercise for the current project, which was a trial (research) project. In particular, the value of ongoing interactions throughout he course of the trial project cannot be overemphasized.

VI. Final Remarks

In conclusion, it is our pleasure to acknowledge the outstanding cooperation and assistance of all parties who participated in this trial margins review. The PRG found excellent cooperation from all five parties to the review effort: the NRC staff; the utility team from Maine Yankee and Yankee Atomic Electric; the Lawrence Livermore staff; and LLNL's subcontractor teams from EQE Inc. and Energy Incorporated. The PRG believes not only that a project like this should be technically competent (it was) and timely within schedule and budget (it was), but also enjoyable for the participants, including both project team and reviewers. For making this last aspect come true, we thank one and all.

Robert J. Budnitz
Chairman, Peer Review Group

APPENDIX C

UTILITY COMMENTS

MAINE YANKEE ATOMIC POWER COMPANY
ENGINEERING OFFICE

1671 WORCESTER ROAD
FRAMINGHAM, MASSACHUSETTS 01701
617-872-8100

MAINE YANKEE SEISMIC DESIGN MARGIN PROGRAM

UTILITY PERSPECTIVE

Prepared By: W. E. Henries
Lead Mechanical Engineer

J. T. McCumber
Project Engineer

December 1986

I. PURPOSE

This report provides a description of the Utility role in a Seismic Design Margins Program. This description will assist potential generic implementation by documenting our experience with the methods utilized, costs incurred, and impact on normal Utility operations.

II. BACKGROUND

Maine Yankee has been involved with the issue of demonstrating seismic margins for several years, with particular focus generated by the 1982 New Brunswick earthquake.

In addition to identifying the inherent conservatism in Maine Yankee's seismic design, a series of voluntary reviews and walkdowns were initiated in 1982 to find any potential weak areas. These plant reviews were performed by known experts in the fields of structural and seismic design. Several cost beneficial upgrades, primarily associated with equipment anchorage, were identified which if made would enhance seismic adequacy. The equipment anchorage and supports upgraded by Maine Yankee are listed in Table 1.

Maine Yankee and Yankee Atomic Electric Company have been very active in the efforts of the Seismic Qualification Utility Group (SQUG). The equipment at Maine Yankee has been favorably compared to the SQUG experience data base as it was developed.

When assessing the applicability of the Maine Yankee experience to the generic Seismic Design Margins Program, it should be recognized that Maine Yankee may not represent the typical state of seismic adequacy for a plant of its age. This is due to upgrades associated with the seismic review and enhancement program described above.

III. INFORMATION RETRIEVAL/ANALYSIS SUPPORT

The primary role of the Utility in the Seismic Design Margins Program was to provide a complete and accurate description of the operation and construction of the Maine Yankee plant. It was first necessary to provide a description of the safety-related systems, their functions, and interdependencies. The Analysis Team (AT) could then properly categorize them as "Group A" or "Not Group A" Systems (Reference 1). In addition to the systems information needed for the Group A screening, it was also necessary to provide seismic design details to the AT to aid in their seismic capacity determinations. A summary of the information provided to the AT is contained in Table 2.

Once the AT systems modeling and fragilities analysis began, the Utility role evolved into a question response mode. Specific system operation or equipment anchorage/design questions were forwarded to the Utility and the requested answers or design details were returned to the AT. Many of the questions were readily answered but, occasionally, specific details were not available and required more extensive review. Examples of the latter included reactor internals design details and anchorage details for original plant equipment which was not within the scope of the previous seismic review programs.

In order to obtain the necessary reactor internals and control rod drive information, a meeting with the NSSS vendor was arranged and attended by representatives of both the Utility and the AT. The meeting was a success, and the information required to perform the fragility analyses was obtained. Similarly, when original installation information was not readily available, searches through microfilm archives, or actual field measurements were required to answer the AT questions. The availability of data is a major factor in determining the Utility workload and associated costs.

The final role played by the Utility was to review the systems models and assumptions developed by the AT. This effort involves close scrutiny during the entire development process to insure the accuracy of the model and its assumptions. It is important to prevent misconceptions or easily upgraded "weak links" from entering into the models.

Following this approach, Maine Yankee agreed to implement post-earthquake procedural guidance and to make minor anchorage upgrades to increase the confidence level of the system models and equipment fragilities.

The total Utility effort was accomplished without major impact to normal operations. As discussed in Section V, slightly more than 1,000 man-hours were dedicated to the program over a ten-month period. Given advance planning and an absence of a refueling outage, permitted the engineering hours shown on Table 4 to be provided by the existing engineering personnel.

IV. PLANT WALKDOWNS

General

The Seismic Design Margins Program included two walkdowns of the Maine Yankee plant. The first walkdown, lasting a week, was performed early in the program to obtain:

1. A firsthand understanding of Maine Yankee's structures, systems, and equipment,

2. Equipment anchorage details, and

3. Assurance that plant-specific problem areas or seismic interactions were not a concern.

The final plant walkdown, lasting three days, was performed near the end of the program following the completion of preliminary fault tree and fragility analyses. The objectives of the final plant walkdown were to:

1. Gather additional information on systems and equipment needed to complete the analyses,

2. Obtain equipment operating and maintenance information,

3. Insure the validity of system interaction assumptions, and

4. Resolve open issues.

The walkdowns were each scheduled in conjunction with Peer Review Group meetings, so in addition to the AT, the walkdowns included the Peer Review Group (PRG), NRC, and Lawrence Livermore (LLNL) personnel. The specific interests of each group had to be carefully considered in the walkdown planning.

Walkdown Planning

In addition to the actual walkdown activities, PRG meetings, plant access training, body counting, getting dosimetry, and miscellaneous information gathering meetings had to be scheduled within the walkdown time period. Considering all of the above, a productive walkdown required careful preparation and planning.

Coordination with the AT well before the actual walkdowns was necessary to determine the areas of the plant and specific equipment of interest. This advanced notice allowed the Utility Walkdown Coordinator to familiarize himself and other escorts with the exact areas of the plant to be visited, as well as the system's function and anchorage details of the equipment to be inspected. Pre-walkdowns, marked-up drawings and equipment lists were all helpful tools in assuring efficient use of the limited time available.

Other preparations which facilitated plant walkdown activities included:

1. Sending out all of the required health physics forms with instructions to allow walkdown members to arrive at the site with the necessary paperwork completed.

2. Scheduling body counting during the PRG meetings (approximately 15 minutes for each person - took turns leaving the meeting).

3. Notifying key plant personnel of the walkdown schedule and assuring their availability, as needed,

4. Reserving meeting rooms and work rooms, and

5. Obtaining necessary approvals for camera passes.

Walkdown Logistics

The plant walkdowns were made up of a large group of people (around 20), of varied interests, and conducted in a limited amount of time in a limited access environment. Therefore, complex logistics were to be expected.

Plant security regulations limited the number of unbadged people which could be escorted throughout the plant by one escort to five. This meant that four or more prepared escorts were necessary at any time. Also, it was important to match the escort's expertise to the group; i.e., systems engineer for a group seeking plant operations information and a structural/mechanical engineer for a group looking at anchorages.

The objectives of the walkdown members are generalized below:

Analysis Team	- Gather data on equipment throughout the plant. - Obtain information from plant operations, engineering, and maintenance personnel.
Peer Review Group	- Obtain a good overview of the plant. - Inspect equipment of specific interest to them. - Knowledge of AT activities.
NRC and LLNL	- Obtain general overview of plant. - General knowledge of AT activities (programmatic).

A preplanned tour of the plant was arranged at the beginning of the initial walkdown for all of the walkdown members. The list of plant areas and equipment which the AT wanted to see was used to develop the tour. This assured that everyone involved received a good overview of the plant and had a chance to inspect the equipment of interest.

Subsequent plant tours were arranged to facilitate the specific information which had to be gathered. During the initial walkdown, the PRG could only stay three days, so an effort was made to meet their plant walkdown needs during that time. Otherwise, the priority was to assure that the AT obtained information needed.

Meetings were scheduled each evening with all walkdown participants to discuss progress and potential problem areas. Also discussed, was the schedule of the next days activities and information which may be needed.

This allowed time for preparation that night, if required. These meetings were extremely productive, and while they made for a long day, are highly recommended.

Plant Impact

As a result of the preplanning associated with the walkdowns, the impact on normal plant operations was minimized. Additional steps which were taken to lessen the impact included:

1. Staggering the time which various walkdown groups pass through HP to minimize their processing time and to avoid congestion in the change area.

2. Keeping the number of people in the Control Room to a minimum, preferably doing Control Room work during the evening shift.

3. Using simulator and simulator staff to answer operations questions where possible.

4. Letting operations or security personnel know in advance of locked areas which must be accessed.

5. Letting HP personnel know in advance when access to a high radiation area is required so they may arrange proper support.

6. Having radiation work permits made out in advance.

V. COST

The Utility costs to support the Seismic Design Margins Program are summarized in Tables 3 and 4. Several observations regarding these costs are worth noting.

First, the majority of man-hours were spent by Mechanical Engineering since that discipline has historically been responsible for seismic review programs at Maine Yankee; thus, the bulk of data retrieval and program management fell to them. Systems Engineering also required considerable man-hours. Their input was vital in identifying the required systems and components and assuring plant unique operational features were accurately modeled. The Licensing man-hours and Consultant costs are attributable to the fact that this pilot research program also had licensing implications for the Utility.

Secondly, the relatively high travel costs and miscellaneous expenses are attributable to the fact that the AT consultants are all from the West Coast, while the subject plant is on the East. Additionally, the tight program schedule requirements required frequent overnight mailings and phone calls.

Finally, the total program effort may vary for other utilities for two reasons. Reduction in the data retrieval time for the Seismic Design Margin Program was greatly aided by the walkdown documentation and equipment anchorage upgrades previously performed at Maine Yankee. Since 1982, three separate plant walkdowns have been conducted at Maine Yankee as part of its voluntary seismic review program. As a result of these walkdowns, many equipment anchorages and masonry walls have been strengthened beyond FSAR requirements. These upgrades, and the relatively recent design analyses, allowed easy information retrieval for the Utility and fragility screening for the AT. On the other hand, the time involved with retrieving original design data for those components not within the scope of the previous walkdowns was significant. The inability to readily retrieve old (16 to 18 years) design records or to obtain as-built details due to radiation or operational constraints prevented the Utility from being as responsive to AT requests as we would have liked. Other utilities should thus estimate their man-hours accordingly.

VI. IMPRESSIONS/RECOMMENDATIONS

The concept of using plant walkdowns and experience screening to evaluate seismic margins is excellent and long overdue. The value obtained by eliminating those components shown by experience not to be seismically fragile is enormous. Just as important, the walkdown methodology provides a thorough systems interaction review. This program is a major step in developing a cost effective method of identifying plant-specific seismic margins and, perhaps more important, easily fixed seismic "weak links."

While we praise the overall thrust of the program, several recommendations are provided below which may improve future reviews.

1. Separate the seismic review from the licensing area. This change will allow closer Utility/AT cooperation without the glare of the "public meeting" requirements.

2. Define the review earthquake with greater precision. The ongoing concerns regarding the confidence level of the ground response spectral shape and ZPA resulted in too much wasted effort.

3. Precisely define the need for long-term safe shutdown (Group B functions) capability. Considerable effort was expended on these components which we believed were excluded from the review scope per Reference 1.

4. Provide better field guidance on the use of the SQUG experience data base (i.e., MOV/AOV interaction, cable trays).

5. Develop improved methods of determining/defining "high confidence" levels for components such as yard tanks. As demonstrated by our experience, the significance of yard tank capacity is high and the analytical methods for determining actual capacity seem unnecessarily conservative.

6. Decide if nonseismic failures should be included in the HCLPF determination and, if so, how.

VII. REFERENCES

1. NUREG/CR-4482, "Recommendations to the NRC on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants," by P. G. Prassinos, M. K. Ravindra, and J. B. Savy, March 1986.

TABLE 1

Maine Yankee Equipment Anchorage and Support Upgrades

Emergency Buses 5, 6, 7, and 8

Battery Chargers 1, 2, 3, and 4

Battery Inverters 1, 2, 3, 4, and 5

Floor Mounted Diesel Generator Control Panels

Main Control Board

Control Room Auxiliary Cabinets

- Electrical Control Board
- Radiation Monitoring System
- Heat Tracing Cabinet
- Air Conditioning Control Panel
- EHC Panel
- Reactor Regulating System
- Feedwater Regulating System
- Vibration and Loose Parts Monitoring Cabinet
- Reactor Protective System
- Core Loading Panel
- Radiation Monitoring System
- Meteorological Survey Cabinet

Safety-Related Instrument Racks

Masonry Wall Reinforcement

Service Water Piping Support

TABLE 2

Summary of Information Provided for Maine Yankee Seismic Design Margins Review

- Maine Yankee FSAR
- Ground and Floor Response Spectra for 0.18g NUREG/CR-0098 an 0.2g Regulatory Guide 1.60 Earthquakes
- Approximately 15 Specifications
- Approximately 40 Calculations (or portions thereof)
- Approximately 550 Drawings
- Emergency Operating Procedures
- Pump NPSH Curves
- "Tagging" and Modification Control Procedures
- Maine Yankee Technical Specifications
- Maine Yankee Training Manuals
- Equipment Testing Frequencies and Procedures
- Abnormal Operating Procedure for Flooding of the Circulating Water Pump House
- Abnormal Operating Procedure for Earthquakes
- Reports on:
  - Class I Structures Dynamic Modeling
  - Components Required for Accident Mitigation and Safe Shutdown
  - Reactor Protective System
  - Appendix R Alternate Shutdown System
  - Maine Yankee Masonry Walls
  - 1983 Seismic Walkdown and Modifications
  - Seismic Review Program Summary
  - Maine Yankee Seismic Hazard Analysis (UHS)
  - Conservative Seismic Capacities of Maine Yankee's Reactor Containment
  - Evaluation of LOCA-Related Loadings on the Reactor Coolant System
  - Expansion Anchor Information and IEB 79-02 Anchor Bolt Testing

TABLE 3

Maine Yankee Seismic Margins Program Costs*

	Man-Hours	Dollars
Mechanical Engineering	585	
Systems Engineering	195	
I&C Engineering	15	
Electrical Engineering	45	
Nuclear Engineering**	75	
Licensing	100	
Plant Engineering/Operations	45	
Consultant (Includes Travel)		$21,000
Travel		11,400
Miscellaneous (Drawings, Xerox, Mail, Phone)		1,525
TOTAL***	1,060	$33,925

* Does not include cost of anchorage upgrades.
** Includes PRA review.
*** Does not include costs to review final report.

TABLE 4

Monthly Man-Hour Breakdown

Month	Man-Hours
May	67
June	50
July	170
August	154
September	88
October	159
November	171
December	66
January	100
February	35
TOTAL	1,060

MAINE YANKEE ATOMIC POWER COMPANY •

AUGUSTA, MAINE 04336
(207) 623-3521

February 26, 1987
MN-87-22

GDH-87-43

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Ashok C. Thadani, Director
PWR Project Directorate #8
Division of Licensing

References: (a) License No. DPR-36 (Docket No. 50-309)
(b) NRC to M.Y. letter dated
(c) M.Y. to NRC letter dated

Subject: Maine Yankee Seismic Margins Program

Gentlemen:

For the past eight months, Maine Yankee has been working with the NRC staff and its consultants on a seismic margins program. This program, was designed to demonstrate the feasibility of the margin assessment methodology, and to resolve NRC staff concerns about the adequacy of Maine Yankee's seismic design. The program has been completed and a draft report issued.

The program has demonstrated that the Maine Yankee plant possesses seismic ruggedness well beyond the plant's original design. The program also identified several areas where the seismic ruggedness of the plant could be effectively enhanced.

The changes that we have decided to make along with the scheduled completion dates are listed below.

Description of Change	Schedule Completion Date
• Diesel fuel day tank anchorage upgrade	done
• Control Room cooler anchorage upgrade	done
• Welding cart/gas bottle tiedown	done
• Security lighting tiedown	done
• Maine control board alarm tiedown	done
• Strengthen blockwall VE 21-1	1987 Outage
• Upgrade anchors for fans 44A & B	1987 Outage
• Install internal anchors for transformers 507 & 608	1987 Outage
• Replace safety class batteries 1 & 3	1987 Outage
• Replace safety class batteries 2 & 4	1988 Outage
• Refueling water storage tank anchorage upgrade	1988 Outage

8492L/GDW

United States Nuclear Regulatory Commission
Attention: Mr. Ashok C. Thadani, Director

The draft seismic margins report indicates that the Maine Yankee plant could withstand seismic events well in excess of the plant's original design basis without endangering public health and safety. The report also indicates that the limiting component is the refueling water storage tank (RWST) with a high confidence of low probability of failure (hclpf) of 0.21g. As indicated in the foregoing table, we plan to upgrade the seismic ruggedness of this tank, by improving its anchorage, no later then the 1988 refueling outage. We are making an effort to complete all or part of these upgrades during the 1987 refueling outage. Although the anchorage upgrades have not been finally designed, we expect to upgrade the tank's seismic ruggedness to a hclpf value in the range of 0.27g.

We would appreciate your concurrence that completion of these committed modifications resolves this long standing licensing concern so that we may apply our attention and resources to other safety enhancing efforts.

Very truly yours,

MAINE YANKEE ATOMIC POWER COMPANY

G D Whittier

G. D. Whittier, Manager
Nuclear Engineering and Licensing

GDW/bjp

Enclosure

cc: Dr. Thomas E. Murley
Mr. Pat Sears
Mr. Cornelius F. Holden

8492L/GDW

NRC FORM 335
(2-84)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE.

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG/CR-4826
UCID-20948
Vol. 1

2. TITLE AND SUBTITLE
Seismic Margin Review of the Maine Yankee Atomic Power Station
Vol. 1. Summary Report

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
March	1987

5. AUTHOR(S)
P. G. Prassinos, R. C. Murray, and G. E. Cummings

6. DATE REPORT ISSUED

MONTH	YEAR
March	1987

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Lawrence Livermore National Laboratory
P.O. Box 808, L-197
Livermore, CA 94550

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER
NRC FIN No. A-0461

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Engineering Safety
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a. TYPE OF REPORT
Technical

b. PERIOD COVERED (Inclusive dates)
June 1986 - March 1987

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This Summary Report is the first of three volumes for the Seismic Margin Review of the Maine Yankee Atomic Power Station. Volume is the Systems Analysis of the first trial seismic margin review. Volume 3 documents the results of the fragility screening for the review. The three volumes demonstrate how the seismic margin review guidance (NUREG/CR-4482) of the Nuclear Regulatory Commission (NRC) Seismic Design Margins Program can be applied.

The overall objectives of the trial review are to assess the seismic margins of a particular pressurized water reactor, and to test the adequacy of this review approach, quantification techniques, and guidelines for performing the review. Results from the trial review will be used to revise the seismic margin methodology and guidelines so that the NRC and industry can readily apply them to assess the inherent quantitative seismic capacity of nuclear power plants.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS
Seismic Margins

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page)
Unclassified
(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

☆ U. S. GOVERNMENT PRINTING OFFICE:1987- 131-682:60035

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE & FEES PAID
USNRC
WASH. D.C.
PERMIT No. G-67

NUREG/CR-4826
UCID-20948
Vol. 2

Seismic Margin Review of the Maine Yankee Atomic Power Station

Systems Analysis

Manuscript Completed: February 1987
Date Published: March 1987

Prepared by
D. L. Moore, D. M. Jones, M. D. Quilici, J. Young

Energy Incorporated
Kent, WA 98031

Under Contract to:
Lawrence Livermore National Laboratory
7000 East Avenue
Livermore, CA 94550

Prepared for
Division of Engineering Safety
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN A0461

ABSTRACT

This Systems Analysis is the second of three volumes for the Seismic Margin Review of the Maine Yankee Atomic Power Station. Volume 1 is the Summary Report of the first trial seismic margin review. Volume 3, Fragility Analysis, documents the results of the fragility screening for the review. The three volumes are part of the Seismic Margins Program initiated in 1984 by the Nuclear Regulatory Commission (NRC) to quantify seismic margins at nuclear power plants.

The overall objectives of the trial review are to assess the seismic margins of a particular pressurized water reactor, and to test the adequacy of this review approach, quantification techniques, and guidelines for performing the review. Results from the trial review will be used to revise the seismic margin methodology and guidelines so that the NRC and industry can readily apply them to assess the inherent quantitative seismic capacity of nuclear power plants.

TABLE OF CONTENTS

Section		Page
	ABSTRACT	iii
	LIST OF TABLES	vii
	LIST OF FIGURES	ix
1.	INTRODUCTION	1-1
2.	METHODOLOGY	2-1
	2.1 General Approach	2-1
	2.2 Step 2 - Initial Systems Review	2-1
	2.2.1 Gather System Information	2-1
	2.2.2 Classify Front-Line Systems	2-1
	2.2.3 Identify Group A Front-Line Components	2-2
	2.2.4 Classify Support Systems	2-2
	2.2.5 Identify Group A Support System Components	2-2
	2.2.6 Identify Plant-Unique Features	2-3
	2.2.7 Prepare for First Plant Walkdown	2-3
	2.3 Step 4 - First Plant Walkdown	2-3
	2.3.1 Peer Review Group Meeting	2-3
	2.3.2 Plant Walkdown	2-3
	2.3.3 Plant Staff Discussions	2-4
	2.4 Step 5 - Systems Modeling	2-4
	2.4.1 Review Event Trees	2-4
	2.4.2 Develop Fault Trees	2-5
	2.4.3 Develop Data Base	2-7
	2.4.4 Determine System Cut Sets	2-7
	2.5 Step 6 - Second Plant Walkdown	2-8
	2.6 Step 7 - Determine Minimal Cut Sets	2-8
	2.6.1 Finalize Event Trees	2-8
	2.6.2 Finalize Fault Trees	2-8
	2.6.3 Link Fault Trees	2-8
	2.6.4 Determine Preliminary Boolean Equation	2-9
	2.6.5 Determine Final Boolean Equation	2-9

TABLE OF CONTENTS (Cont'd)

Section			Page
3.	SYSTEMS ANALYSIS		3-1
	3.1	System Identification	3-1
	3.1.1	Group A Systems	3-1
	3.1.2	Systems Removed from Group A	3-2
	3.2	Systems Analysis of Front-Line Systems	3-3
	3.2.1	Auxiliary Feedwater System	3-3
	3.2.2	High Pressure Safety Injection System	3-5
	3.2.3	Power-Operated Relief Valves (Feed and Bleed)	3-8
	3.3	Systems Analysis of Support Systems	3-9
	3.3.1	Component Cooling Water Systems	3-9
	3.3.2	Service Water System	3-11
	3.3.3	Electric Power System	3-13
	3.3.4	Actuation Systems	3-16
	3.4	Probability Calculations	3-16
4.	ACCIDENT SEQUENCE ASSESSMENT AND RESULTS		4-1
	4.1	Event Trees	4-1
	4.1.1	No LOCA Case	4-1
	4.1.2	Small LOCA Case	4-2
	4.2	Core Damage Sequences	4-3
	4.2.1	T_1LP_1	4-3
	4.2.2	T_1LD	4-3
	4.2.3	S_2D	4-4
	4.2.4	S_2LP_2	4-4
	4.2.5	S_2LD	4-4
	4.3	Boolean Equations for No LOCA and LOCA Cases	4-4
	4.3.1	No LOCA Case	4-5
	4.3.2	Small LOCA Case	4-6
	4.4	Plant-Level Boolean Equation	4-6

TABLE OF CONTENTS (Cont'd)

5. ENGINEERING AND METHODOLOGY INSIGHTS 5-1

5.1 Engineering and Operational Insights 5-1

5.2 Insights on the Methodology and Execution 5-2

5.2.1 System Classification and Screening Guidelines 5-2

5.2.2 Preparation for Walkdowns and Documentation 5-4

5.2.3 Systems Analysis and Pruning Process 5-4

5.2.4 Minimal Cut Set Evaluation Process 5-5

5.2.5 Schedule and Resources 5-6

REFERENCES R-1

APPENDIX

A Identifiers and Symbols A-1

B Auxiliary Feedwater System B-1

C High Pressure Safety Injection System C-1

D Primary Pressure Relief System D-1

E Primary Component Cooling Water System E-1

F Secondary Component Cooling Water System F-1

G Service Water System G-1

H Electric Power System H-1

I Actuation Systems I-1

LIST OF TABLES

2-1	Definition of plant safety functions	2-10
3-1	Front-line system vs support system dependency matrix	3-19
3-2	Support system vs support system dependency matrix	3-20
3-3	AFW screening overview table	3-21
3-4	Standard footnotes for screening overview tables	3-22
3-5	AFW system cut sets	3-23
3-6	HPSI/CSPPCL screening overview table	3-24
3-7	HPSI system cut sets	3-26
3-8	PORVs screening overview table	3-27
3-9	PORV (no LOCA) cut sets	3-28
3-10	PORV (small LOCA) cut sets	3-29
3-11	PCC screening overview table	3-30
3-12	SCC screening overview table	3-32
3-13	SWS screening overview table	3-33
3-14	Electric power screening overview table	3-34
3-15	Nonseismic event failure probability calculations	3-37
4-1	Basic event descriptions	4-7
4-2	Sequence T_1LP_1 cut sets	4-8
4-3	Sequence T_1LD cut sets	4-9
4-4	Sequence S_2D cut sets	4-10
4-5	Sequence S_2LP_2 cut sets	4-11
4-6	Sequence S_2LD cut sets	4-12
4-7	Boolean equations for no LOCA and small LOCA accident sequences	4-13
5-1	Systems analysis resource expenditure	5-7
A-1	Fault tree symbols	A-1
A-2	Fault tree event identifiers	A-2

LIST OF TABLES (Cont'd)

B-1	Auxiliary feedwater (AFW)	B-1
B-2	AFW valve table	B-2
B-3	AFW cooling requirements	B-3
C-1	High pressure safety injection (HPSI) and containment spray pump area cooling (CSPPCL)	C-1
C-2	HPSI/CSPPCL valve table	C-3
C-3	HPSI cooling requirements	C-5
D-1	Primary pressure relief system (PPS)	D-1
E-1	Primary component cooling (PCC)	E-1
E-2	PCC valve table	E-2
E-3	PCC cooling requirements	E-3
E-4	PCC cooling loads	E-4
F-1	Secondary component cooling (SCC)	F-1
F-2	SCC valve table	F-2
F-3	SCC cooling requirements	F-3
F-4	SCC cooling loads	F-4
G-1	Service water system (SWS)	G-1
G-2	SWS cooling requirements	G-2
H-1	AC power	H-1
H-2	DC power	H-2
H-3	Diesel generators (DG)	H-3
H-4	DG cooling requirements	H-4
I-1	Actuation systems	I-1

LIST OF FIGURES

2-1 Graphic representation of the screening operations (Figure 2-6 from NUREG/CR-4482) 2-11

3-1 Auxiliary feedwater 3-39
3-2 High pressure safety injection 3-40
3-3 Primary pressure relief 3-41
3-4 Primary component cooling water 3-42
3-5 Secondary component cooling water 3-44
3-6 Service water system 3-46
3-7 AC power system 3-47
3-8 DC power system 3-48
3-9 Diesel generators 3-49

4-1 Seismic event, LOOP event tree 4-14
4-2 Seismic event, LOOP concurrent with small LOCA event tree 4-15

B-1 Auxiliary feedwater system fault tree B-4
B-2 Auxiliary feedwater system fault tree, pruned and merged B-9

C-1 High pressure safety injection system fault tree C-6
C-2 High pressure safety injection system fault tree, pruned and merged C-16

D-1 Power-operated relief valve fault tree D-2
D-2 Power-operated relief valve fault tree (no LOCA), pruned and merged D-4
D-3 Power-operated relief valve fault tree (small LOCA), pruned and merged D-7

E-1 Primary component cooling system fault tree E-6

LIST OF FIGURES (Cont'd)

F-1 Secondary component cooling system fault tree F-6

G-1 Service water system fault tree.................................. G-3

H-1 AC power (MCC 7A) fault tree H-5
H-2 AC power (MCC 8A) fault tree H-6
H-3 Diesel generator 1A fault tree................................... H-7

I-1 Safety injection actuation system (Train A) fault tree.. I-2

1. INTRODUCTION

This report documents the approach, results, and insights of the systems analysis tasks for the Seismic Margin Review of the Maine Yankee Atomic Power Station. This project is part of the Seismic Design Margins Program performed by Lawrence Livermore National Laboratory for the Nuclear Regulatory Commission. The purpose of a seismic margin review is to determine the capability of a plant to resist with high confidence an earthquake level greater than the design basis Safe Shutdown Earthquake (SSE). This seismic capacity is expressed as the earthquake acceleration level for which there is a high confidence of a low probability of failure (HCLPF value).

The objectives of this seismic margin review of the Maine Yankee Atomic Power Station are to:

- Provide an assessment of an actual plant's capability to withstand a specific earthquake level greater than the SSE.
- Demonstrate the use of the Expert Panel's approach (NUREG/CR-4334) and guidelines (NUREG/CR-4482) for seismic margin review.
- Provide a basis for upgrading the Expert Panel's approach and guidelines, if improvements are needed.
- Provide a benchmark for possible future seismic margin reviews, including cost and time resources.

The overall approach for this seismic margin review, as detailed in NUREG/CR-4482 (Prassinos et al., 1986), was to:

- Select a review level earthquake.
- Determine the systems, components, and structures that are important to plant seismic capacity by applying a successive set of screening filters based on system function, nonseismic unavailability, and seismic fragility.
- Use plant walkdowns to gather information for the screening process, and for the determination of fragilities and plant seismic capacity.
- Model the system and plant response to the review level earthquake through fault trees and event trees, and develop a Boolean equation for the endpoint of core damage.
- Calculate the plant seismic capacity (HCLPF) using the Boolean equation and fragility information.

This report is Volume 2 of a three-volume report documenting the project results. Volume 1 provides the overall project results, and Volume 3 provides the fragility and HCLPF analyses. Volume 2 provides the systems screening analysis, the logic models of the plant and systems, and the plant-level Boolean equation for core damage. Chapter 2 describes the methodology used in these systems analysis tasks. Chapter 3 provides the results of the screening assessment and discusses each of the systems, including the detailed system fault tree models. Chapter 4 presents the plant event trees, and the Boolean equations. The resulting minimal cut sets are also discussed. Comments and conclusions on the methodology and seismic margin review process are presented in Chapter 5. References are given at the end of the report. The appendices contain definitions of abbreviations and symbols for the fault trees and basic events, and the actual system fault trees.

2. METHODOLOGY

2.1 General Approach

The approach for the Maine Yankee seismic margins review followed the same eight steps outlined in Chapter 2 of NUREG/CR-4482 (Prassinos et al., 1986), which are graphically represented in Figure 2-1 (Figure 2-6 of NUREG/CR-4482). The systems analysis tasks involved Steps 2, 4, 5, 6, and 7. The extensive guidance on these steps in NUREG/CR-4482 will not be repeated here. Rather, this discussion will focus on the plant specific steps and considerations for the Maine Yankee review and any departure from the methodology described in NUREG/CR-4482.

2.2 Step 2 - Initial Systems Review

The initial systems review consisted of seven tasks:

- Gather system information,
- Classify front-line systems (Group A or Group NOT-A),
- Identify Group A front-line system components,
- Classify support systems (Group A or Group NOT-A),
- Identify Group A support system components,
- Identify any Group A plant unique features,
- Prepare for first plant walkdown.

2.2.1 Gather System Information

An initial information request was made to Yankee Atomic personnel. The information sources received included the Maine Yankee FSAR, systems training manuals, piping and instrumentation diagrams, electrical/actuation system schematics, Technical Specifications, emergency and abnormal operating procedures, surveillance, tagging and temporary modifications control procedures, and various structural specifications and calculations. From these sources, and from answers to specific questions asked throughout the review, the systems review was conducted.

2.2.2 Classify Front-Line Systems

For each initiating event to be considered, the Maine Yankee front-line systems were classified as Group A or Group NOT-A according to the safety functions listed in Table 2-1 (Table 2-2 of NUREG/CR-4482). For a PWR, the Group A safety functions are reactor subcriticality, normal cooldown, and emergency core cooling (early). Based on guidance in NUREG/CR-4482, a seismic induced loss of offsite power (LOOP) is assumed for a margins review. With a LOOP, the normal cooldown systems would not be available, and are therefore not considered further. In addition, an initiating event which includes a small LOCA due to the seismic event, or a nonseismic event such as a reactor coolant pump seal LOCA, or a safety/relief valve stuck open was considered. Based on guidance from NUREG/CR-4482 medium and large LOCAs were not considered.

The impact of a containment isolation failure during no LOCA and small LOCA events was considered. It was deemed to be not important to core damage frequency for a LOOP transient, although it could affect the accident source term and offsite dose. For a small LOCA, failure of containment isolation could affect recirculation by an eventual loss of inventory in the containment sump, but this would only occur in the very long term. Given an emergency source of ac power, recovery actions such as refilling the RWST or manually isolating containment are likely. Therefore, potential impacts of containment isolation failure were not considered further.

From an understanding of Group A systems and chosen initiating events, preliminary event trees were developed to show the potential accident sequences leading from an initiating seismic event to core damage. Guidance in NUREG/CR-4482 suggested that event trees and schematics from the NRC sponsored Accident Sequence Evaluation Program (ASEP) could be used. However, the ASEP event trees and system schematics were not available for Maine Yankee, and the ASEP generic trees were too nonspecific for use in this analysis. Therefore, methods for event tree development from the IREP procedures guide (Carlson, 1983) and the PRA procedures guide (NRC, 1983) were used.

2.2.3 Identify Group A Front-Line Components

Once the Group A front-line systems were determined, the components within these systems were identified. For each component, its function, necessary support systems such as power or cooling, normal status (operating, standby, open, closed), and failure mode (if applicable) were noted. Components of Group A systems that were not required to support the safety function were generally screened out. However, components which did not support the Group A function of the system but whose failure could cause the failure of a necessary component were identified. A primary example is a heat exchanger that is not required to remove heat during an accident, but is required to maintain its pressure boundary integrity in order to assure system success. Components located on potential flow diversion paths from the main piping were also noted. Flow diversion paths less than one-third the size of the main piping or with flow restriction orifices were screened out only for open systems. For closed systems, all diversion paths were considered. Simplified schematics for the systems and components were then developed for use in the first plant walkdown. More components were shown on the simplified system schematics than would be evaluated, such as manual valves, check valves and recirculation lines. These components were included because they were needed to understand the system operation. All questions which arose and assumptions made were recorded for discussion during the walkdown.

2.2.4 Classify Support Systems

Those systems which provide a required support function for the operation of a Group A front-line system were classified as Group A. In turn, some of the Group A support systems required the functions of other support systems, and these other support systems were also classified as Group A. From the components identified in the prior step, a front-line system versus support system dependency matrix was developed.

2.2.5 Identify Group A Support System Components

In a manner similar to that described for Group A front-line systems, the Group A support system components were identified. Those support system components which do not support the Group A function of the system were screened out, unless their failure could cause the failure of a necessary component. The resulting list of support system components was used in developing a support system versus support system dependency matrix. Simplified schematics were also developed for each Group A support system.

2.2.6 Identify Plant-Unique Features

Any features (systems or components) unique to the Maine Yankee plant that could potentially affect the seismic margin of the plant were identified to be considered along with the Group A front-line and support systems. An example of a plant-unique feature is the dam and fire water pond at Maine Yankee.

2.2.7 Prepare for First Plant Walkdown

In preparation for the first walkdown, several items were submitted to the fragilities analysis team as the system documentation became available. These items included the Group A component lists, the simplified system schematics, the preliminary event trees and the lists of the questions and assumptions to be discussed. The lists of components designated each component as required for Group A system operation, or as required for pressure boundary integrity only. Some of the system assumptions and bases listed were based on knowledge of other PRAs and required verification for Maine Yankee. At this point some alternate systems (such as the alternate shutdown decay heat removal Appendix R system) were still being evaluated and were included on the preliminary event trees prior to being screened out.

2.3 Step 4 - First Plant Walkdown

The first plant walkdown provided an opportunity to accomplish three tasks:

- Hold a Peer Review Group meeting
- Conduct the plant walkdown
- Hold discussions with plant staff

Each of these tasks contributed to the systems analysis. Documentation of the first walkdown included marked up schematics, revisions to system component lists, answers to questions, verification of assumptions, and new system information.

2.3.1 Peer Review Group Meeting

During the Peer Review Group meeting, clarification on the treatment of several open items was sought. These items included whether or not to evaluate the boric acid transfer system for the reactor subcriticality function, long-term room cooling for pumps, and the initial switchover to recirculation from high pressure safety injection.

2.3.2 Plant Walkdown

Several systems analysis tasks were to be accomplished in the first walkdown. Most important of these were:

- Verification of screening completed thus far using a checkoff list.
- Identification of the relationships between component layout and the timing for local recovery actions.
- Location of some components, such as the diesel generator start circuits and load shed/sequencers.

- Identification of the impact of potential failures of the system components.

- Identification of possible physical interactions between Group A system components and other items (block walls, fire water).

- Walkdown of secondary component cooling water piping to check integrity and potential interactions.

In performing the walkdown it was useful to have the plant arrangement drawings in addition to the simplified system schematics. As the walkdown was conducted with the analysis team members divided into two groups, it became important for the groups to consult at the end of each day to compare notes and prepare for the next day.

2.3.3 Plant Staff Discussions

Discussions held with members of the plant staff were to gain additional system information and answers to the questions existing at that point. Information on valve failure positions, normal positions of manual valves, unidentified instrument air and power supplies, and operating procedures would be used in completing the systems documentation. Also, simplified system schematics were verified and plant-unique features discussed.

2.4 <u>Step 5 - Systems Modeling</u>

To complete the systems modeling task it was necessary to:

- Review, update, and document the event trees and their success criteria.

- Develop and document fault trees for the front-line systems which are included in the event tree sequences, and for their required support systems.

- Develop a data base consisting of probability cutoffs for screening, and data for calculating component unavailabilities and human error probabilities.

- Determine the minimal cut sets for the front-line systems (including support system faults) to verify the fault tree logic and identify critical items to review during the second plant walkdown.

2.4.1 Review Event Trees

During the first plant walkdown, several insights were gained which were used in revising the preliminary event trees. The types of revisions are illustrated by the following:

- The recirculation function of the containment spray pumps (which provide water from the containment sump through the residual heat removal (RHR) heat exchangers to the high pressure safety injection (HPSI) pumps for long-term core cooling) were included. Specifically, the fans, ducting and dampers which provide room cooling for the containment spray pump area were added to the HPSI system fault tree. Thus, failure of room cooling would eventually fail HPSI in the recirculation mode.

- The alternate shutdown decay heat removal (ASDHR) system was screened out of the analysis. The main control room panels were judged to be seismically sound, precluding the need for use of the alternate shutdown panel (ASP). Also, operation of the turbine-driven auxiliary feedwater (AFW) pump is required for ASDHR, and is already considered in the AFW system. Finally, the auxiliary charging pump is judged too small to replace the HPSI pumps for feed and bleed or core inventory makeup.

- The evaluation of the boric acid transfer (BAT) system was delayed and eventually screened out, because the reactor internals and control rod drive system were found to have high seismic capacity, thus eliminating the need for BAT.

- An anticipated transient without a SCRAM (ATWS) was not evaluated because the reactor internals and control rod drive system were found to have high seismic capacity.

Two separate event trees were developed, with different initiating events. One tree evaluates mitigating system success following a seismic event causing loss of offsite power (T_1). The second tree addresses system success following a seismic event which causes loss of offsite power, concurrent with a small LOCA (S_2).

The final event trees are discussed in greater detail in Section 4.1.

2.4.2 Develop Fault Trees

Once the functions, success criteria, and support system interfaces for the Group A systems were identified, system fault trees were developed using standard techniques. For each system, potential flow diversion paths were analyzed. A path was excluded if it met at least one of the following criteria, which are based on probabilistic arguments.

- There is a normally closed automatic valve isolating the line.
  Exception: Include the path if interlocks exist which may prevent the valve from closing, and these interlocks are connected to a component or system which is modeled, or the interlock and valve are powered by opposite buses.

- There are two normally open valves on the line which will automatically close.
  Exception: Include the path if the line up to the second valve has a potentially low seismic capacity (i.e., contains a heat exchanger or is not seismic class 1).

- The line or flow restriction orifice is less than one-third the diameter of the main pipe.
  Exception: Include the path if it is a closed system. However, the ability of the operator to isolate small leaks will also be taken into consideration.

During this process, additional questions and issues were raised. A list of these questions was compiled for referral to the plant staff. It was necessary to include some diversion paths until information was received which would allow screening of the path.

The components and paths identified by the above steps were used in developing the system fault tree logic. The fault tree symbols and event identifiers used are found in Appendix A. The first set of front-line and support system fault trees developed include every system component identified (unless it was screened out as a diversion path component). These fault trees would later be pruned using failure probability cutoffs. As seismic and nonseismic failure modes were not differentiated at this point, a specific component failure mode was not identified. Rather, an "XX" was used as a generic failure mode identifier. Support system faults are added to the tree as developed events. The support system fault trees would later be merged into the front-line system fault trees at these points. Failures of support systems were included at each component level rather than being combined at the top of the fault tree. Check valves were not included in the fault tree logic based on their low unavailability. Manual valves normally in the correct position were also not included in the fault tree logic because of their low probability of being in an incorrect position. However, if a manual valve could be in an incorrect position and also fail all trains of a redundant safety system, then its probability of incorrect positioning was evaluated further to determine if it was below the screening probability cutoff.

Once the initial system fault trees were completed, preliminary front-line and support system cut sets (first, second, and third order) were obtained using Micro-PRANK. These cut sets were analyzed to verify the logic of the fault tree.

Following verification of the fault tree logic, probability cutoffs described in the following section were used to prune the fault trees. Seismic, random, common cause, test and maintenance, and human error failures were considered for each component. Along with seismic failures of specific components, seismic failures of adjacent components, walls or buildings which may fail a component of interest were also considered. For seismic events screened in, the failure mode identifier "EQ" was used. The same event names were used for several component failures if the seismic failure of the components was highly correlated. For example, CCW-HTX-EQ-4B5A represents the seismic failure of the PCC and the SCC coolers since they are identical and located adjacent to each other. All components with seismic HCLPFs greater than 0.3g, or nonseismic failure probabilities below the probability cutoffs were pruned from the fault tree. The support system inputs for these components remained on the tree. It took several passes to prune the fault trees as fragility calculations were finalized and system spatial dependencies were identified in the second walkdown.

Also pruned from the fault trees were any components which are isolated by valves removed from the tree in the step described above. This eliminated flow diversion paths which would be successfully isolated following the initiating event. Support system interfaces for these pruned diversion path components were also removed from the tree. The separation of instrumentation, racks, and impulse lines which make up the actuation systems included in the fault trees was also checked to determine if a single physical seismic interaction could fail multiple actuation channels. If there was sufficient separation (no more than one channel is failed by a physical seismic failure), and the transmitters and racks were of high seismic capacity, then actuation system faults were pruned from the fault trees.

Upon completion of the pruning process, the fault tree "OR" gates with zero or one remaining inputs were collapsed into their output gates. The "AND" gates which had one or more inputs removed were deleted from the tree. The pruned support system fault trees were then merged with the pruned front-line system fault trees. Developed events on the trees became the top gate of the support system tree to be merged at that point.

The final result of the fault tree development process was a fault tree for each front-line system included in the event tree sequences. Each fault tree includes basic and undeveloped events representing the seismic and nonseismic failures of the front-line and support system components which were not screened out.

2.4.3 Develop Data Base

A data base containing probability cutoffs and nonseismic failure data was developed to aid in pruning the fault trees. Seismic failure data from the fragilities team which was required in the component screening was included in the data base.

Generic component unavailability and failure rate data from ASEP was used to calculate random nonseismic failures. Beta factors from EPRI NP-3967 (Fleming, 1985), with supplements from ASEP and other PRAs were used to develop nonseismic common cause unavailabilities for components. A list of components for which information on testing and maintenance is required was compiled for verification on the second walkdown. All of the components were adequately represented by generic ASEP test and maintenance unavailabilities. To determine probabilities for human errors and recovery, time based data from the IREP NUREG/CR-2787 (Kolb, 1982) with some guidance from ASEP, was used.

In screening system components, seismic failures for those components with a HCLPF capacity calculated to be greater than 0.3g were screened out. Random, common cause, test and maintenance, and human error failures with probabilities less than the following guidelines were also screened out.

- 0.01 if the failure leads to the loss of only one train in one system.
- 0.001 if the failure leads to the loss of all trains in one system.
- 0.001 if the failure leads to the loss of one train in multiple systems.
- Exception: If the failure probability is greater than the cutoff, but has a high probability of recovery, making the combined probability less than the cutoff, the component failure may be screened out.

Component unavailabilities, human error probabilities, and nonseismic common cause failure probabilities are presented for the components of interest in Chapter 3.

2.4.4 Determine System Cut Sets

The final front-line system fault trees obtained from the steps described in Section 2.4.2 (pruned and merged with the support system fault trees) were analyzed to determine the first, second, and third order cut sets for the system. This minimal cut set evaluation before the second walkdown was not included in the NUREG/CR-4482 steps, but proved to be very useful. Micro-PRANK, a minimal cut set routine included in the personal computer based Fault Tree Workstation, was used for this analysis. The resulting cut sets aided in recognizing some critical items (especially unexpected single faults). These items were identified as warranting a special look in the second walkdown, and discussion with the plant staff. Among the items listed for further evaluation during the second walkdown were the new procedure for isolating portions of the PCC, the station transformers, and the containment spray pump area fans.

2.5 Step 6 - Second Plant Walkdown

A second plant walkdown was performed to look at those items that had been added to the component list since the first plant walkdown. This included items such as block walls which may impact the major system components. Also, any questions which arose during the fault tree development task were discussed with plant personnel and, if necessary, a second examination given to the subject items. Plant-specific test and maintenance (planned and unplanned) data for important components (pumps and diesel generators primarily) was also gathered.

The second plant walkdown also provided an opportunity to discuss the seismic review methods and any issues which had arisen with the Peer Review Group. It was also a chance to support the fragility analysis team in their efforts to calculate component capacities, and to obtain a preliminary estimate of which components may be screened out due to sufficient seismic capacity.

2.6 Step 7 - Determine Minimal Cut Sets

In order to obtain the minimal cut sets and Boolean equations for the two event trees (no LOCA and small LOCA) the following steps were completed:

- Review and finalize the event trees.
- Review and finalize the fault trees.
- Link the fault trees according to the event tree sequences.
- Obtain a preliminary Boolean equation for each event tree.
- Repeat these steps to incorporate the complete fragility analysis results to obtain the final Boolean equations.

2.6.1 Finalize Event Trees

The event trees produced in Step 5 were reviewed to determine consistency with the additional information gathered during the second plant walkdown.

2.6.2 Finalize Fault Trees

The pruned and merged front-line system fault trees developed in Step 5 were also reviewed to incorporate the answers to the questions discussed during the second walkdown, the test and maintenance data, and the preliminary fragility analysis results. Screening overview tables were developed to trace the status of each system component. For each item, the tables list the component name, its screening status (in or out, for seismic and nonseismic failures), the reason for its screening status, the applicable event name, and the component unavailability or seismic capacity value used for comparison with the probability and HCLPF cutoffs.

These finalized pruned fault trees were used in determining the Boolean equations. As discussed in NUREG/CR-4482, it is possible to determine preliminary Booleans prior to pruning the system fault trees, but this would result in roughly an order of magnitude more cut sets. The method of using pruned fault trees was chosen because it is more efficient.

2.6.3 Link Fault Trees

The fault trees were first linked to form sequence level fault trees, and then to form event tree level fault trees. The pruned front-line system fault trees which had been merged with the required pruned support system fault trees were linked in combinations which represent the various event tree sequences. For example, if failure of auxiliary feedwater along with failure of the high pressure safety injection system would result in core damage, these two system fault trees were linked by an "AND" gate to create a larger fault tree which represents this core damage sequence.

Once fault trees were developed for each sequence, the sequence level trees were linked to create a fault tree which represents an entire event tree. For example, given an initiating event, if there are three sequences, of which any one will lead to core damage, the fault trees for these three sequences were linked by an "OR" gate. These linked fault trees were then analyzed to determine the Boolean equation for each event tree.

2.6.4 Determine Preliminary Boolean Equation

Analysis of the linked fault trees was performed in two stages to determine the Boolean equations. A first analysis was performed with the analysis routine Micro-PRANK. The linked sequence level fault trees were analyzed separately for each sequence.

Once cut sets for each sequence were obtained, they were combined and reduced by hand to develop the cut sets for an entire event tree. In doing this, it was important to ensure that only minimal cut sets were included for the event tree level result. Since it was possible that an event that occurs as a single fault for one sequence appeared in a double fault for another sequence, when these sequence cut sets were combined the double faults containing that event were deleted.

After event tree level cut sets were determined in this manner, they were transferred to the fragility analysis team for preliminary plant HCLPF determination. Later, the large event treee level fault trees were analyzed using VAX SETS to obtain more detailed Boolean equations. These equations were used to verify the results obtained from Micro-PRANK. Because this is a seismic review, only the cut sets which include at least one seismic event are of interest. Therefore, those cust sets that contained only nonseismic events (e.g., random, human error) were deleted from the Boolean equations.

2.6.5 Determine Final Boolean Equation

When the component fragility analysis calculations had been finalized, those results were incorporated in the plant level Boolean equations. To do so, the steps described in Sections 2.6.1 through 2.6.4 were repeated. This resulted in a change in screening status for several components, which in turn altered the final Boolean. The final Boolean equations were used by the fragility analysis team in determining the final plant capacity.

Table 2-1. Definition of plant safety functions.
(Table 2-2 from NUREG/CR-4482)

IDENTIFICATION OF SAFETY FUNCTIONS

1. Reactor Subcriticality - shutting down the nuclear reaction such that the only heat being generated is decay heat.
2. Normal Cooldown - providing cooling to the reactor core through the use of the normal power conversion system, normally defined as the main steam, turbine bypass, condenser, condensate, and main feedwater subsystems.
3. Emergency Core Cooling (Early) - providing cooling to the reactor core in the early (transient) phase of an event sequence by the use of one or more emergency systems designed for this purpose. The exact timing of "early" is somewhat plant specific and sequence dependent. However, for our purposes it can be deemed to be the time period during which these systems are initially called upon to operate.
4. Emergency Core Cooling (Late) - providing cooling to the reactor core in the late (stabilized) phase of an event sequence by the use of one or more emergency systems designed for this purpose. In context with the above definition of "early", for our purposes "late" can be deemed to begin with the switchover to recirculation (for LOCAs) or with the achievement of residual heat removal conditions (for transients).
5. Containment Heat Removal - removing heat from the containment to the ultimate heat sink during the late (stabilized) phase of an event sequence by the use of one or more safety systems designed for this purpose.
6. Containment Overpressure Protection (Early) - controlling the buildup of pressure in the containment caused by the evolution of steam by condensing this steam during the early phase of an event sequence by using one or more safety systems designed for this purpose. "Early" in the context of containment functions is not the same as "early" for core cooling. In this case "early" is deemed to be the time period commencing when this function is required after the beginning of core melt when these systems are operating in the injection mode.
7. Containment Overpressure Protection (Late) - controlling the buildup of pressure in the containment caused by the evolution of steam by condensing this steam during the late phase of an event sequence using one or more safety systems designed for this purpose. In the context of the previous definition, "late" in this case is deemed to start when these systems are operating in the recirculation mode.

IDENTIFICATION OF THE FUNCTIONAL GROUPS FOR PWRS AND BWRS.

PWR

Group A:	Functions 1,2,3
Group NOT-A:	Functions 4,5,6,7 + All plant functions not related to Safety

BWR

Group A:	Functions 1,2,3,4,5,6, 7
Group NOT-A:	All plant functions not related to Safety

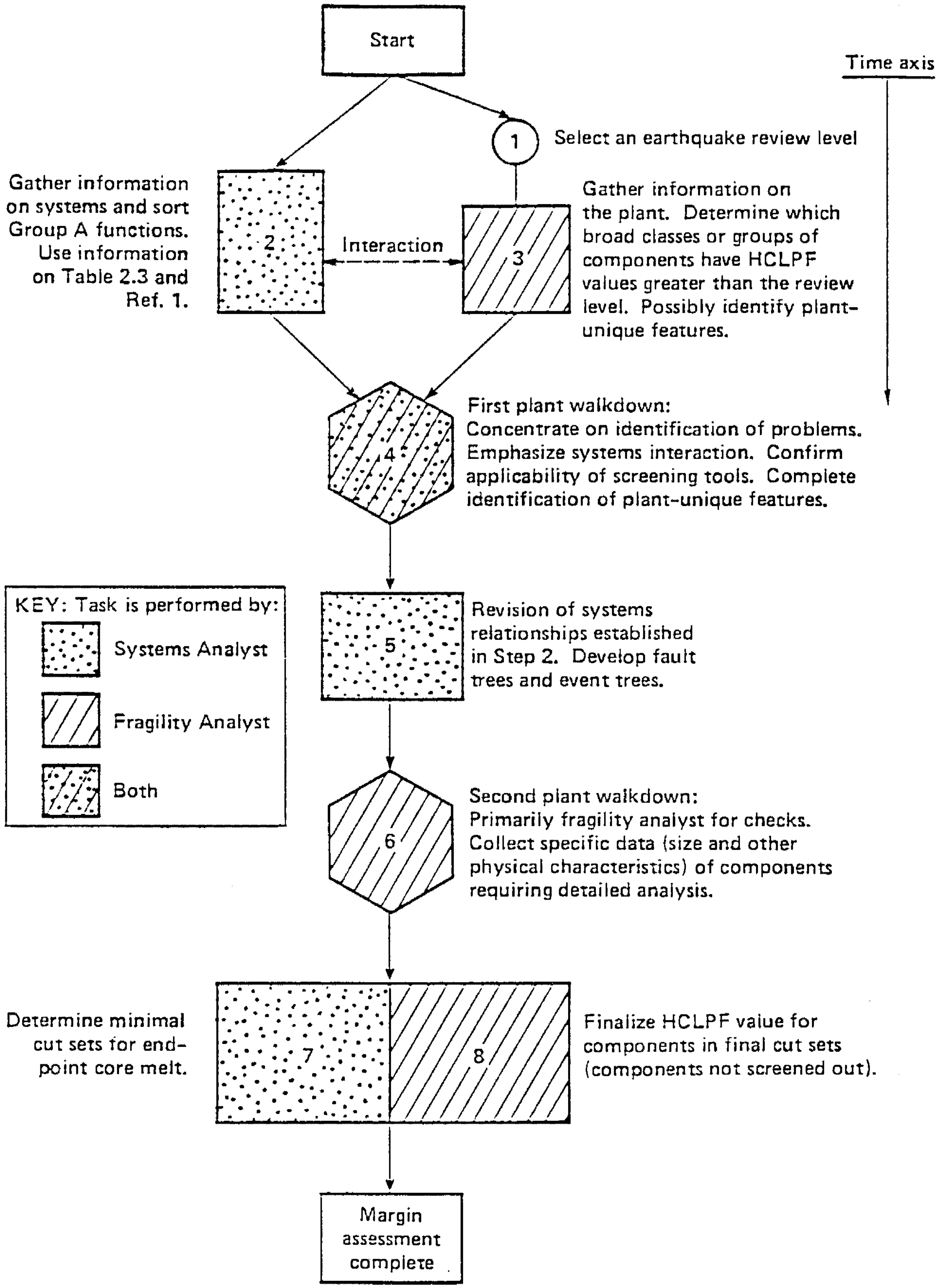


Figure 2-1. Graphic Representation of the Screening Operations (Figure 2-6 from NUREG/CR-4482).

3. SYSTEMS ANALYSIS

This chapter presents the results of the systems modeling and quantification steps. Section 3.1 discusses the identification of Group A systems, both front-line and support. It also presents the basis for screening out potential Group A systems from further evaluation. Each Group A front-line system is then described in Section 3.2, and the fault trees and screening evaluations are presented. Section 3.3 provides similar information for the Group A support systems. The failure probability and unavailability bases for the components that could not be screened out immediately is presented in Section 3.4.

3.1 System Identification

The purpose of the system identification and classification was to determine which plant systems are potentially available to bring the plant to a safe shutdown following a seismic event. As discussed in Chapter 2, those front-line systems (with the necessary support systems) whose failure will result in failure of reactor subcriticality or loss of emergency core cooling (early) are designated as "Group A" systems. Unless there are plant unique features or other reasons for inclusion, all other systems are designated as "Group Not-A" systems. Failures or successes of the Group Not-A systems are not evaluated.

3.1.1 Group A Systems

Those systems identified as Group A which perform the reactor subcriticality and early core cooling functions are the reactor protection system (control rods and reactor internals), the auxiliary feedwater system, the primary pressure relief system (power-operated relief valves for feed and bleed), and the high pressure safety injection system.

Based on the fragility analysis of the reactor internals and the generic HCLPF for the control rod mechanisms, the seismic capacity of this reactor subcriticality system was found to be adequate at the review earthquake level. Nonseismic failures of the system are below the probability screening cutoffs. Due to these findings, fault trees of the reactor internals and control rod mechanisms were not developed, and reactor subcriticality systems would not impact the plant HCLPF calculation.

Upon loss of the main feedwater system (due to loss of offsite power) and a corresponding drop in steam generator level, the auxiliary feedwater (AFW) system supplies water from the demineralized water storage tank or the primary water storage tank to the steam generators for decay heat removal. A minimum level must be maintained in at least one of the three steam generators for successful heat removal.

The power-operated relief valves (PORV), which with the safety relief valves comprise the primary pressure relief system, are used for bleeding the pressurizer in the reactor coolant system (RCS) during feed and bleed core heat removal. The PORVs are actuated by the operator for this function. If the seismic event is accompanied by a small LOCA, opening of only one PORV would provide sufficient feed and bleed capability. Otherwise, both PORVs must be opened.

The high pressure safety injection (HPSI) system is also required for feed and bleed capability, and for core cooling during a LOCA. The HPSI system provides borated water from the refueling water storage tank to the reactor coolant system loops for makeup during feed and bleed, and core cooling. Also included in the HPSI analysis are the fans which provide cooling for the containment spray pump area, and the valves which are required to operate to achieve recirculation. Although recirculation is not a Group A function, it is necessary to ensure the long-term operating ability of the containment spray pumps for emergency core cooling during recirculation. This departure from the NUREG/CR-4334 and 4482 guidelines is discussed in Chapter 5.

The necessary support systems identified for the Group A front-line systems are electric power (ac, dc, and the emergency diesel generators), component cooling water (primary and secondary), service water, and actuation (primarily the safety injection actuation system, SIAS). Some air accumulators are also required, but are included with the front-line systems they support. The front-line to support system relationships are shown in the dependency matrix in Table 3-1. The support system - support system dependencies are shown in Table 3-2.

3.1.2 Systems Removed from Group A

Some systems which might be called upon to perform a required Group A function are the boric acid transfer system, the low pressure safety injection system, the safety injection tanks, the secondary pressure control system, and the alternate shutdown decay heat removal system. This section discusses the basis for not evaluating these systems in detail.

The boric acid transfer (BAT) system provides emergency boration from the boric acid storage tank to the RCS in the event of an ATWS or loss of shutdown margin. As the probability of failure of the reactor internals due to the review level earthquake or nonseismic event is insignificant, emergency operation of the BAT system is not likely to be required for prevention of core damage, and was excluded from further consideration.

The low pressure safety injection (LPSI) system and the safety injection tanks (SIT) provide sufficient borated water to flood and cool the core following a medium or large LOCA. As discussed in Chapter 2, only a small LOCA is postulated as concurrent with the seismic event. Therefore, the LPSI and SIT systems would not be required for core cooling. The feed and bleed actions by the PORVs and the HPSI system provide sufficient core cooling for transients or a small LOCA. Also, the LPSI pumps are not used in the recirculation mode at Maine Yankee.

The secondary pressure control system provides overpressure protection for the steam generators and main steam system piping by means of safety relief valves and an atmospheric steam dump valve. The safety relief valves are expected to open for pressure relief, and it is not of major concern if they should stick open, although the RCS would rapidly depressurize. If the HPSI system should fail, the atmospheric dump valve may be used to depressurize and cool the secondary system which in turn would cool the primary side, enabling the use of the LPSI system for depressurization. However, with the reactor coolant pumps made inoperable by loss of offsite power, there is no assurance that sufficient depressurization would occur (Fletcher, 1981). The atmospheric dump valve may also be used for depressurization to enable the use of a low pressure water supply (such as fire water) to feed the steam generators in the event of AFW failure. However, the procedure for this operation depends on use of the AFW system at the start of the procedure for initial cooldown. Due to these usage limitations and dependencies, the secondary pressure control system was not evaluated further.

The alternate shutdown decay heat removal (ASDHR) system provides an alternate means of controlling and monitoring a plant shutdown, and was designed to meet the Appendix R requirements for fire mitigation. The alternate shutdown panel (ASP) provides a remote monitor and control location, but as the main control room (MCR) panels were found to be of sufficient seismic capacity, use of the ASP is not necessary. The ASDHR system utilizes the turbine-driven AFW pump and the auxiliary charging pump to achieve shutdown. The turbine-driven pump is analyzed with the Group A AFW system, and the auxiliary charging pump is of too small a capacity to replace the HPSI pumps for feed and bleed or coolant makeup operations following a small LOCA. Due to these limitations of the ASDHR system, it is not analyzed further as a Group A system.

3.2 Systems Analysis of Front-Line Systems

Information on the Group A front-line and support systems includes the Final Safety Analysis Report (FSAR), system piping and instrument diagrams (P&IDs), standard and abnormal operating procedures (SOPs and AOPs), system training manuals, and system analyses and calculations. Maine Yankee and Yankee Atomic personnel are a major source of information for questions and clarification.

Major items covered in the following system analysis discussion include the function of the system, success criteria, system components, boundaries of the system, support system interfaces, operator and recovery actions, necessary assumptions, bases for fault tree development, fault tree logic, failure probabilities of events, fault tree pruning and merging logic, and system cut sets.

3.2.1 Auxiliary Feedwater System

The auxiliary feedwater system is used to maintain a minimum level in the steam generators for decay heat removal, following loss of the main feedwater system. For system success, operation of at least one of the three AFW pumps is required to maintain a minimum water level in at least one of the three steam generators. The AFW system consists of the demineralized water storage tank (DWST), three pumps (two motor-driven and one turbine-driven), and associated flow control and isolation valves. Although the motor-driven pump trains at Maine Yankee are termed emergency feedwater, they will be classified as part of the auxiliary feedwater system in this report. A complete list of AFW components, valves and cooling requirements is included in Appendix B.

The two motor-driven pumps (located in the auxiliary feed pump house, el. 21 ft) start automatically on a steam generator low level signal. Upon loss of offsite power, the diesel generator load sequencers restart the motor-driven pumps after a 20-second time delay. If necessary, the turbine-driven pump (located in the steam and feed pump valve area, el. 21 ft) is placed in service by the operator by aligning the steam inlet valves to the turbine drive. Flow to each steam generator is regulated by an air-operated flow control valve (located in the auxiliary feed pump area, el. 23 ft) which is paired with an air-operated isolation valve. A schematic of the AFW system is shown in Figure 3-1.

Support systems required for AFW operation are 4160-V ac emergency power, 125-V dc power, instrument air (from various accumulators), and main steam to drive the turbine.

The following items were used as the basis for development of the fault tree logic:

- An alternate water supply for the AFW system is the primary water storage tank (PWST). However, there is no check valve on the line between where the PWST feed joins the suction for the turbine-driven pump and the DWST, making it possible for the PWST to be drained through a ruptured DWST. Therefore, the PWST is considered an alternate supply only for the motor-driven pumps.

- A potential source of makeup to the DWST is from condensate makeup. As this makeup is supplied by gravity feed to the DWST, it could also be drained by a DWST rupture. Therefore, this source of makeup is not included in the analysis.

- Sufficient separation of the steam generator instrumentation exists, such that a single physical seismic failure would not fail the complete AFW instrumentation and actuation systems. Based on the high seismic capacity of the racks and transmitters, the instrumentation is not analyzed further in the analysis.

- The turbine-driven AFW pump may be placed in service from the main control room or locally. Since the diagnostic portion of these two actions are related, this dependency is explicitly modeled in the fault tree. If the operator fails to place the pump in service from the control room, it is not likely that he will do so locally.

- Check valve failures, random valve and MDP failures, and valve failure due to plugging, testing or maintenance are of low probability. Therefore, these failures are screened out of the analysis.

- Because the AFW flow control and isolation valves fail open upon loss of instrument air or solenoid power, the related air accumulators and power supplies are not included in the fault tree.

- Because the AFW pump oil coolers are considered an integral part of the pumps, they are not shown as separate components in the fault tree.

- Several recovery actions are not included in the fault tree. These actions include opening the manual flow control bypass valves to allow flow, the use of fire water to maintain water level in the steam generators, and the potential recovery of the main feedwater system. Inclusion of these recovery actions is judged not to impact the plant HCLPF significantly.

- The recirculation lines from the AFW pumps to the DWST are not included as potential flow diversion paths. Each recirculation line contains a flow restriction orifice and is less than the screening value of one-third the size of the discharge line.

- The lines off the AFW pump discharge to the main feedwater header are not included as potential flow diversion paths. Each line contains a normally closed manual valve.

- The line containing the chemical feed tank (TK-89) and pump (P-115) is not included as a potential flow diversion path. The line is less than one-third the size of the AFW pump discharge line and contains a normally closed manual valve.

- Of the valves which supply steam to the turbine-driven pump, two (MS-P-168 and MS-T-163) are air-operated and receive air from the same accumulator (TK-25). The third (MS-A-173) is mechanically operated and is locked open.

The basic fault tree for the AFW system is shown in Appendix B. This fault tree includes the AFW components, which have not been screened out at this point, and their required support system inputs. The faults for each component are nonspecific as to the type of failure. Specific failures (e.g., seismic, random, common cause) were added as each component was analyzed. Linear sections of the system are represented by pipe segments (PS-1, PS-2, etc.). The top event is failure of the AFW system, this failure occurs through common cause failure of the AFW, failure of the main control room panels, or failure within the system to maintain a minimum level within any one of the steam generators. This latter event is caused by lack of flow through all three pairs of control and isolation valves, which in turn may be caused by lack of flow from all three AFW pumps.

From this basic fault tree, screening criteria described in the methodology chapter were applied to each component in order to prune the fault tree. These screening results are summarized in Table 3-3. The footnotes for Table 3-3, and for all the other screening tables, are listed on Table 3-4. Once the AFW fault tree was pruned, it was merged with the necessary support system fault trees. (Construction and pruning of the support system trees is discussed in Section 3.3.) The completed AFW system fault tree is presented in Appendix B.

Analysis of the pruned, merged fault tree is performed to determine the AFW system cut sets. A validity check of these cut sets ensures that the fault tree logic is correct. The minimal cut sets (first and second order) for the AFW system are listed in Table 3-5. For ranking purposes, this analysis assumes that the failure probability for each seismic event is 1.0. The HCLPF for these events will be developed by the fragility analysis team. The calculations for the failure probabilities of the nonseismic events are shown in Section 3.4.

The only singlet for the AFW system is AFW-CCF-FC-AFW, nonseismic common cause failure of the pumps or air-operated valves. Of the 23 double faults, the first 19 contain one or more seismic events. Seismic failure of the DWST will fail the turbine-driven AFW pump and the primary water supply for the motor-driven pumps. Failure of the PWST fails the motor-driven pump backup water supply. Seismic failure of the transformers, PCC/SCC coolers, air conditioner chillers, circulation water pump house, DG day tanks, or failure to refill the DG fuel tanks will result in loss of power to the motor-driven pumps, as will common cause failure of the DGs.

3.2.2 High Pressure Safety Injection System

The high pressure safety injection system is used to supply makeup to the reactor coolant system for post-accident core cooling and during feed and bleed. For system success injection of borated water from the refueling water storage tank (RWST) to at least one RCS loop by operation of at least one HPSI pump is required. The HPSI system consists of the RWST, three motor-driven pumps (one is an installed spare), two pairs of RWST isolation valves, two pairs of pump discharge valves, and three pairs of injection isolation valves (one pair per RCS loop). Complete lists of the HPSI components, valves and cooling requirements are included in Appendix C.

Also included with the HPSI system are the two fans that provide cooling for the containment spray pump area. Although the containment spray pumps are used for recirculation, which is not a Group A function, it is necessary to ensure the long-term availability of these pumps. In part, this is accomplished by maintaining a cool operating environment. This departure from the NUREG/CR-4482 screening guidelines is discussed in Chapter 5.

A schematic of the HPSI system is shown in Figure 3-2. One HPSI pump (P-14A or B, located in the primary auxiliary building (PAB), el. 21 ft), is normally operating as a charging pump, the other (standby) pump is automatically started by a safety injection actuation signal (SIAS). The third pump (P-14S, also in the PAB) is a spare that must be placed in service manually. Upon loss of offsite power, the diesel generator load sequencers restart the two in-service pumps. The pump suction valves on the RWST discharge open on a SIAS. One motor-operated valve of each pump discharge valve pair (HSI-M-41 and 42, located in the PAB, el. 23 ft) is opened upon a SIAS.

Support systems required for HPSI and containment spray pump area fan operation are 4160- and 480-V ac emergency power, 125-V dc power, primary and secondary component cooling, and safety injection actuation trains A and B.

The following items provided the basis for the fault tree logic:

- Failure of the spray chemical addition tank (SCAT), which is connected to and located adjacent to the RWST, may lead to failure of the interconnecting line and thus drain the RWST.

- Sufficient separation of the SIAS instrumentation exists, such that seismic failure of a rack or instrument would not fail the entire actuation system. Also, the operator is capable of initiating the actuation system. Therefore, SIAS instrumentation is not included further in the analysis.

- Control power for the motor-operated valves is assumed to be transformed off the bus which provides the valve motive power.

- Check valve failures, random valve failures, and valve failure due to plugging, testing or maintenance are of low probability. Therefore, these failures were not included in the analysis.

- Pipe ruptures were included only if screened in during the walkdowns.

- Because the recirculation mode is not a Group A function, flow from the residual heat removal heat exchangers to the HPSI pump suction header is not included in the fault tree.

- Several recovery actions are not included in the fault tree. These actions include opening the pump discharge valves which do not open on a SIAS, and placing the spare pump in service by opening manual valves and racking in the breaker. Their inclusion would not significantly impact the plant HCLPF.

- Block wall VE 21-1, 2 may fail the containment spray pump area fans should it collapse.

- The recirculation lines from the HPSI pumps to the seal water heat exchanger and volume control tank are not included as potential flow diversion paths. Each line contains a restriction orifice and is less than the screening criteria of one-third the size of the discharge line.

- The lines from the boric acid transfer pump discharge, volume control tank and RHR heat exchangers to the HPSI pump suction header are not included as potential flow diversion paths. Each line contains a check valve to prevent backflow, and either a normally closed motor-operated valve, or a MOV which closes upon a SIAS.

- The line from the P-14A suction to the suction of the auxiliary charging pump (P-7), and the line from the RWST to P-7 are not included as potential flow diversion paths. Each line contains a motor-operated valve, and the P-7 discharge line contains a normally closed manual valve.

- The lines from the HPSI pump discharge to the charging header are not included as potential flow diversion paths. Each line contains an air-operated valve (CH-A-32 and 33) which closes on a SIAS. The two lines combine and then split into three lines which go to the charging header, loop fill header, and seal water heater. The line to the charging header contains an MOV (CH-F-38) which closes on a SIAS. The line to the loop fill header contains a normally closed flow control valve (CH-F-70). Downstream of the seal water heater is an isolation valve (SL-P-3) which also closes on a SIAS. As CH-A-32 and CH-A-33 are seismically sound, they will successfully isolate these lines and the regenerative heat exchanger and seal water filter.

The basic fault tree for the HPSI system is shown in Appendix C. This fault tree includes the HPSI components which were not yet screened out and their required support system inputs. The type of failure for the components is not specified. Specific failures (seismic, random, etc.) were added as each component was analyzed. The top event of the tree is loss of HPSI (short-term cooling) or recirculation (long-term cooling). This event may occur through failure of the main control room panels, failure of both spray pump area fans, or failure of the HPSI system. The latter failure is caused by a system common cause failure, or by a failure within the system which prevents injection to the RCS system. Loss of injection is characterized by lack of flow to any of the three RCS loops, which may be caused by loss of both HPSI pumps. There are several basic events involved in each pump and valve failure.

From the basic fault tree, the screening methods outlined in Chapter 2 were applied to each component in order to prune the tree. The results of this screening process are detailed in Table 3-6. Once the front-line system tree was pruned, it was merged with the pertinent support system fault trees (described in Section 3.3). The final HPSI fault tree is shown in Appendix C.

Once pruned and merged with its support systems, the HPSI fault tree was analyzed to determine the system cut sets. These cut sets were used to verify the logic of the fault tree. The minimal cut sets and event descriptions for the HPSI system are listed in Table 3-7. For ranking purposes, seismic failures were assumed at this point to have a probability of 1.0. HCLPFs were later assigned by the fragility analysis team. Failure probabilities for the nonseismic events have been calculated in Section 3.4.

Of the nine single faults for the HPSI system, six are seismic events. Seismic failure of the 4160- to 480-V transformers, the PCC/SCC coolers, the air conditioner chillers, the circulation water pump house, or the DG day tanks; or failure to refill the DG fuel tanks or common cause DG failure leads to loss of power to both HPSI pumps and both spray pump area fans. HPSI is also lost upon the seismic failure of the RWST or common cause failure. There are two doublets, both of which consist of only nonseismic events.

3.2.3 Power-Operated Relief Valves (Feed and Bleed)

The power-operated relief valves (PORVs) are opened by the operator for feed and bleed. This action must take place within approximately 30 minutes of the initiating seismic event and loss of AFW. There are two sets of criteria for PORV system success. The first requires that both PORVs must be opened if a LOCA does not accompany the seismic event. The second requires that at least one PORV must be opened if a small LOCA occurs with the seismic event. In addition to the two PORVs, there is a motor-operated block valve for each PORV that must be open. A list of the system components is given in Appendix D.

The primary pressure relief system that includes the PORVs is represented in Figure 3-3. The PORVs (located at el. 65 ft in the reactor containment) are opened remotely from the main control room for feed and bleed. The motor-operated isolation valves are normally open, but allow isolation of a PORV for maintenance or if it fails to reseat. At least one isolation valve must be open at all times. The only support system required for PORV operation is 480-V ac emergency power.

The following items provided the basis for the development of the fault tree logic:

- Control power for the PORVs and block valves is transformed off the bus which provides the valve motive power.

- The automatic pressure relief function of the PORVs and the safety relief valves is not of interest in this application of the PORVs (feed and bleed).

- Random, testing or maintenance probabilities of failure for the PORVs and block valves are low. These events were not included in the analysis.

- The pressurizer, pressurizer quench tank, quench tank cooler and quench tank pumps were not included in the fault tree analysis. A seismic failure (rupture) of any of these items will not hinder the feed and bleed function. The safety relief valves were also excluded from the fault tree.

- If the control switch for an isolation valve is in the "OFF" or "CLOSE" position, it must also be opened by the operator for feed and bleed.

The unpruned fault tree for the PORVs is shown in Appendix D. Included are failures for each of the system components and the required support systems. The events in this tree do not specify the types of failures, these will be added as the tree is pruned. The top event shows failure to support feed and bleed. This is caused by a common cause failure of the PORVs and block valves, by failure of the main control room panels, or by a system failure that prevents flow through both PORVs (for the case with a small LOCA). For the case in which no LOCA has occurred, lack of flow through only one

PORV will cause system failure. There are several events shown which may lead to loss of flow through a PORV.

From the fault tree described above, the screening methods from Chapter 2 were used to prune the tree. The results of applying these screening criteria to the system components are found in Table 3-8. The pruned PORV fault tree was merged with the necessary support system fault trees (developed in Section 3.3). The complete PORV fault trees are found in Appendix D.

The final PORV fault trees were analyzed to determine the system cut sets. The cut sets were then used to check the tree logic. Table 3-9 lists the minimal cut sets for the system when no LOCA has occurred. Table 3-10 lists the cut sets for the case when a small LOCA has occurred. For minimal cut set evaluation and ranking, the failure probabilities of the seismic events were assumed to be 1.0 for this analysis. HCLPFs will be assigned later by the fragility analysis team. The nonseismic probabilities were calculated as previously described.

For the case with no LOCA, any event which causes failure of one PORV is a single fault. This turns out to be all twelve events included in the fault tree, which includes five seismic failures. In addition to the faults described below, random failures of the DGs and failure to isolate portions of the PCC are included. There are no double or triple faults for this case.

For the small LOCA case, only those events which lead to failure of both PORVs make up the single-order cut sets. There are nine of these faults, five of which are seismic. Seismic failure of the station transformers, PCC/SCC coolers, air conditioner chillers, circulation water pump house, DG day tanks, failure to refill the DG fuel tanks or DG common cause failure causes loss of power to both PORVs. Common cause failure of the PORVs, or operator failure to actuate feed and bleed, will also fail the system. There are two system double faults, both of which include only nonseismic events. Both faults lead to PORV failure as a result of support system failures. There are no system triplets.

3.3 Systems Analysis of Support Systems

The support systems which are required for operation of the Group A front-line systems described in Section 3.2 are primary and secondary component cooling water, service water, electric power and actuation. The analysis of these systems is similar to that for the front-line systems.

3.3.1 Component Cooling Water Systems

Component cooling water consists of the primary component cooling water (PCC) system and the secondary component cooling water (SCC) system. PCC and SCC provide the cooling required by plant equipment for normal operation, and decay heat removal during cooldown or accidents. The PCC and SCC are redundant systems, in that PCC will provide cooling for one train of a front-line system and SCC will provide cooling for the other train. Within the PCC and SCC systems, operation of at least one pump and one cooler are required for system success. Each system consists of a surge tank, two motor-driven pumps, two coolers (heat exchangers) and valves for isolating nonessential portions of the system. Complete lists of the components, valves, cooling requirements and cooling loads are in Appendix E for the PCC, and Appendix F for the SCC.

Schematics of the PCC and SCC systems are shown in Figures 3-4 and 3-5 respectively. One pump in each system (located in the turbine building at el. 21 ft) is normally operating. The standby pump will start automatically on a supply header low pressure signal. Upon loss of offsite power, the diesel generator load sequencer will start the pump which had been operating in each system, after a 10-second delay. If all four pumps were operating, only the preferred pumps (P-9A and P-10A) will be restarted. If a preferred pump fails to start, the alternate must be started manually. In each system, one cooler (also located in the turbine building at el. 21 ft) is normally in service. The cooler in standby has its cooling water outlet valve closed. As there must be flow through a cooler for system success, the cooler bypass valves (PCC-T-20 and SCC-T-23) must be closed. The cooler bypass and isolation valves are linked by a single operator and fail to the full cooling position. The cooling water flows from the coolers to the non-isolated loads and is returned to the pump suction headers.

Support systems required for component cooling water availability are 4160-V ac emergency power, 125-V dc power, and service water.

The following items provided the basis for the development of the fault tree logic:

- A new procedure change will instruct the operator to isolate portions of the PCC system following an earthquake if the PCC surge tank low level alarm is annunciated. This is accomplished by closing valves PCC-M-90, PCC-M-150, PCC-M-219, and PCC-A-268. It is assumed that failure or inability to close these valves will result in loss of PCC, due to a potential breach of the system pressure boundary. Only those loads which are not isolated were included in the PCC fault tree.

- The nonseismic loads in the SCC system are isolated by valves SCC-A-460 and SCC-A-461, which close automatically on a suction header low pressure signal, indicating a breach in the system pressure boundary. It is assumed that failure of these valves to close will result in loss of SCC. (Although it is on a return line, SCC-A-461 must close as there is no check valve to prevent backflow through the line.) Only those loads which are not isolated were included in the SCC system fault tree.

- All four pump motors are equipped with drip-proof shields. It is judged that these shields will provide protection from potential spray from the fire water lines located over the pumps if the lines break.

- It is assumed that control power for the motor-operated valves is transformed off the same bus which provides the valve motive power.

- Check valve failures, random valve failures, and failure due to plugging, testing or maintenance are of low probability and are excluded from the analysis.

- The instrument air supply for the SCC isolation valves is not modeled beyond the accumulator (TK-110) which is seismically designed. The accumulator inlet line contains two check valves to prevent depressurization, and TK-110 contains enough air to reposition the valves once and hold them closed for 24 hours.

- The chemical additive tank and supply line are not included as a potential flow diversion path in either system. The supply and return lines each contain a normally closed manual valve. (The PCC P&ID incorrectly shows these valves as normally open.)

- The gland leak-off tank (TK-93, 94), pump (P-112, 113), and filter (FL-69, 70) are not included as a potential flow diversion path in either system. Only a small amount of leak-off flow is present in comparison to cooling water flow.

- The surge tank vent, overflow and waste lines are not included as a potential flow diversion path in either system. The vent and overflow lines contain an automatic valve, and the waste line contains a normally closed manual valve.

- The recovery action of placing the standby cooler in service by opening the manual outlet valve is not included in the system fault trees. This action would not significantly change the plant HCLPF.

- The failure of the SCC line to the penetration coolers by the collapse of block wall VE 21-3, 4 is included in the SCC system fault tree.

The basic fault trees for the PCC and SCC systems are shown in Appendices E and F, respectively. These trees include all system components and nonisolated cooling loads, along with the required support system inputs. As in the front-line system fault trees, the types of failures were not specified. The top event of each tree represents loss of cooling to the required safety system components. This event may occur by common cause failure of the cooling water pumps, failure of a nonisolated cooling load (breach in system pressure boundary), failure of the required system isolation to occur, or failure within the system to provide flow. The last event may be caused by lack of flow through the in-service cooler, or by loss of both system pumps. There are several events on the tree which may lead to these faults.

The screening criteria described in Chapter 2 were applied to the cooling water system components in order to prune the PCC and SCC fault trees. The results of this screening are summarized in Tables 3-11 and 3-12 for PCC and SCC, respectively. The failure probability calculations and results are detailed in Section 3.4. Although the PCC and SCC coolers and the air conditioner chillers are located in separate systems, there is only one event for cooler seismic failure (CCW-HTX-EQ-4B5A) and one for chiller seismic failure (CCW-ACX-EQ-CHILL). This is because the coolers are located in the same area, and are identical components. The pruned cooling water fault trees were then merged into the front-line system trees. Therefore, the analysis of the merged front-line fault trees includes failures due to loss of PCC and SCC.

3.3.2 Service Water System

The service water (SW) system is not directly a support system for the Group A front-line systems, but is a support system for the support systems required by Group A systems. It provides cooling for the PCC and SCC systems. Operation of at least one SW pump is required for system success. For PCC success there must be SW flow to the PCC cooler in service, and for SCC success there must be SW flow to the SCC cooler in service. The service water system consists of four traveling screens, four motor-driven pumps, and manual valves to align service water flow to the PCC and SCC coolers. Lists of the SW components and cooling requirements are in Appendix G.

A drawing of the SW system is shown in Figure 3-6. Two of the pumps (located in the circulating water pump house, el. 7 ft) are normally operating, the two standby pumps must be placed in service manually when necessary. Upon loss of offsite power all four pumps receive a start signal from the diesel generator load sequencer, the pumps are interlocked so only one in each pair will run to prevent overloading (i.e., if P-29A starts, P-29C will trip off). One PCC and one SCC cooler are normally in service. The PCC and SCC coolers in standby have their SW outlet valves closed. Service water flows from the PCC and SCC coolers to the seal pit. Support systems required for SW availability are 480-V ac emergency power and 125-V dc power.

The following items provided the basis for the development of the fault tree logic:

- The capacity of the traveling screens is large enough that it is assumed to be unlikely for all four to be blocked badly enough to choke all four SW pumps. The screens are designed for the circulating water system, with much larger flows than the SW system. Also, the screens are heavily used only a few times a year, therefore they have been excluded from the analysis.

- The loss of all four SW pumps due to high water level in the pump house is judged to be improbable. An alarm is sounded at a 3-inch water level, and the circulating water pumps trip off at a 10-inch water level. As the SW pump motors are mounted above the pumps, a 10-inch water level is not threatening.

- All valves on the main SW lines, except for the standby cooler outlets and pump discharge header cross-tie, are assumed to be normally open.

- Check valve failures and plugging of manual valves are of low probability and excluded from the fault tree.

- The lines for the screen wash systems are not included as potential flow diversion paths, as they are less than one-third the size of the pump discharge header. Also the screen wash system and traveling screen motors are interlocked, with neither powered from an emergency bus. Thus upon loss of offsite power the screen wash system is inoperable.

- The mussel control pump and discharge line are not included as a potential flow diversion path, as it is normally isolated and only used in special operations.

- All sample lines, pumps and collection tanks are excluded as potential flow diversion paths. These lines are considerably less than one-third the size of the main SW lines.

- Recovery actions, such as placing a standby cooler in service, are not included in the fault tree. This does not significantly impact the HCLPF analysis.

- Collapse of the circulating water pump house will fail all four service water pumps.

The service water system fault tree is shown in Appendix G. This tree includes nonspecific faults for each of the SW components, along with the required support system interfaces. The top event, loss of service water flow through the PCC/SCC coolers, is caused by common cause failure of the SW pumps, or by a system failure which prevents flow to the coolers. The latter would be caused by isolation of all four coolers, or failure of all four service water pumps.

Screening techniques from Chapter 2 were used to prune the SW system fault tree. The results of this screening process are listed in Table 3-13. Of all the front-line and support systems analyzed, the service water system is unique in that all system component failures (seismic and nonseismic) were screened out, leaving only the power inputs and circulating water pump house failure. When support system fault trees were merged with the front-line system trees, the power inputs were pruned to eliminate circular logic (SW fails PCC failing DG-1A failing Bus 7 failing SW pump A). Therefore the only fault from the service water system in the front-line system cut sets is collapse of the pump house.

3.3.3 Electric Power System

The electric power system consists of the emergency ac power (ACP) system, the dc power (DCP) system, and the onsite electric power (OEP) system (i.e., the diesel generators). The OEP system provides power to the emergency buses of the ACP system upon loss of offsite power. The ACP system provides operating power to the plant equipment, feeds the DCP system battery chargers, and provides 120-V ac power to plant instrumentation. The DCP system provides 125-V dc power to the switchgear, vital bus inverters, instrumentation and controls. The OEP system consists of two diesel generators (DGs), each with a fuel oil supply system, air starting system, and distribution and control panels. The ACP system consists of two 4160-V ac emergency buses, two 4160 to 480-V station transformers, two 480-V ac emergency buses, six 480-V ac emergency motor control centers (MCCs), and four 120-V ac inverters. The DCP system consists of four 125-V dc buses, each with a station battery and battery charger, and four distribution panels. Complete lists of the ACP and DCP system components are included in Appendix H. The OEP system components and cooling requirements are also in Appendix H.

Simplified schematics of the ACP, DCP, and OEP systems are shown in Figures 3-7, 3-8, and 3-9. Both DGs (located in the turbine building auxiliary bay, el. 22 ft) are normally in standby. A DG is automatically started by a one-second loss of voltage on its associated emergency bus (Bus 5 for DG-1A, Bus 6 for DG-1B). A DG will also start on a 10-second low voltage condition in its associated bus, concurrent with a SIAS (this case was not included in the analysis). A separate fault tree is constructed for each DG and each MCC (MCC 7B and 7B1, and MCC 8B and 8B1 are treated as the same MCC as they are connected by a tie without a breaker). The 4160-V ac buses, 480-V ac buses, and 125-V dc buses are included within the MCC fault tree logic. The 120-V ac buses and inverters were excluded as they supply instrumentation and actuation systems which were screened out. Also, for solenoid-operated valves requiring 120-V ac power, the batteries which feed the inverters have been shown as the inputs to the front-line systems. Support systems for the electric power system are PCC and SCC for the DG coolers.

The following items provided the basis for developing the fault tree logic:

- Cooling for the switchgear, cable tray, and battery rooms is provided by fans FN-31 and FN-32. Upon loss of offsite power, ac power loads are shed and only some reloaded by the load sequencer. It is assumed that there is sufficient load reduction to not require switchgear cooling for a long period of time, so the fans were removed from the fault tree.

- The diesel generator room exhaust fans must be operating, and the air intake and exhaust dampers open. It is assumed the dampers are powered off the same bus as the DG room fan.

- Because the operator must make a periodic check of the DG day tank fuel level, it is judged that the tank will not be allowed to drain through a broken vent line without some preventive action taken.

- It is judged that flooding of the diesel rooms is of low probability. The curbs at the room entrances protect against external flooding from the Turbine Building and the threaded fire water piping in the rooms is normally dry. Heat and smoke are required to actuate the spray of aqueous foam.

- Replacement of the current lead-antimony station batteries with lead-calcium batteries will sufficiently raise their seismic capacity.

- Relay chatter is not included in the analysis based on the guidance in NUREG/CR-4334 and 4482.

- Nonseismic circuit breaker failure is of low probability and excluded from the analysis.

- The bus cross-tie breakers are physically constructed to prevent closing unless one of the bus normal feed breakers are open. Therefore no failures were postulated for spurious closure of the cross-tie breaker resulting in the failure of both buses.

- The tie from the 480-V ac buses to the inverters is a synchronizing tie, which ensures that the 120-V ac vital system is at the same frequency as the rest of the ACP system. The only power supply to the inverters is the batteries.

- Although there is a procedure for feeding a 125-V dc bus with an alternate battery (e.g., battery 3 to feed DC-1), this is a recovery action and is not included in the fault tree. Because the new batteries have high HCLPFs, this action is not significant to the plant HCLPF.

- Although the 480-V ac MCCs provide power to the battery chargers, this relationship is not shown on the fault tree as it would create circular logic (MCC 7A fails BC-1 failing DC-1 failing Bus 7 SWGR failing MCC 7A). For the purpose of starting the DGs, the required dc power is only available from the station batteries.

- The logic of the diesel generator load shed/sequencers is not developed in the fault trees. As long as the relays of the sequencer are not physically damaged, the sequencer may be reset from the main control board (MCB) or from local panels near the DGs. Once reset, loads on the sequencer may be started from the MCB. The relays are located in the main electrical panel in the control room.

- The diesel engine, generator, blower which cools the generator, air starting units, integral fuel tank and pumps, integral cooling water pumps, and lubricating oil system are treated as a single component in the fault trees (DG1A and DG1B).

- The external DG fuel oil systems are represented separately in the fault trees. Each system consists of a day tank, an auxiliary fuel oil transfer pump and an underground fuel oil storage tank.

- The external DG air starting tanks are represented separately in the fault trees. For each DG there are two air receiver banks, consisting of three compressed air tanks each. One bank is sufficient to start a DG.

- The auxiliary boiler fuel oil supply lines from the auxiliary fuel oil transfer pumps are not included as potential flow diversion paths. Each line contains a normally closed air-operated isolation valve which fails closed.

- Refilling the integral fuel tanks by opening the manual valves is included in the fault trees as this is an operating procedure performed approximately every 2 hours. Refilling the day tanks by placing the oil pumps in service is also included in the trees.

- Some recovery actions are not included in the fault trees. This includes opening the manual valves between air receiver banks, and opening the manual valves which allow fire water to supply the DG coolers.

- The DG-1B cooler SCC outlet valve, SCC-T-305, is physically held open to provide continual flow through the cooler.

The fault trees for MCC 7A, Bus 7, Bus 5, and dc Bus 1, for MCC 8A, Bus 8, Bus 6 and dc Bus 3, and for DG-1A are shown in Appendix H. The trees for MCC 7B, MCC 8B, and DG-1B are analogous to these. The top events of the MCC trees are failure of the MCC, due to faults within the MCC or failure of the 480-V bus. Loss of the 480-V bus is due to faults within the bus, failure of the station transformer or failure of the 4160-V bus. Loss of the 4160-V bus is due to faults within the bus, failure of the DG, failure of the load sequencer, insufficient switchgear cooling or loss of the dc bus. Failure of the dc bus is caused by faults within the bus, failure of the battery or failure of the battery charger.

The top event of the DG fault trees is loss of power output from the diesel generator, due to failure of the output or bus cross-tie breakers, failure of the DG control and distribution panels, loss of dc power required to start the DG, or failure of the DG. Failure of the DG is caused by faults within the diesel generator, failure of the fuel oil supply, failure of the DG start signal (caused by loss of bus voltage), failure of the air starting banks, failure of the room fan or dampers, or insufficient cooling of the engine.

The screening criteria were applied to the electric power system components to prune the fault trees. The results of this process are summarized in Table 3-14. The failure probability calculations and results used are detailed in Section 3.4. The pruned electric power system fault trees were then merged with the front-line system trees in order to determine the front-line system minimal cut sets. Faults from the electric power systems which will lead to front-line system failure are included in the front-line system cut set tables in Section 3.2.

3.3.4 Actuation Systems

Various actuation systems are required to initiate engineered safeguards systems, these include the safety injection actuation system (SIAS), the containment isolation system (CIS), the containment spray actuation system (CSAS), the recirculation actuation system (RAS), and the reactor protection system (RPS). The AFW actuation system was included with the AFW system. Of these, SIAS is the only system required by the evaluated Group A systems. The SIAS consists of two trains both of which are actuated by the same four instrument channels (2/4 logic). These channels consist of instruments which measure pressurizer pressure and containment pressure. SIAS is actuated on low-low pressurizer pressure or high containment pressure. A list of instrumentation for the actuation systems is shown in Appendix I.

Loss of power within a logic channel results in a channel trip. Loss of power in the actuation logic leads to an actuation signal. Loss of power also results in the disenabling of the SIAS block. For these reasons power inputs were not included in the SIAS fault trees. As sufficient separation of the instrument channels exists, the system was not modeled down to the sensor and relay level (undeveloped events).

The fault tree for SIAS Train A is shown in Appendix I. The fault tree for Train B is analogous to this. Operator actions to unblock and initiate SIAS are included in the trees. Since there is sufficient separation of the required instrumentation and there is operator ability to actuate the system, automatic actuation of SIAS is not judged to be an issue. Therefore, the SIAS inputs were pruned from the front-line and support system fault trees and the SIAS fault trees were not further developed.

3.4 Probability Calculations

The data for the probability calculations of the nonseismic common cause, random, test and maintenance, and operator failures are shown in Table 3-15. Unless otherwise stated, mean beta factors from EPRI NP-3967 (Fleming, et al., 1985) and mean unavailabilities and failure rates from ASEP were used in the calculations. The results were later converted to median values based on the following:

Error Factor	Mean to Median Conversion
3	1.26
5	1.6
10	2.66

$$\text{Median} = \frac{\text{Mean}}{\text{Conversion}}$$

The following items were used as the basis for the calculations:

- The automatic valves in the AFW system are air-operated, for which a beta factor was not found. These valves fail open on loss of air or power so the probability of common cause failure was judged to be below the screening cutoff value.

- Steam binding of the three AFW pumps was the major common cause failure mode considered.

- The HPSI common cause failure calculation does not include the spare pump, P-14S. Common cause failure of the three pairs of injection valves is below the cutoff value. Common cause failure of the five remaining valve pairs (RWST outlet, RHR heat exchanger isolation, charging line isolation, pump discharge, and recirculation sump valves) was included in the calculation.

- The time used in determining HPSI pump fail to run (FTR) (12 hours) is based on the expectation that after this time the LPSI pumps would be placed in sevice, or the auxiliary charging pump could be used.

- A beta factor of 0.08 was used for the PORVs. The beta factor for a PWR safety relief valve is 0.07, and for an MOV it is 0.08.

- The DG auxiliary fuel oil pumps are small (low head), but are located in a relatively harsh environment. Therefore a beta factor of 0.11 was used. The probability of failure to run for these pumps is insignificant.

- The failure to start figure for random failure of the DGs is taken from ORNL data (Battle, 1985). The testing and maintenance calculation reflects the estimate by Maine Yankee personnel of six days of T&M time per year.

- To place the turbine-driven AFW pump in service the operator must open the steam supply valves and monitor the flow. The time allowed to do this is roughly 45 minutes. It is assumed that after 20 to 30 minutes of attempting to start the pump from the main control room (MCR) he would attempt to do so locally. The human error probability (HEP) for starting the pump from the control room is 0.05.

- The operator may attempt to start the turbine-driven AFW pump locally after a hardware or actuation/power failure. It is assumed to take about 10 minutes to diagnose failure of the motor-driven pump, 10 minutes to attempt starting the TDP from the MCR, and 10 minutes to get to the TDP to start it locally. The operator then has roughly 10 minutes left to start the pump. The HEP for 5 to 10 minutes is 0.25, and for 10 to 20 minutes is 0.10; the average is 0.18.

- To align the PWST to the motor-driven AFW pumps, the pumps must be tripped to prevent cavitation (5 to 10 minutes), and the manual valves from the PWST opened and the pumps restared before the SGs empty (30 to 40 minutes). Isolation of the DWST is not required due to the placement of check valves. The HEP for tripping the pumps is 0.25, and for opening the valves and restarting the pumps it is 0.03, for a total of 0.28.

- The operator actions for feed and bleed are guided by the ERGs and FRGs. Various studies show the response time to be 15 to 45 minutes. Because no plant specific study was available, the 20 to 30 minute HEP was used (0.05). This valve is higher than most other feed and bleed estimates (ASEP gives approximately 0.01), but the operators will first attempt to start the AFW pumps, and it is an earthquake situation.

- The new procedure for PCC isolation calls for the operator to close four valves from the MCR in the event of an earthquake with low level alarm on the PCC surge tank. If there are PCC ruptures inside containment, the operator may not have much time to perform this action, and indications other than the surge tank level alarm may not be checked or available. A 10- to 20-minute time period, with a 0.1 HEP, is used. Recovery is not considered likely.

- Appoximately every 2 hours the operator must refill the DG integral fuel tanks, and occasionally refill the day tanks from the underground storage tanks. The steps for this are included in the loss of offsite power procedures and are simple, but must be performed local to the DGs. Failure to do so would cause the DG to stop, making it necessary to reprime and restart. Because the procedure is not difficult to perform, and the failure is recoverable, a probability of 0.01 is used.

Table 3-1. Front-line system vs support system dependency matrix.

	SUPPORT SYSTEMS								
	AC Power		DC Power		CCW		SIAS		IA
FRONT-LINE SYSTEMS	Bus 5 Bus 7	Bus 6 Bus 8	DC-1 DC-2	DC-3 DC-4	PCC	SCC	Train A	Train B	TK-25
HPSI/ RECIRC									
P-14A to RCS	X		X		X		X		
P-14B to RCS		X		X		X		X	
FN-44A	X						X		
FN-44B		X						X	
AFW									
P-25A		X		X					
P-25B									X
P-25C	X		X						
PORVs									
PR-S-14	X								
PR-S-15		X							

Note: To determine the front-line system dependencies on the support systems, locate the front-line component in the first column and read across the row to find the support system dependencies.

Table 3-2. Support system vs support system dependency matrix.

			AC Power										DC Power													
			4160V Bus		480V Bus		120V Vital Bus				DG		125V Bus				CCW		SW	SIAS		HVAC				1A
SUPPORT SYSTEM			5	6	7	8	1	2	3	4	1A	2A	1	2	3	4	PCC	SCC		A	B	FN-20A	FN-20B	FN-31	FN-32	TK-110
AC Power	4160V Bus	5									X		X											X	X	
		6										X			X									X	X	
	480V Bus	7	X										X											X	X	
		8		X											X									X	X	
	120V Vital Bus	1											X													
		2												X												
		3													X											
		4														X										
	Diesel Generator	1A											X				X					X				
		1B													X			X					X			
DC Power	125V Bus	1			X																			X	X	
		2			X																			X	X	
		3				X																		X	X	
		4				X																		X	X	
CCW		PCC	X																X	X	X					
		SCC		X															X							X
SWS					X	X																				
SIAS	Channel	A											X													
		B												X												
		C													X											
		D														X										
	Act.	SIS-A											X													
		SIS-B													X											
HVAC		FN-20A			X																					
		FN-20B				X																				
		FN-31			X																					
		FN-32				X																				

Note: To determine the support system dependencies on other support systems, locate the support system in the first column, and read across the row to determine dependencies on the other support systems.

Table 3-3. AFW screening overview table.

Component	Screening Logic	Event[1]	Value[2]
AFW-A-101	Out-High HCLPF	AFW-AOV-XX-101	>0.3g
AFW-A-201	Out-High HCLPF	AFW-AOV-XX-201	>0.3g
AFW-A-301	Out-High HCLPF	AFW-AOV-XX-301	>0.3g
AFW-A-338	Out-High HCLPF	AFW-AOV-XX-338	>0.3g
AFW-A-339	Out-High HCLPF	AFW-AOV-XX-339	
AFW-A-340	Out-High HCLPF	AFW-AOV-XX-340	
E-86A	Out-Integral to P-25A	---	--
E-86B	Out-Integral to P-25B		
E-86C	Out Integral to P-25C		
MS-A-162	Out-Group Not-A (Atm. Steam Dump)	---	--
MS-A-173	Out-High HCLPF	AFW-AOV-XX-A173	>0.3g
MS-M-161	Out-Group Not-A (Atm. Steam Dump)	---	--
MS-M-255	Out-Group Not-A (Aux. Steam)	---	--
MS-P-168	Out-High HCLPF	AFW-PCV-XX-P168	>0.3g
MS-T-163	Out-High HCLPF	AFW-PCV-XX-T163	>0.3g
P-25A	Out-High HCLPF	AFW-MDP-XX-PTRNA	>0.5g
P-25B	In-Random (High HCLPF)	AFW-TDP-LF-P25B	8.0E-2
P-25C	Out-High HCLPF	AFW-MDP-XX-PTRNC	>0.5g
T-1	Out-Considered with TDP P-25B	---	--
TK-16	In-Low HCLPF	AFW-TNK-EQ-PWST	0.27
TK-21	In-Low HCLPF	AFW-TNK-EQ-DWST	0.17
TK-25 (Accum.)	In-Unknown HCLPF	AFW-TNK-EQ-TK25	NA (>0.3g)
TK-111 (Accum.)	Out-AOVs fail open on loss of air	---	--
TK-123 (Accum.)	Out-AOVs fail open on loss of air	---	--
System	In-Common Cause failure for pumps and AOVs	AFW-CCF-FC-AFW	2.0E-4

NA indicates that a HCLPF was not available at the time of screening, and the component remained in the analysis. The value in parenthesis is the HCLPF that was available later.

1, 2 -- See Table 3-4 for standard footnotes on the screening overview tables.

Table 3-4. Standard footnotes for screening overview tables.

Footnote Number	
1	Events using the XX cause code are shown in the basic system fault trees. Events using other cause codes (EQ, FC, etc.) are shown in the pruned system fault trees. Components with no event listed were screened out without being added to a fault tree. Appendix A contains the definitions of symbols and abbreviations.
2	Values shown are the preliminary HCLPF capacity for seismic failures, or the calculated probability of failure for nonseismic failures.
3	Components which make up the various actuation systems, which were screened out based on separation and operator ability to actuate manually. These systems included: SIAS-A, B (SIS-ACT-FA-TRMA, B) Steam Generator Level (RPS-ACT-FA-SGLEV) PCC Pump Discharge Pressure (PCC-PST-FA-ACTPP) SCC Pump Discharge Pressure (SCC-PST-FA-ACTPP) SCC Header Isolation Pressure (SCC-PST-FA-1750A, B) DG Load Shed/Sequencer (OEP-DGN-VF-A, BLDSQ) DG Initiation - Bus Voltage (OEP-ACT-FA-BUS5, 6)
NA	Indicates that a HCLPF was not available at the time of screening, and the component remained in the analysis. The value in parentheses is the HCLPF that was available later.

Table 3-5. AFW system cut sets.

Ranked Single Faults	Nonseismic Probability
AFW-CCF-FC-AFW	2.0E-04

Ranked Double Faults		Nonseismic Probability
AFW-TNK-EQ-PWST	AFW-TNK-EQ-DWST	SF
ACP-TFM-EQ-57X68	AFW-TNK-EQ-DWST	SF
CCW-HTX-EQ-4B5A	AFW-TNK-EQ-DWST	SF
CCW-ACX-EQ-CHILL	AFW-TNK-EQ-DWST	SF
SWS-BKW-EQ-CIRC	AFW-TNK-EQ-DWST	SF
OEP-TNK-EQ-TK62X	AFW-TNK-EQ-DWST	SF
AFW-XHE-FO-EFWXX	AFW-TNK-EQ-DWST	2.8E-01
AFW-XHE-FO-TRBMC	ACP-TFM-EQ-57X68	5.0E-02
ACP-TFM-EQ-57X68	AFW-TDP-LF-P25B	5.0E-02
CCW-HTX-EQ-4B5A	AFW-TDP-LF-P25B	5.0E-02
CCW-ACX-EQ-CHILL	AFW-TDP-LF-P25B	5.0E-02
SWS-BKW-EQ-CIRC	AFW-TDP-LF-P25B	5.0E-02
OEP-TNK-EQ-TK62X	AFW-TDP-LF-P25B	5.0E-02
CCW-HTX-EQ-4B5A	AFW-XHE-FO-TRBMC	5.0E-02
CCW-ACX-EQ-CHILL	AFW-XHE-FO-TRBMC	5.0E-02
SWS-BKW-EQ-CIRC	AFW-XHE-FO-TRBMC	5.0E-02
OEP-TNK-EQ-TK62X	AFW-XHE-FO-TRBMC	5.0E-02
OEP-XHE-FO-FUEL	AFW-TNK-EQ-DWST	1.0E-02
OEP-CCF-FC-DGN	AFW-TNK-EQ-DWST	2.6E-03
OEP-XHE-FO-FUEL	AFW-TDP-LF-P25B	5.0E-04
AFW-XHE-FO-TRBMC	OEP-XHE-FO-FUEL	5.0E-04
OEP-CCF-FC-DGN	AFW-TDP-LF-P25B	1.3E-04
OEP-CCF-FC-DGN	AFW-XHE-FO-TRBMC	1.3E-04

SF indicates that the cut set events are seismic failures, and their HCLPFs will be provided by the fragility analysis team.

Table 3-6. HPSI/CSPPCL screening overview table.

Component	Screening Logic	Event[1]	Value[2]
CH-A-32	Out-High HCLPF	HPI-AOV-XX-CHA32	>0.3g
CH-A-33	Out-High HCLPF	HPI-AOV-XX-CHA33	>0.3g
CH-F-38	Out-Isolated by CH-A-32, 33	HPI-MOV-XX-CHF38	--
CH-M-1	Out-Path ruled out as flow diversion	---	--
CH-M-87	Out-Path ruled out as flow diversion	---	--
CS-M-1	Out-Group Not-A (Recirculation)	---	--
CS-M-2	Out-Group Not-A (Recirculation)	---	--
CS-M-91	Out-Group Not-A (Recirculation)	---	--
CS-M-92	Out-Group Not-A (Recirculation)	---	--
E-34	Out-Path ruled out as flow diversion	---	--
E-67	Out-Isolated by CH-A-32, 33	HPI-HTX-XX-REGEN	--
E-96	Out-Isolated by CH-A-32, 33	HPI-HTX-XX-SLWTR	--
FL-34B	Out-Isolated by CH-A-32, 33	HPI-FLT-XX-SLWTR	--
FN-44A	Out-High HCLPF	CSS-FAN-XX-FN44A	>0.3g
FN-44B	Out-High HCLPF	CSS-FAN-XX-FN44B	>0.3g
HSI-M-11	Out-High HCLPF	HPI-MOV-XX-MV11	>0.3g
HSI-M-12	Out-High HCLPF	HPI-MOV-XX-MV12	>0.3g
HSI-M-21	Out-High HCLPF	HPI-MOV-XX-MV21	>0.3g
HSI-M-22	Out-High HCLPF	HPI-MOV-XX-MV22	>0.3g
HSI-M-31	Out-High HCLPF	HPI-MOV-XX-MV31	>0.3g
HSI-M-32	Out-High HCLPF	HPI-MOV-XX-MV32	>0.3g
HSI-M-41	Out-High HCLPF	HPI-MOV-XX-MV41	>0.3g
HSI-M-42	Out-High HCLPF	HPI-MOV-XX-MV42	>0.3g
HSI-M-40	Out-Recovery only	---	--
HSI-M-43	Out-Recovery only	---	--
HSI-M-50	Out-High HCLPF	HPI-MOV-XX-MV50	>0.3g
HSI-M-51	Out-High HCLPF	HPI-MOV-XX-MV51	>0.3g
HSI-M-54	Out-Group Not-A (Recirculation)	---	--
HSI-M-55	Out-Group Not-A (Recirculation)	---	--

Table 3-6 (Cont'd)

Component	Screening Logic	Event[1]	Value[2]
LSI-M-40	Out-Group Not-A (LPSI)	---	--
LSI-M-41	Out-Group Not-A (LPSI)	---	--
P-14A	Out-High HCLPF	HPI-MDP-XX-PTRNA	>0.5g
P-14B	Out-High HCLPF	HPI-MDP-XX-PTRNB	>0.5g
P-14S	Out-Spare, recovery only	---	--
P-61A	Out-Group Not-A (Recirculation)	---	--
P-61B	Out-Group Not-A (Recirculation)	---	--
P-61S	Out-Group Not-A (Recirculation)	---	--
SL-P-3	Out-Isolated by CH-A-32, 33	HPI-MOV-XX-SLP3	--
TK-4	In-Low HCLPF	HPI-TNK-EQ-RWST	0.21g
TK-54	Out-High HCLPF	HPI-TNK-XX-SCAT	0.59g
VE 21-1, 2	Out-Although failure impacts FN-44A, B, upgraded wall is of sufficient capacity	---	>0.5g
System	In-Common cause failure for pumps & MOVs	HPI-CCF-FC-HPSI	2.1E-3

1, 2 - See Table 3-4 for standard footnotes on the screening overview tables.

Table 3-7. HPSI system cut sets.

Ranked Single Faults	Nonseismic Probability
ACP-TFM-EQ-57X68	SF
HPI-TNK-EQ-RWST	SF
CCW-HTX-EQ-4B5A	SF
CCW-ACX-EQ-CHILL	SF
SWS-BKW-EQ-CIRC	SF
OEP-TNK-EQ-TK62X	SF
OEP-XHE-FO-FUEL	1.0E-02
OEP-CCF-FC-DGN	2.6E-03
OEP-CCF-FC-HPSI	2.1E-03

Ranked Double Faults		Nonseismic Probability
PCC-XHE-FO-ISOL	OEP-PSF-FC-DG1B	6.7E-03
OEP-PSF-FC-DG1B	OEP-PSF-FC-DG1A	4.5E-03

SF indicates that the cut set events are seismic failures, and their HCLPFs will be provided by the fragility analysis team.

Table 3-8. PORVs screening overview table.

Component	Screening Logic	Event[1]	Value[2]
E-2	Out- Not required for Group A (Feed & Bleed)	---	--
PR-M-16	Out-High HCLPF	PPS-MOV-XX-PRM16	>0.3g
PR-M-17	Out-High HCLPF	PPS-MOV-XX-PRM17	>0.3g
PR-S-11	Out-Not required for feed and bleed	---	--
PR-S-12	Out-Not required for feed and bleed	---	--
PR-S-13	Out-Not required for feed and bleed	---	--
PR-S-14	Out-High HCLPF	PPS-SOV-XX-PRS14	>0.3g
PR-S-15	Out-High HCLPF	PPS-SOV-XX-PRS15	>0.3g
System	In-Common cause failure for PORVs and block valves	PPS-CCF-FC-PORVs	8.0E-4

1, 2 - See Table 3-4 for standard footnotes for the screening overview tables.

Table 3-9. PORV (no LOCA) cut sets.

Ranked Single Faults	Probability
CCW-HTX-EQ-4B5A	SF
CCW-ACX-EQ-CHILL	SF
SWS-BKW-EQ-CIRC	SF
OEP-TNK-EQ-TK62X	SF
ACP-TFM-EQ-57X68	SF
PCC-XHE-FO-ISOL	1.0E-01
OEP-PSF-FC-DG1A	6.7E-02
OEP-PSF-FC-DG1B	6.7E-02
PPS-XHE-FO-FDBLD	5.0E-02
OEP-XHE-FO-FUEL	1.0E-02
OEP-CCF-FC-DGN	2.6E-03
PPS-CCF-FC-PORVS	8.0E-04

No Double Faults

SF indicates that the cut set events are seismic failures, and their HCLPFs will be provided by the fragility analysis team.

Table 3-10. PORV (small LOCA) cut sets.

Ranked Single Faults	Nonseismic Probability
CCW-HTX-EQ-4B5A	SF
CCW-ACX-EQ-CHILL	SF
SWS-BKW-EQ-CIRC	SF
OEP-TNK-EQ-TK62X	SF
ACP-TFM-EQ-57X68	SF
PPS-XHE-FO-FDBLD	5.0E-02
OEP-XHE-FO-FUEL	1.0E-02
OEP-CCF-FC-DGN	2.6E-03
PPS-CCF-FC-PORVS	8.0E-04

Ranked Double Faults		Nonseismic Probability
PCC-XHE-FO-ISOL	OEP-PSF-FC-DG1B	6.7E-03
OEP-PSF-FC-DG1A	OEP-PSF-FC-DG1B	4.5E-03

SF indicates that the cut set events are seismic failures, and their HCLPFs will be provided by the fragility analysis team.

Table 3-11. PCC screening overview table.

Component	Screening Logic	Event[1]	Value[2]
AC-1B	In-Unknown HCLPF of chiller	CCW-ACX-EQ-CHILL	NA (0.38g)
E-3A	Out-High HCLPF	PCC-HTX-XX-E3A	>0.3g
E-4A	Out-Standby, recovery only	---	--
E-4B	In-Unknown HCLPF	CCW-HTX-EQ-4B5A	NA (0.31g)
E-25	Out-High HCLPF	PCC-HTX-XX-E25	>0.3g
E-54-1 thru 6	Out-High HCLPF	PCC-HTX-XX-E54X	>0.3g
E-82A	Out-High HCLPF	PCC-HTX-XX-E82A	>0.3g
E-91B	Out-High HCLPF	PCC-HTX-XX-E91B	>0.5g
E-92B	Out-High HCLPF	PCC-HTX-XX-E92B	>0.3g
P-9A	Out-High HCLPF	PCC-MDP-XX-PTRNA	>0.5g
P-9B	Out-High HCLPF	PCC-MDP-XX-PTRNB	>0.5g
P-7 Cooler	Out-High HCLPF	PCC-HTX-XX-P7	>0.5g
P-12A Cooler	Out-High HCLPF	PCC-HTX-XX-P12A	>0.5g
P-14A Cooler	Out-High HCLPF	PCC-HTX-XX-P14A	>0.5g
P-14S Cooler	Out-High HCLPF	PCC-HTX-XX-P14S	>0.5g
P-61A Cooler	Out-Integral to pump	---	--
P-61S Cooler	Out-Integral to pump	---	--
Penetration Coolers	Out-High HCLPF	PCC-HTX-XX-PEN	>0.5g
PCC-A-53	Out-Return line isolation only	---	--
PCC-A-216	Out-High HCLPF	PCC-AOV-XX-AV216	>0.3g
PCC-A-238	Out-Return line isolation only	---	--
PCC-A-268	Out-High HCLPF	PCC-AOV-XX-AV268	>0.3g
PCC-A-270	Out-Replaced in procedure by PCC-A-268	---	--
PCC-M-43	Out-Return line isolation only	---	--
PCC-M-90	Out-High HCLPF	PCC-MOV-XX-MV90	>0.3g
PCC-M-150	Out-High HCLPF	PCC-MOV-XX-MV150	>0.3g
PCC-M-219	Out-High HCLPF	PCC-MOV-XX-MV219	>0.3g

Table 3-11 (Cont'd)

Component	Screening Logic	Event[1]	Value[2]
PCC-T-19	Out-High HCLPF	PCC-TCV-XX-TCV19	>0.3g
PCC-T-20	Out High HCLPF	PCC-TCV-XX-TCV20	>0.3g
TK-5	Out-High HCLPF	PCC-TNK-XX-SRGTK	>0.5g
System	Out-Common cause failure of pumps below cutoff (0.001)	PCC-CCF-FC-PCCW	1.4E-4

NA indicates that a HCLPF was not available at the time of screening, and the component remained in the analysis. The value in parentheses is the HCLPF that was available later.

1, 2 - See Table 3-4 for standard footnotes on the screening overview tables.

Table 3-12. SCC screening overview table.

Component	Screening Logic	Event[1]	Value[2]
AC-1A	In-Unknown HCLPF of chiller	CCW-ACX-EQ-CHILL	NA (0.38g)
AC-2	In-Unknown HCLPF of chiller	CCW-ACX-EQ-CHILL	NA (0.38g)
E-3B	Out-High HCLPF	SCC-HTX-XX-E3B	>0.3g
E-5A	In-Unknown HCLPF	CCW-HTX-EQ-4B5A	NA (0.31g)
E-5B	Out-Standby, recovery only	---	--
E-82B	Out-High HCLPF	SCC-HTX-XX-E82B	>0.3g
E-91A	Out-High HCLPF	SCC-HTX-XX-E91A	>0.5g
E-92A	Out-High HCLPF	SCC-HTX-XX-E92A	>0.3g
P-10A	Out-High HCLPF	SCC-HTX-XX-PTRNA	>0.5g
P-10B	Out-High HCLPF	SCC-HTX-XX-PTRNB	>0.5g
P-12B Cooler	Out-High HCLPF	SCC-HTX-XX-P12B	>0.5g
P-14B Cooler	Out-High HCLPF	SCC-HTX-XX-P14B	>0.5g
P-61B Cooler	Out-Integral to pump	---	--
Penetration Coolers	Out-High HCLPF	SCC-HTX-XX-PEN	>0.5g
SCC-A-460	Out-High HCLPF	SCC-AOV-XX-AV460	>0.3g
SCC-A-461	Out-High HCLPF	SCC-AOV-XX-AV461	>0.3g
SCC-T-23	Out-High HCLPF	SCC-TCV-XX-TCV23	>0.3g
SCC-T-24	Out-High HCLPF	SCC-TCV-XX-TCV24	>0.3g
TK-59	Out-High HCLPF	SCC-TNK-XX-SRGTK	>0.5g
VE 21-3, 4	Out-Although failure impacts SCC line to penetration coolers, wall has high HCLPF	---	>0.5g
System	Out-Common cause failure of pumps below cutoff (0.001)	SCC-CCF-FC-SCCW	1.4E-4

NA indicates that a HCLPF was not available at the time of screening, and the component remained in the analysis. The value in parentheses is the HCLPF that was available later.

1, 2 - See Table 3-4 for standard footnotes on the screening overview tables.

Table 3-13. SWS screening overview table.

Component	Screening Logic	Event[1]	Value[2]
Circ. Water Pump House	In-Low HCLPF	SWS-BKW-EQ-CIRC	0.30g
E-4A	Out-standby, recovery only	---	--
E-4B	Out-Seismic failure considered for CCW	SWS-HTX-XX-E4B	--
E-5A	Out-Seismic failure considered for CCW	SWS-HTX-XX-E5A	--
E-5B	Out-Standby, recovery only	---	--
P-29A	Out-High HCLPF	SWS-MDP-XX-PTRNA	>0.5g
P-29B	Out-High HCLPF	SWS-MDP-XX-PTRNB	>0.5g
P-29C	Out-High HCLPF	SWS-MDP-XX-PTRNC	>0.5g
P-29D	Out-High HCLPF	SWS-MDP-XX-PTRND	>0.5g
System	Out-Common cause failure of pumps below cutoff (0.001)	SWS-CCF-FC-PUMPS	1.4E-4

1, 2 - See Table 3-4 for standard footnotes on screening overview tables.

Table 3-14. Electric power screening overview table.

Component	Screening Logic	Event[1]	Value[2]
120 VAC Bus 1 thru 4	Out-Included with MCB	---	--
120 VAC Bus 1A thru 4A	Out-High HCLPF	---	>0.3g
125 VDC Bus 1 thru 4	Out-High HCLPF	DCP-BDC-XX-BUS1 (etc.)	>0.3g
125 VDC Cab. DC/CE-1	Out-High HCLPF	---	>0.5g
125 VDC Cab. DC/CE-2	Out-High HCLPF	---	>0.5g
125 VDC Panel DP/P	Out-High HCLPF	---	>0.5g
125 VDC Panel DP/BU	Out-High HCLPF	---	>0.5g
480 VAC MCC 7A,7B,7B1	Out-High HCLPF	ACP-PSF-LP-MCC7A(B)	>0.5g
480 VAC MCC 8A,8B,8B1	Out-High HCLPF	ACP-PSF-LP-MCC8A(B)	>0.5g
480 VAC Bus 7,8	Out-High HCLPF	ACP-PSF-LP-BUS7(8)	>0.5g
4160 VAC Bus 5,6	Out-High HCLPF	ACP-PSF-LP-BUS5(6)	>0.5g
BATT-1 thru 4	Out-Assumed to be upgraded	DCP-BAT-XX-BAT1(etc.)	>0.3g
	low common cause failure	DCP-CCF-LP-BATT	1.0E-4
BC-1 thru 4	Out-High HCLPF	DCP-BAT-XX-BCH1(etc.)	>0.3g
INVR-1 thru 4	Out-High HCLPF	---	0.34g
Transformer X-507	In-Low HCLPF	ACP-TFM-EQ-57X68	0.3g
Transformer X-608	In-Low HCLPF	ACP-TFM-EQ-57X68	0.3g
Bus Cross-Tie Breakers	Out-High HCLPF	ACP-BKR-CC-3T5	>0.3g
	Out-High HCLPF	ACP-BKR-CC-4T6	>0.3g
Main Control Board	Out-High HCLPF	MCR-ACT-XX-CNTRL	>0.3g
Electrical Control Board	Out-High HCLPF	---	>0.5g
Aux. Logic Cabinets	Out-High HCLPF	---	>0.5g
ESF Aux. Panels	Out-High HCLPF	---	>0.5g
Air Cond. Control Panel	Out-High HCLPF	---	>0.5g

Table 3-14 (Cont'd)

Component	Screening Logic	Event[1]	Value[2]
SPDS Cabinets	Out-High HCLPF	---	>0.5g
Instrument Racks[3]	Out-High HCLPF	---	>0.5g
Cable Trays[3]	Out-High HCLPF	---	>0.5g
Impulse Lines[3]	Out-Based on train and channel separation	---	--
DG-1A	In-Random and common cause failures, High HCLPF	OEP-PSF-FC-DG1A OEP-CCF-FC-DGN	6.7E-2 2.6E-3
DG-1B	In-Random and common cause failures, High HCLPF	OEP-PSF-FC-DG1B OEP-CCF-FC-DGN	6.7E-2 2.6E-3
DG Output Breakers	In-Combined with DG failure	OEP-BKR-OO-ACB1A(B)	--
DG-1A,B Engine Control Panels	Out-High HCLPF	OEP-ACT-XX-1A(B)CTL	>0.3g
DG-1A,B Distribution Panels	Out-High HCLPF	OEP-ACT-XX-1A(B)CTL	>0.5g
DG-1A,B Control Panels	Out-High HCLPF	OEP-ACT-XX-1A(B) CTL	>0.5g
DG-1A,B Air Inlet and Exhaust Dampers	Out-High HCLPF	OEP-SOD-XX-AIRA(B) OEP-MOD-XX-EXHA(B)	>0.5g >0.5g
FN-20A	Out-High HCLPF	OEP-FAN-XX-FN20A	>0.5g
FN-20B	Out-High HCLPF	OEP-FAN-XX-FN20B	>0.5g
FN-31	Out-Switchgear load sufficiently decreased, fan cooling no longer required	ACP-FAN-XX-FN31	--
FN-32	Out-Switchgear load sufficiently decreased, fan cooling no longer required	ACP-FAN-XX-FN32	--

Table 3-14 (Cont'd)

Component	Screening Logic	Event[1]	Value[2]
P-33A	Out-High HCLPF	OEP-MDP-XX-P33A	>0.5g
P-33B	Out-High HCLPF	OEP-MDP-XX-P33B	>0.5g
PCC-A-493	Out-High HCLPF	OEP-AOV-XX-AV493	>0.3g
SCC-T-305	Out-Chained open	OEP-AOV-XX-AV305	--
SB 39-1	Out-Seismic failure impacts Group Not-A components (FN-7A,B)	---	0.35g
SB 39-2	Out-Seismic failure impacts components which were screened out (FN-32 exhaust duct)	---	--
SB El.39 floor	Out-High HCLPF	---	0.35g
TK-28A	Out-High HCLPF	OEP-TNK-XX-TK28A	>0.5g
TK-28B	Out-High HCLPF	OEP-TNK-XX-TK28B	>0.5g
TK-62A	In-Unknown HCLPF	OEP-TNK-EQ-TK62X	NA (0.43g)
TK-62B	In-Unknown HCLPF	OEP-TNK-EQ-TK62X	NA (0.43g)
TK-76A-1 thru 6	Out-High HCLPF	OEP-TNK-XX-76A-1(etc.)	>0.5g
TK-76B-1 thru 6	Out-High HCLPF	OEP-TNK-XX-76B-1(etc.)	>0.5g

NA indicates that a HCLPF was not available at the time of screening, and the component remained in the analysis. The value in parentheses is the HCLPF that was available later.

1, 2, 3 - See Table 3-4 for standard footnotes on screening overview tables.

Table 3-15. Nonseismic event failure probability calculations.

Event	Beta	Mean Unavailability or Failure Rate	Time(hr)	Probability
AFW-CCF-FC-AFW	--	1.0E-4 (steam binding)	--	1.0E-4
HPI-CCF-FC-HPSI	0.08 (MOV)	3.8E-3 (MOV FTO)	--	
	0.17 (MDP)	3.2E-3 (MDP FTS)	--	
		5.3E-5/hr (MDP FTR)	12	2.1E-3
PPS-CCF-FC-PORVs	0.08 (MOV)	3.8E-3 (MOV FTO)	--	
	0.08 (SOV)	6.3E-3 (SOV FTO)	--	8.0E-4
PCC-CCF-FC-PCCW	0.03 (MDP)	3.2E-3 (MDP FTS)	--	
		5.3E-5/hr (MDP FTR)	24	1.4E-4
SCC-CCF-FC-SCCW	0.03 (MDP)	3.2E-3 (MDP FTS)	--	
		5.3E-5/hr (MDP FTR)	24	1.4E-4
SWS-CCF-FC-PUMPS	0.03 (MDP)	3.2E-3 (MDP FTS)	--	
		5.3E-5/hr (MDP FTR)	24	1.4E-4
OEP-CCF-FC-DGN	0.05 (DG)	2.1E-2 (DG FTS)	--	
		1.26E-3/hr (DG FTR)	24	2.6E-3
OEP-CCF-FC-P33X	0.11 (RHR pumps)	3.2E-3 (MDP FTS)	--	3.5E-4
AFW-TDP-LF-P25B	--	1.6E-2 (TDP T&M)	--	
		3.2E-2 (TDP FTS)	--	5.0E-2
		1,3E-4/hr (TDP FTR)	12	
OEP-PSF-FC-DG1A,B	--	2.1E-2 (DG FTS)	--	
		1.26E-3/hr (DG FTR)	24	
		1.6E-2 (DG T&M)	--	6.7E-2
AFW-XHE-FO-TRBMC	--	--	--	0.05

Table 3-15 (Cont'd)

Event	Beta	Mean Unavailability or Failure Rate	Time(hr)	Probability
AFW-XHE-FO-TRBLO	--	--	--	0.18
AFW-XHE-FO-EFWXX	--	--	--	0.28
PPS-XHE-FO-FDBLD	-	--	--	0.05
PCC-XHE-FO-ISOL	--	--	--	0.1
OEP-XHE-FO-FUEL	--	--	--	0.01

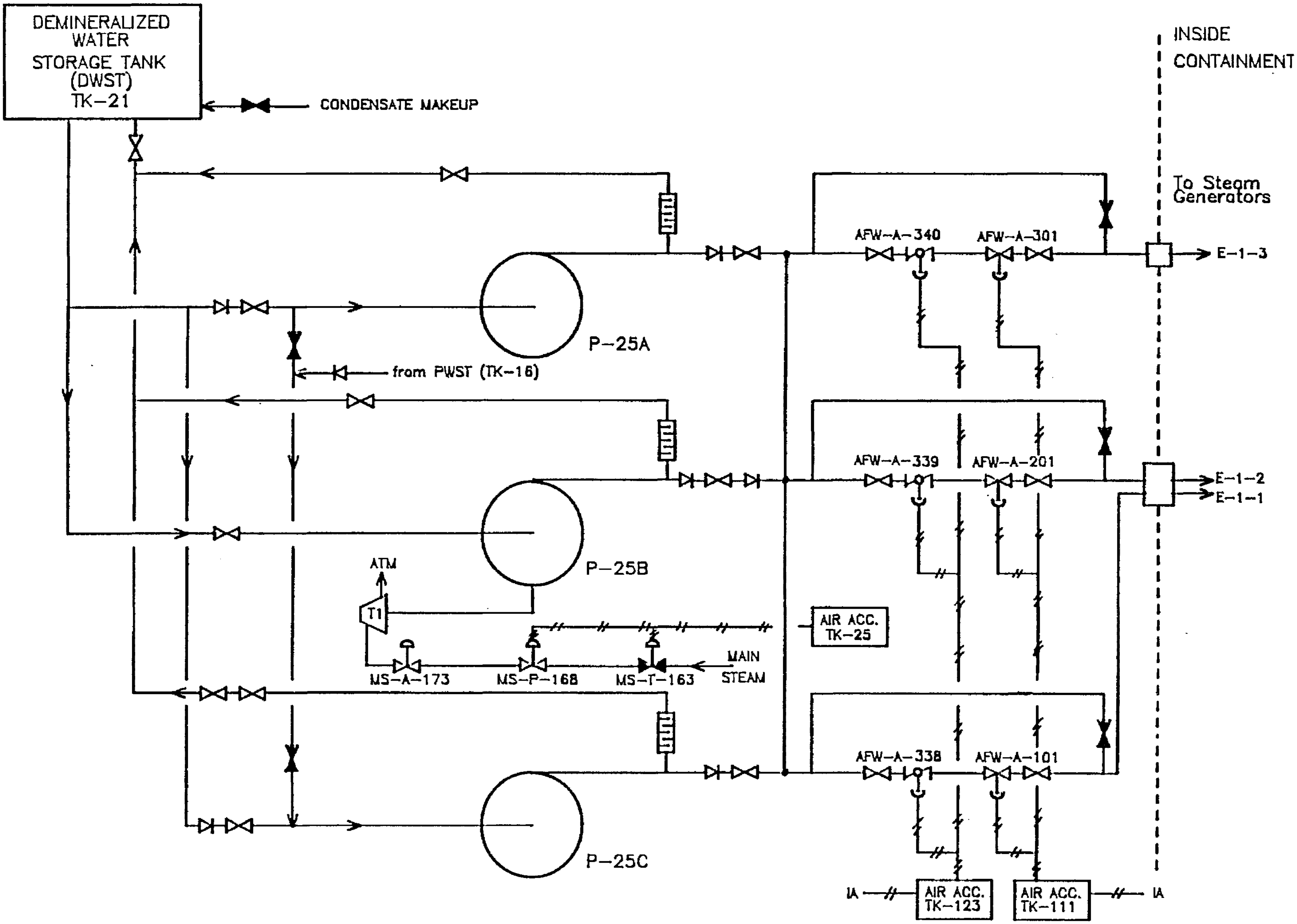


Figure 3–1 Auxiliary Feedwater

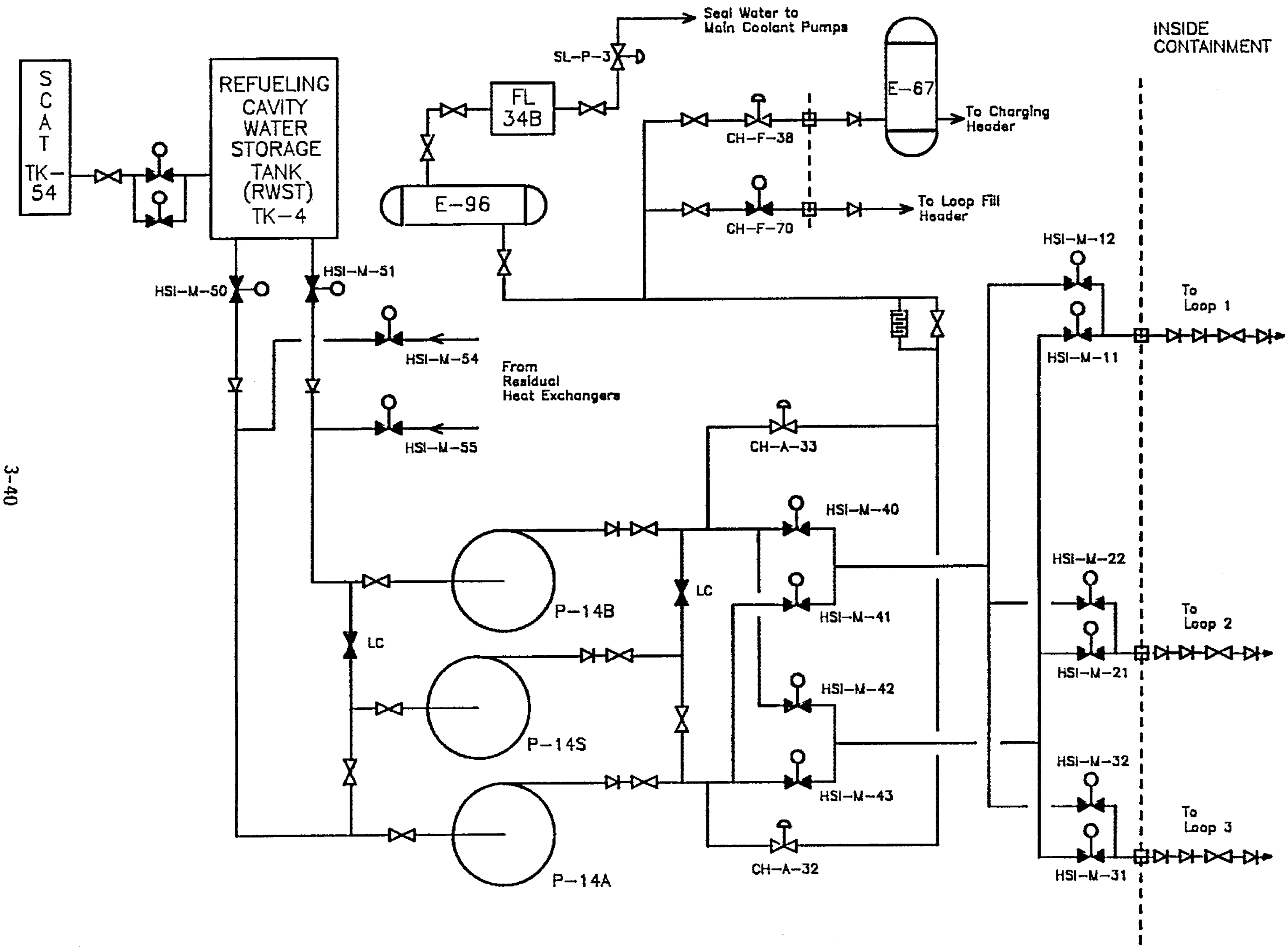


Figure 3–2 High Pressure Safety Injection

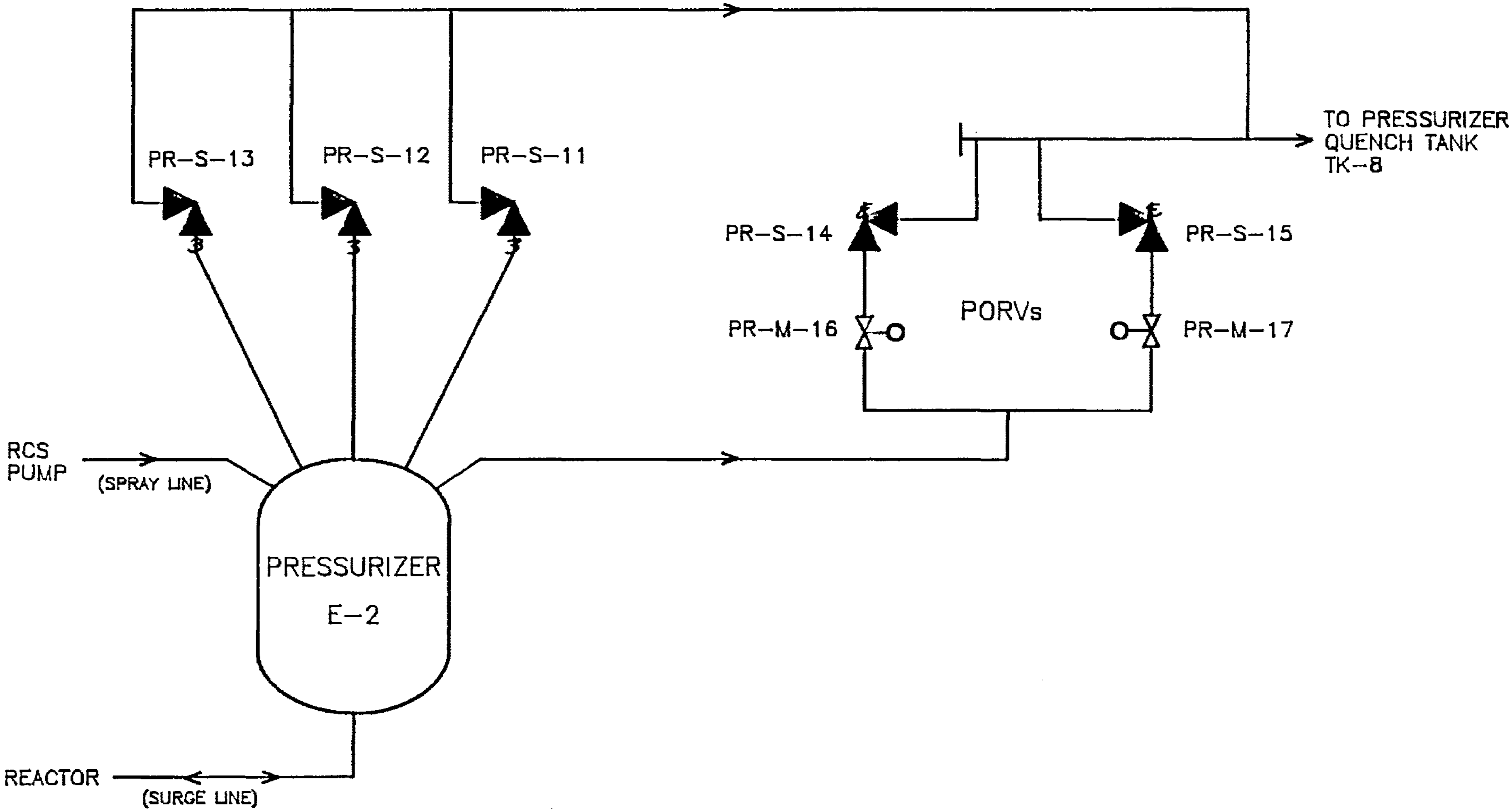


Figure 3-3 Primary Pressure Relief

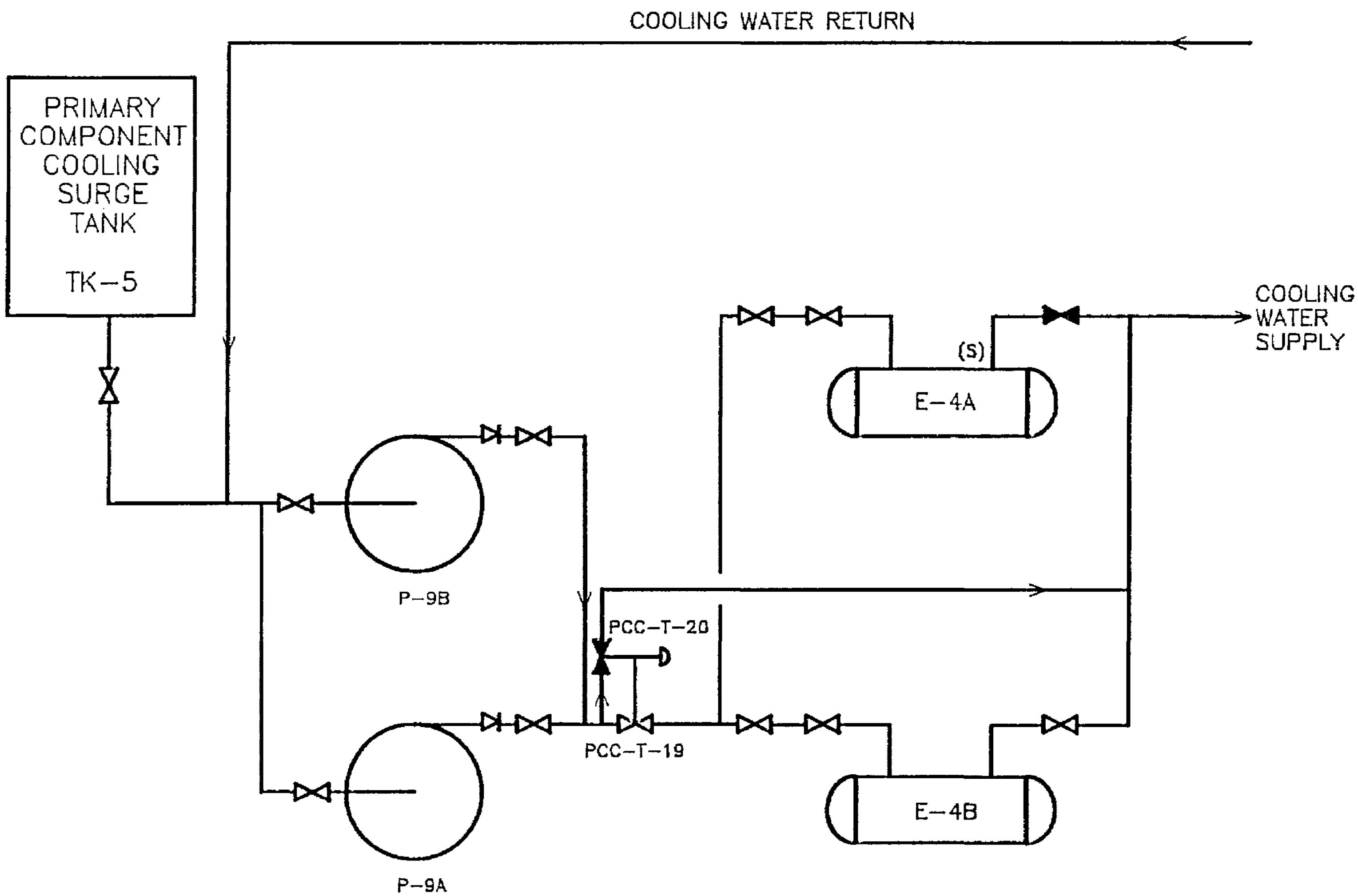


Figure 3-4 Primary Component Cooling Water

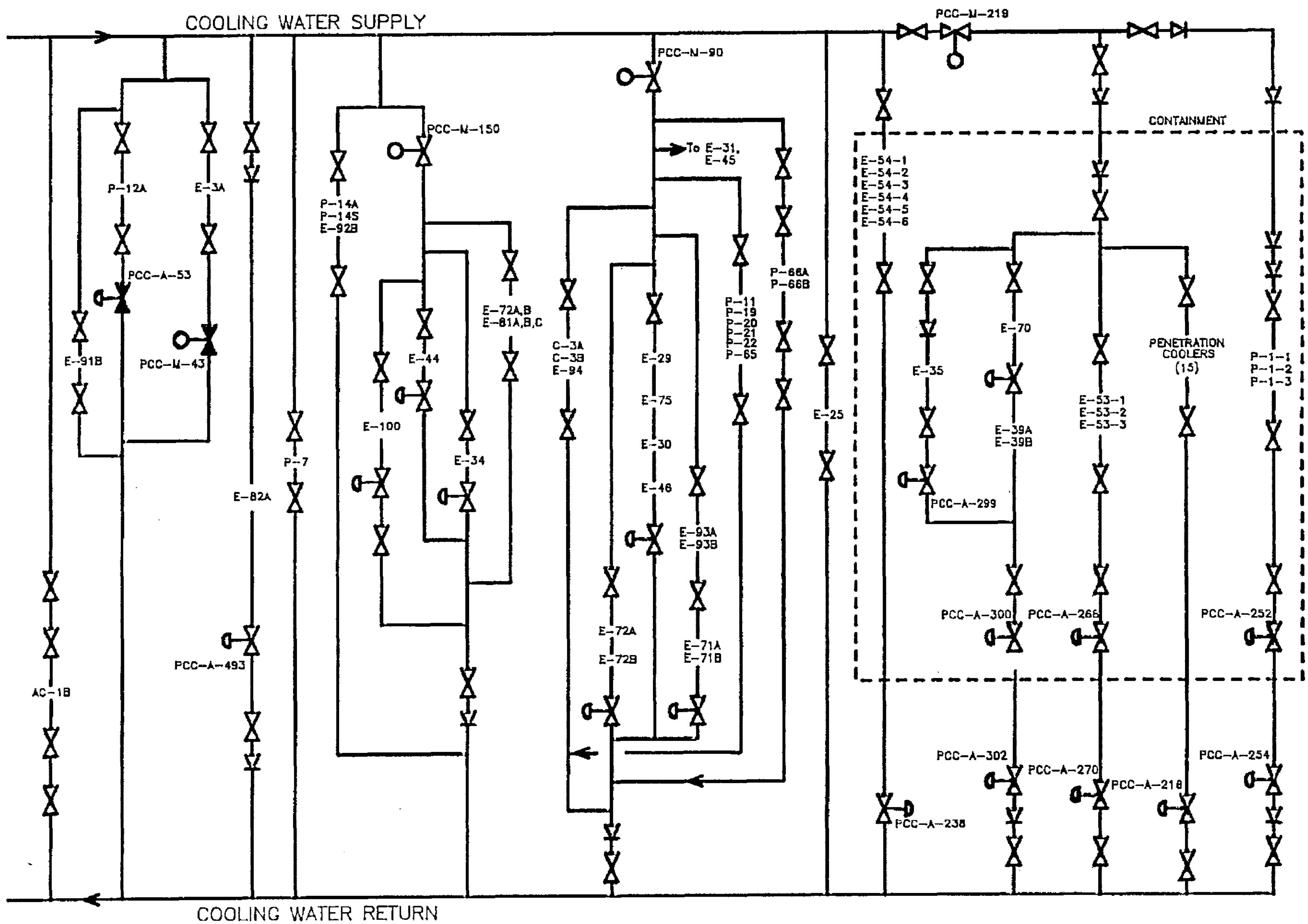


Figure 3-4 (cont.)

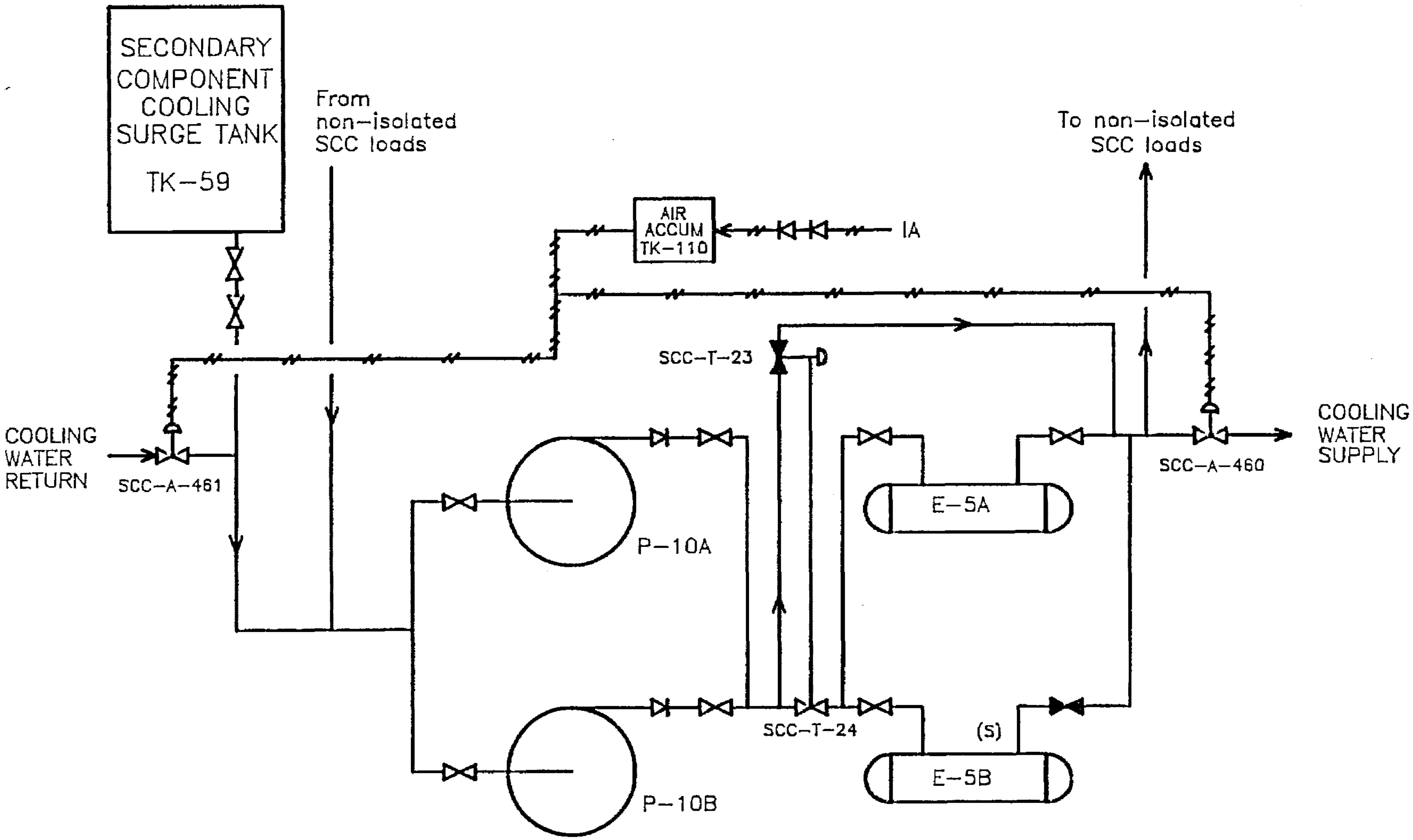


Figure 3-5 Secondary Component Cooling Water

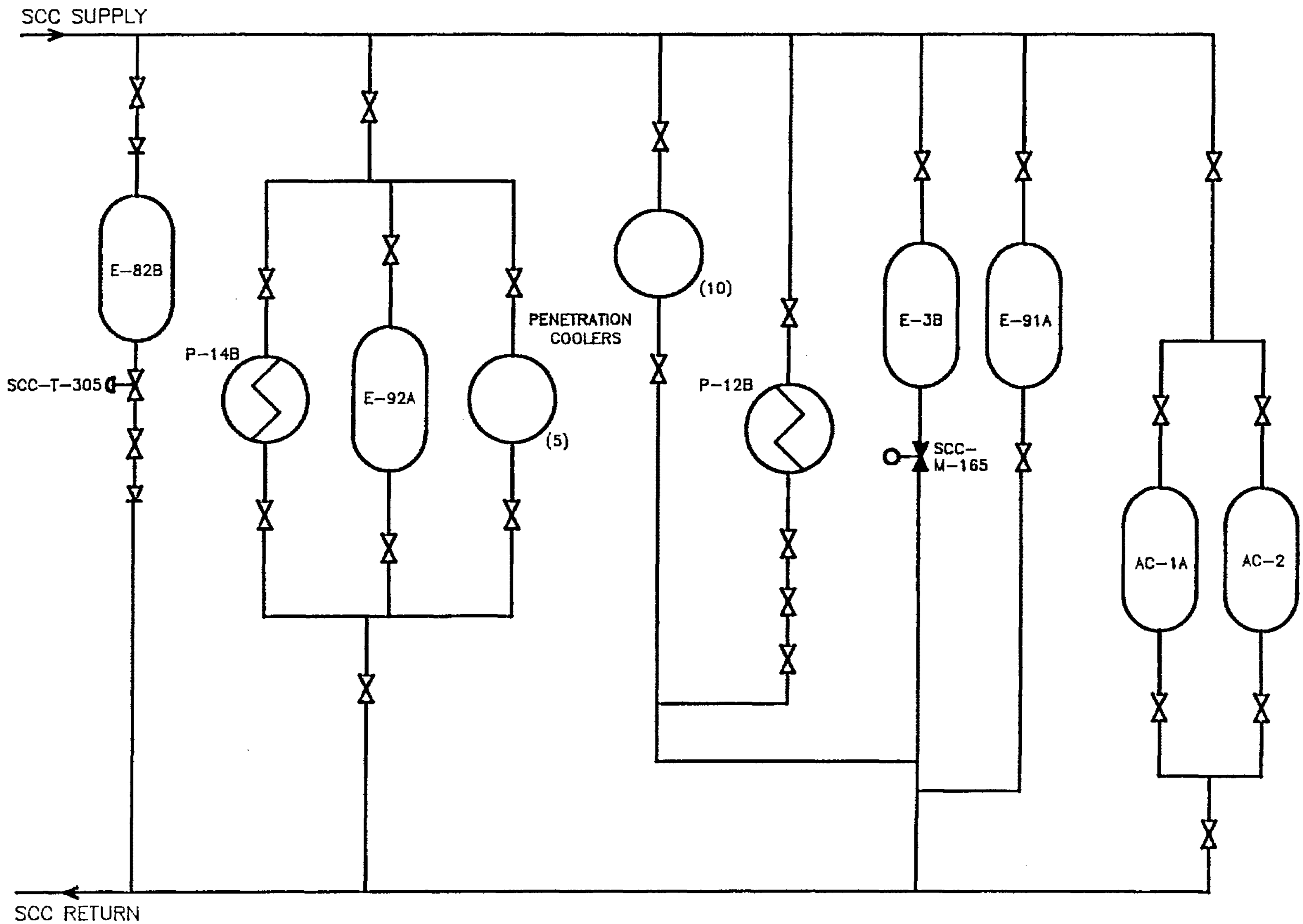


Figure 3-5 (cont.)

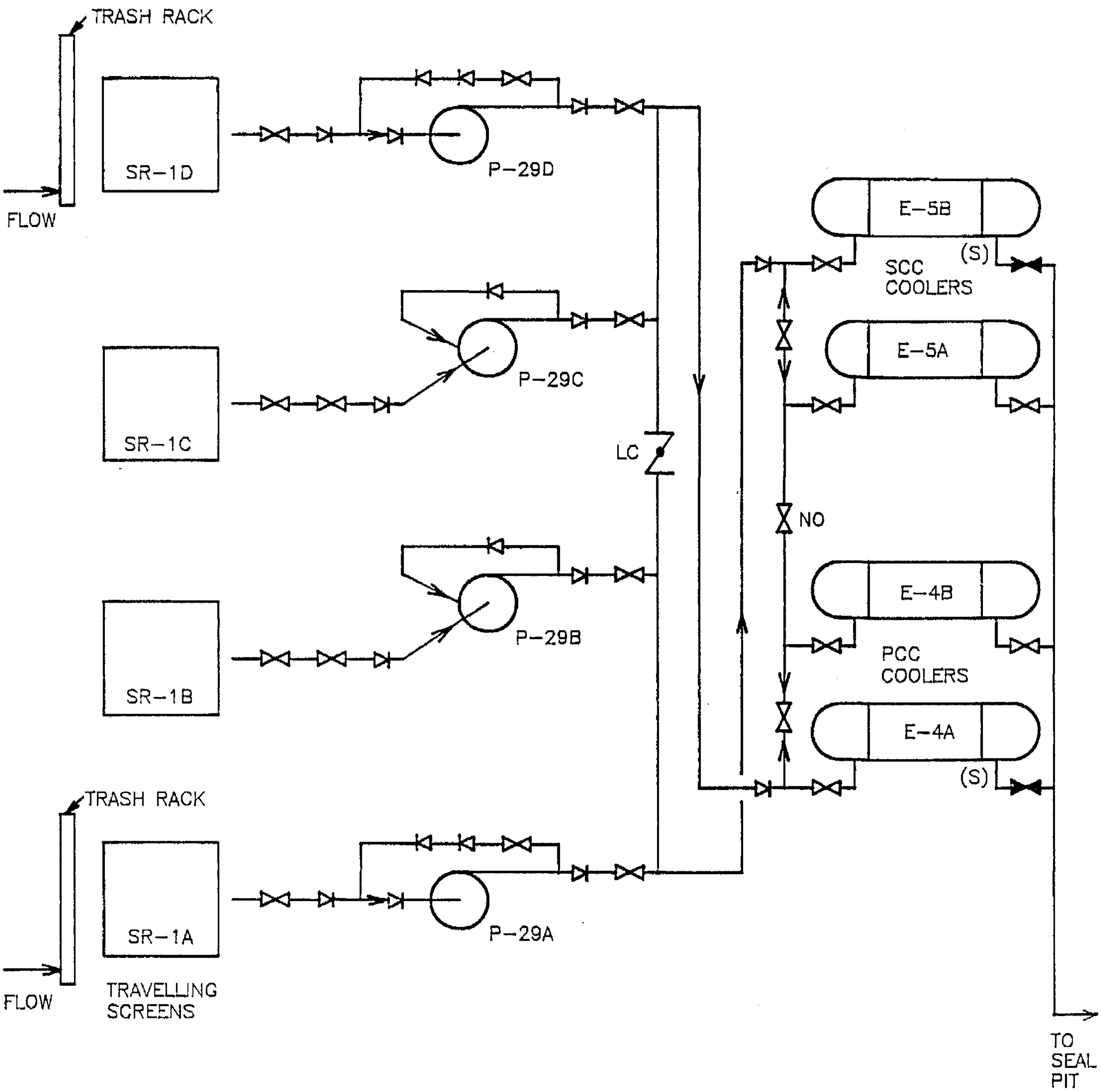


Figure 3-6 Service Water System

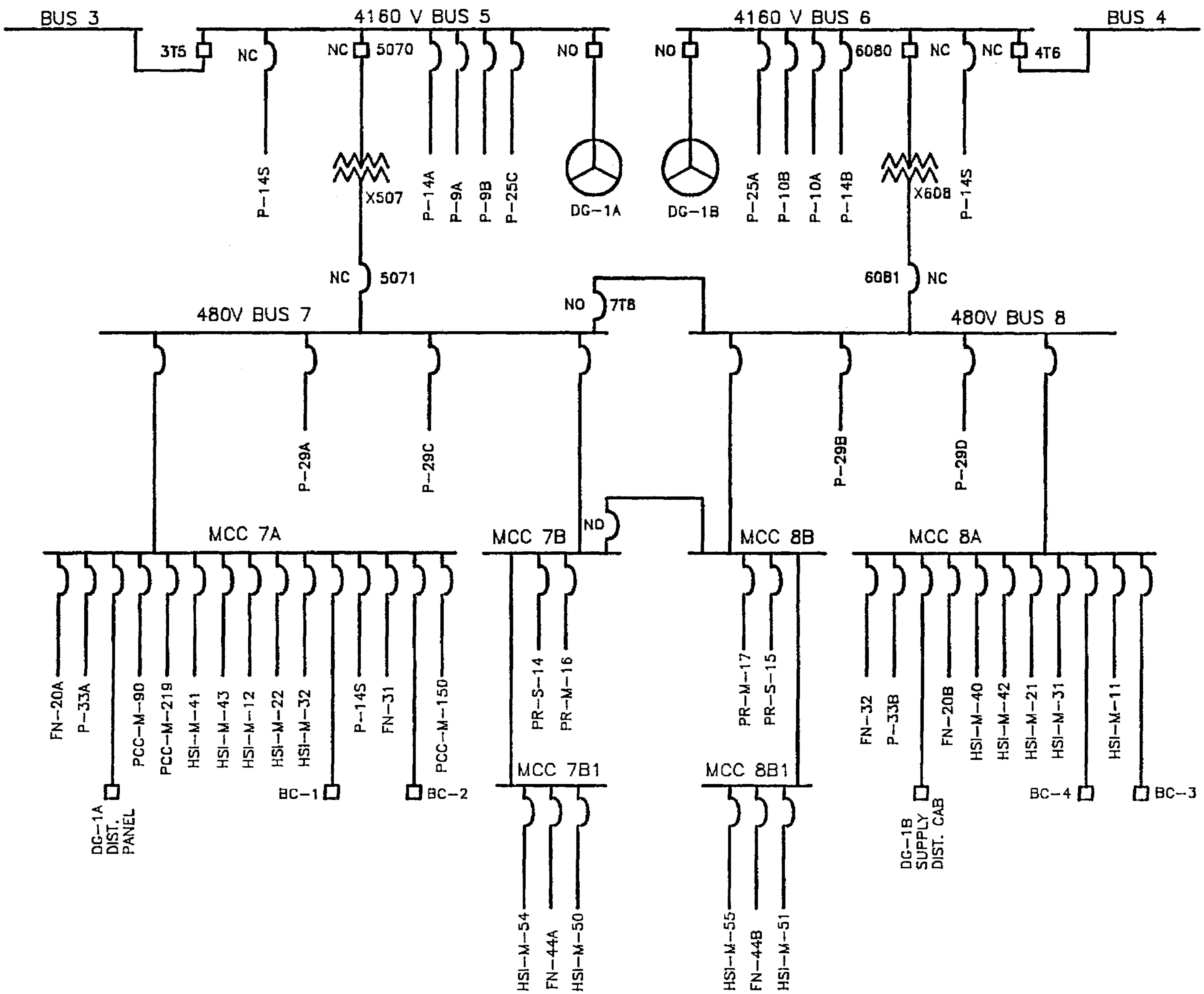


Figure 3-7 AC Power

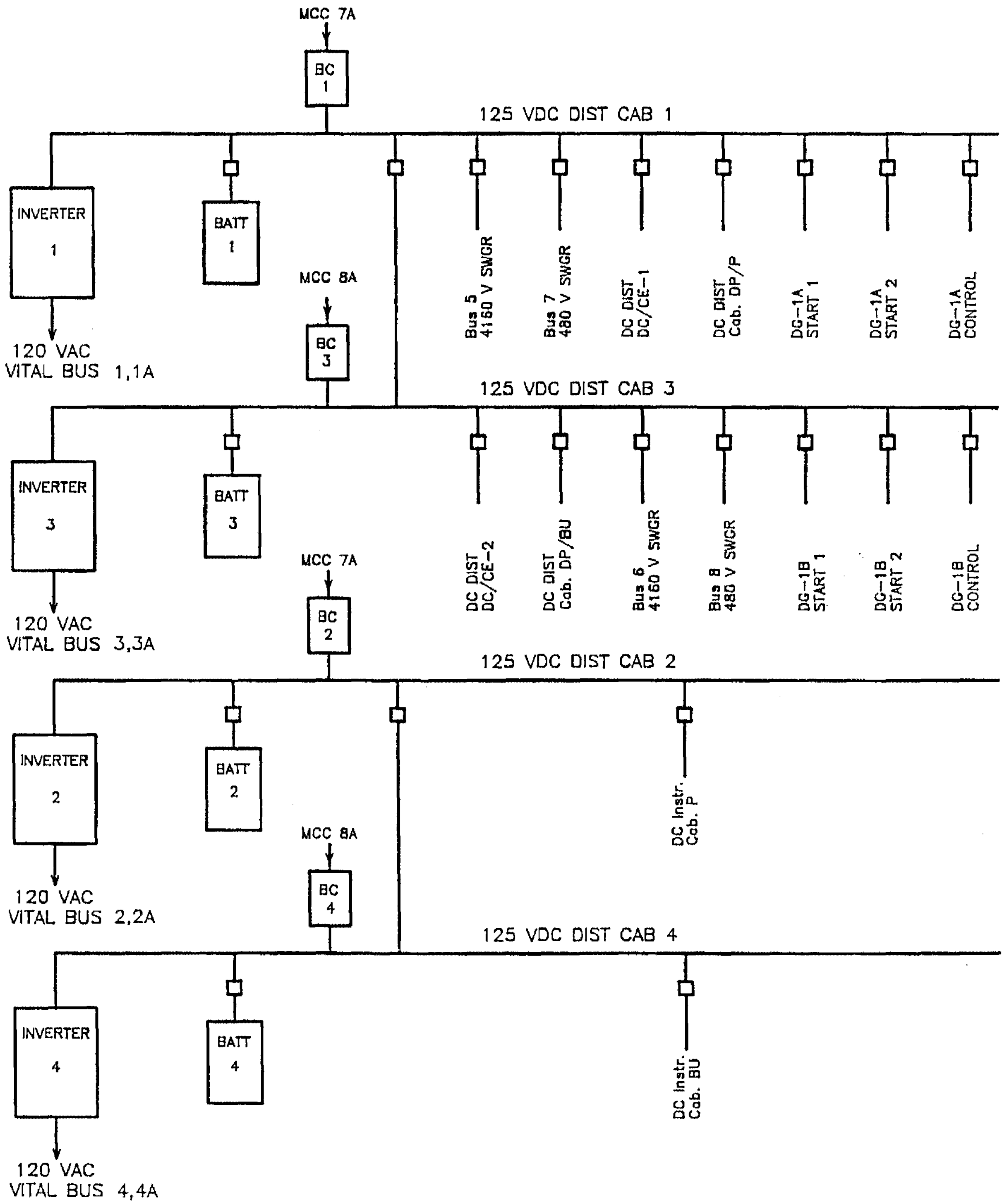


Figure 3–8 DC Power

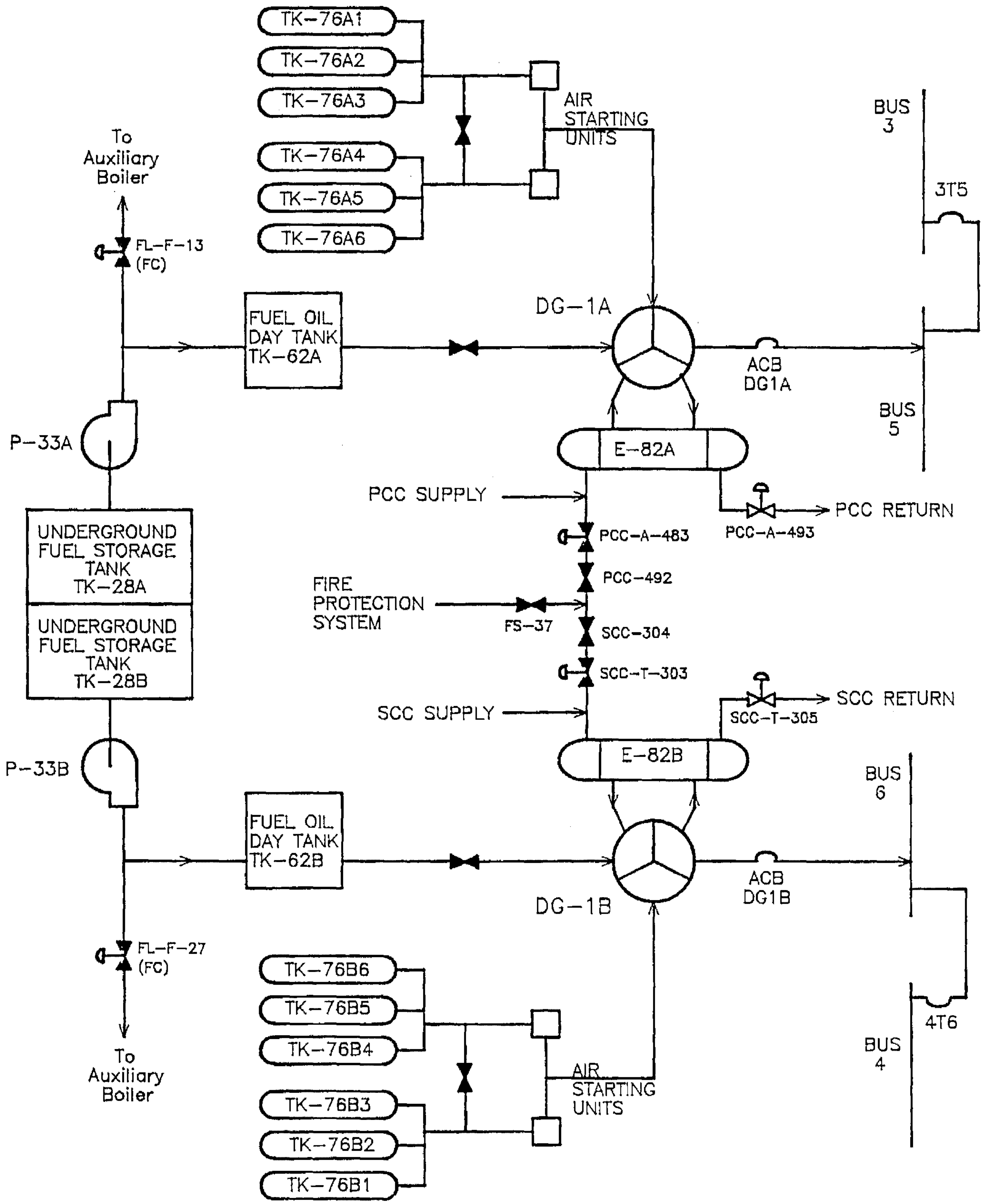


Figure 3-9 Diesel Generators

4. ACCIDENT SEQUENCE ASSESSMENT AND RESULTS

In order to obtain the plant level Boolean equation which represents the cut sets for the accident sequences, two event trees were developed. These event trees, one which includes a small LOCA and one which does not, are described in Section 4.1. The core damage sequences that are derived from the event trees are discussed in Section 4.2. The Boolean equations obtained from the analysis of each event tree are described in Section 4.3. Section 4.4 presents the plant-level Boolean equation.

4.1 Event Trees

Because it was not feasible to enter the reactor containment, the review team could not inspect and verify the seismic capacity of the LOCA sensitive piping. Therefore, the plant analysis was segregated into two cases: a seismic event and loss of offsite power with no LOCA, and a seismic event and loss of offsite power with a small LOCA. For this project a small LOCA is defined as one which requires the use of HPSI for core makeup. Based on data from other PRAs this would be a break in the range of roughly three-eighths inch to two-inch diameter. However, no plant-specific thermodynamic analyses were performed under this review to determine the actual size range. The first event tree considers a seismic event without a concurrent LOCA, the second tree considers a seismic event with a small LOCA. The minimal cut sets from the resulting accident sequences are then combined to determine the plant Boolean equation.

4.1.1 No LOCA Case

The event tree for the first case is presented in Figure 4-1. This event tree depicts the success or failure of the various front-line mitigating systems in response to a seismic event with accompanying loss of offsite power. The systems are displayed across the top of the figure. At each node in the event tree, success of the system is represented by the upward branch, and failure by the downward branch. Each path through the event tree represents a sequence. Those sequences that end with a status of "OK" have successfully mitigated the seismic event, and do not result in core damage. Those sequences that end with a status of "CD" have failed to provide all necessary safety functions, and result in core damage. They are identified by the abbreviation for the initiating event (in this case, event T_1) and the abbreviations for any front-line system failures (e.g., event L or P_1). These core damage accident sequences are then evaluated using the system fault trees to determine the minimal cut sets and Boolean equations. The initiating event for the first event tree is a seismic event which causes a loss of offsite power, designated as T_1. Following this event the reactor must be made subcritical (event C). Potential accident sequences due to failure of the reactor subcriticality system were not evaluated further because the fragility analysis of the reactor internals and control rods found the seismic capacity to be greater than the review level earthquake. Nonseismic failures of the subcriticality system are also insignificant (approximately 3.0E-5 per demand).

Upon successful reactor subcriticality, the AFW system is used for core decay heat removal (event L). Successful operation of this system (at least one of three pumps feeding one of three steam generators) will provide core cooling (early) and prevent core damage. Core inventory makeup is not required because the reactor coolant system (RCS) is intact for this event tree. Failures of the RCS integrity (such as a safety relief valve LOCA or RCP seal LOCA) are analyzed in the next section.

Should failure of the AFW system occur, the operator should begin feed and bleed (event P_1). In this case with no concurrent LOCA, both PORVs (and the associated block valves) must be opened within approximately 30 minutes for system success. Failure to open both PORV lines (i.e., one or both remain closed) will result in inadequate core cooling and eventual core damage (sequence T_1LP_1).

Following the successful initiation of feed and bleed, the HPSI system (event D) must be used to provide makeup to the RCS. This requires at least one of two HPSI pumps (the third pump is evaluated for recovery actions only) to provide injection to one of three RCS loops. Also included in HPSI success criteria is the long-term availability of the recirculation system. As discussed in Chapter 3, this included the availability of the containment spray pump area fans. Failure of the HPSI system is part of core damage sequence T_1LD.

4.1.2 Small LOCA Case

The event tree for the second analysis is shown in Figure 4-2. For this case the initiating event is an earthquake that causes loss of offsite power, concurrent with a small LOCA (event S_2). Larger LOCAs (beyond the makeup capacity of the HPSI system) were not evaluated based on the guidance in NUREG/CR-4334 and -4482 (Budnitz et al., 1985, and Prassinos et al., 1986). Ths small LOCA could be caused by nonseismic or seismic failures. The nonseismic failures considered were a stuck-open PORV or safety valve, a significant reactor coolant pump (RCP) seal LOCA (greater than about 20 gallons per minute per RCP), and a coincident small LOCA. Based on generic ASEP data, the probability of a stuck open PORV that is not isolated by the block valve, or the probability of a stuck-open safety valve is equal to or less than the screening value of 1.0E-3. The probability of a significant RCP seal LOCA with the RCP seal design at Maine Yankee is judged to be similarly low, based on discussions with Yankee Atomic. Also, the probability of a nonseismic small LOCA coincident with the earthquake event and recovery time is very small (less than 1.0E-3). Therefore, nonseismic small LOCAs were screened out of the analysis.

The small LOCAs induced by seismic failures that were considered were a stuck-open PORV or safety valve, and small breaks in LOCA sensitive piping. Based on the fragility analysis team review, stuck-open PORVs or safety valves were judged to have HCLPFs greater than 0.3g, and were therefore screened out. However, because the review team was not able to inspect and verify the seismic capacity of small piping and impulse lines inside containment, the HCLPF of these lines is not known. Therefore, small LOCAs could not be screened out. The event tree for the small LOCA case is based on a seismic event that causes loss of offsite power and a small LOCA, and requires core inventory makeup for mitigation.

Again the reactor must first be made subcritical (event C) and as described in Section 4.1.1, the subcriticality system has high seismic capacity. Therefore, failure of the subcriticality function were not considered further.

Following successful reactor subcriticality, the AFW system is used for core decay heat removal (event L), with the same success criteria as described above. With the small LOCA, the HPSI system must be used to provide makeup to the RCS (event D). Failure of HPSI at this point (no pumps feeding any RC loop) will lead to core damage sequence S_2D.

Failure of the AFW system to provide decay heat removal will require the use of feed and bleed. Due to the small LOCA the operator need only open one PORV (with its associated block valve) within 30 minutes to successfully initiate feed and bleed (event P_2). Failure to open at least one PORV (i.e., both remain closed) within this time will result in sequence S_2LP_2. With successful PORV operation, HPSI is again required to provide RCS makeup (event D), with the same success criteria described previously. Failure of AFW and HPSI result in core damage sequence S_2LD.

4.2 Core Damage Sequences

4.2.1 T_1LP_1

To determine the cut sets for sequence T_1LP_1 (seismic event with loss of offsite power followed by AFW failure, followed by PORV failure for feed and bleed) the pruned and merged fault trees for the AFW system and the PORVs (no LOCA case) developed in Section 3.2 were combined. By combining these two trees with an "AND" logic gate, failure of both systems is required to result in the sequence (T_1LP_1). A description of the basic events that make up the accident sequences is found in Table 4-1.

The minimal cut sets for this sequence are listed in Table 4-2. As with the system cut sets, the seismic failures were arbitrarily assigned a probability of 1.0 for ranking purposes. There are no single faults which lead to this accident sequence. Of the double faults, some contain at least one seismic event, and others contain only nonseismic events (e.g., random, common cause, human error). The event ACP-TFM-EQ-57X68 (seismic failure of the station transformers) results in loss of power to both PORVs and both motor-driven AFW pumps. Seismic failure of the DWST (AFW-TNK-EQ-DWST) leads to failure of the turbine-driven AFW pump, as does random failure of the TDP (AFW-TDP-LF-P25B) and failure to place the pump in service from the MCR (AFW-XHE-FO-TRBMC). Seismic failure of the SCC and PCC coolers, the air conditioner chillers or the circulating water pump house (CCW-HTX-EQ-4B5A, CCW-ACX-EQ-CHILL, SWS-BKW-EQ-CIRC) leads to failure of both the PCC and SCC, which in turn fails both diesels. Seismic failure of the diesel day tanks (OEP-TNK-EQ-TK62X) also results in loss of both DGs. The loss of both diesels has the same effect as seismic failure of the transformers. Generally, two separate faults are required to fail both the AFW motor-driven and turbine-driven pumps. An exception is the common cause failure of AFW.

4.2.2 T_1LD

The pruned and merged fault trees for the AFW system and the HPSI system were merged with an "AND" gate to determine the cut sets for this sequence (seismic event with loss of offsite power followed by loss of AFW, success of PORVs, and loss of HPSI).

The minimal cut sets for sequence T_1LD are found in Table 4-3. Again there are no single faults. Most of the double faults contain at least one seismic event, the rest consist only of random, common cause or human failures. In this sequence, seismic failure of the transformers, SCC/PCC coolers, air conditioner chillers, circulating water pump house or DG day tanks leads to loss of power to both HPSI pumps, all HPSI motor-operated valves, and both motor-driven AFW pumps. The AFW faults are the same as those described for sequence T_1LP_1. Seismic failure of the RWST (HPI-TNK-EQ-RWST) or common cause failure (HPI-CCF-FC-HPSI) results in HPSI failure.

It is interesting to note that PCC-XHE-FO-ISOL, which leads to PCC failure, does not appear here with the double faults, as it does with sequence T_1LP_1. This is because both PCC and SCC must fail to fail HPSI, while failure of only one was necessary to fail one of the two required PORVs.

4.2.3 S_2D

The fault tree for this sequence consists only of the HPSI pruned and merged system fault tree. Given the initiating event (seismic event with loss of offsite power and a small LOCA) followed by initiation of AFW, failure of HPSI will result in core damage. Thus the minimal cut sets listed in Table 4-4 for sequence S_2D are the same as those described for the HPSI system in Section 4.2.2.

4.2.4 S_2LP_2

Sequence S_2LP_2 is similar to sequence T_1LP_1, but the PORV success criteria is different. For S_2LP_2 only one PORV must be opened (i.e., both PORVs remain closed for failure). The system fault trees combined to determine the sequence cut sets were those for AFW and PORVs (small LOCA), developed in Section 3.2. The minimal cut sets for this sequence are listed in Table 4-5. There are no single faults that lead to core damage.

The major difference between the second order cut sets for sequence S_2LP_2 and for sequence T_1LP_1 is that the former do not contain any events which fail PCC only. This is as expected, since both PORVs must fail (and thus both DGs) in addition to AFW failure to give sequence S_2LP_2. As PCC fails DG-1A, another event is needed to fail DG-1B.

4.2.5 S_2LD

The final sequence consists of the same events (and thus the same system fault trees) as sequence T_1LD, only the initiating events differ. Therefore, the cut sets for sequence S_2LD listed in Table 4-6 are identical to those in Table 4-3.

4.3 Boolean Equations for No LOCA and LOCA Cases

The Boolean equations for the core damage sequences with no LOCA and core damage sequences with a small LOCA are given in Table 4-7. The numbers correspond to the basic events listed in Table 4-1. Only the first and second order faults which contain at least one seismic failure event are included in the Boolean equations. Although it is possible for core damage to occur through nonseismic failures only, these were not evaluated further for this seismic margins project. Also, the final HCLPF calculations determined that some of the seismic failure events had final HCLPFs greater than 0.3g, so they were pruned out.

The events included in the Boolean equations affect the systems as follows:

Event 4: Seismic failure of the station transformers (ACP-TFM-EQ-57X68) also results in loss of power to all HPSI components, both PORVs, and both AFW motor-driven pumps.

Event 7: Seismic failure of the RWST (HPI-TNK-EQ-RWST) results in loss of HPSI.

Event 8: Seismic failure of the DWST (AFW-TNK-EQ-DWST) results in loss of the turbine-driven AFW pump (the motor driven pumps have the PWST as a backup).

Event 14: Nonseismic common cause failure of the DGs (OEP-CCF-FC-DGN) results in loss of power to all HPSI components, both PORVs and both AFW motor-driven pumps.

Event 15: Nonseismic common cause failure of the AFW system (AFW-CCF-FC-AFW) results in total loss of AFW.

Event 16: Operator failure to refill the DG day and/or integral tanks (OEP-XHE-FO-FUEL) leads to failure of both DGs, resulting in loss of power to all HPSI components, both PORVs and both AFW motor-driven pumps.

Event 17: Operator failure to place the AFW turbine-driven pump in service from the MCR (AFW-XHE-FO-TRBMC) results in loss of the TDP.

Event 20: Collapse of the circulating water pump house (SWS-BKW-EQ-CIRC) will fail all SW pumps, which in turn fails PCC and SCC due to lack of SW flow through the coolers. The effects of PCC and SCC failure are loss of both DGs due to lack of cooling, which results in a loss of power for all HPSI components, both PORVs, and both AFW motor-driven pumps.

Event 22: Random failure of the AFW turbine-driven pump (AFW-TDP-LF-P25B) results in loss of the TDP.

Those numbered events in Table 4-1 that are not in this final list were either pruned because their final HCLPF was greater than 0.3g, subsumed into the above events because they were not in minimal cut sets, or were only in nonseismic cut sets.

4.3.1 No LOCA Case

To determine the reduced Boolean equation for this event tree, sequences T_1LP_1 and T_1LD were combined with an "OR" gate, because occurrence of either of these sequences will lead to core damage.

The combinations of events described above that make up the reduced Boolean cut sets lead to core damage as follows:

- Those cut sets that combine events 4 or 20 with events 8, 15, 17, or 22 result in both sequence T_1LP_1 and T_1LD. A fault from the first group (4 or 20) results in loss of all HPSI components, both PORVs and both AFW motor-driven pumps. A fault from the second group (8, 15, 17, 22) results in loss of the turbine-driven AFW pump.

- Events 14 or 16 also lead to loss of HPSI, both PORVs, and both AFW motor-driven pumps. However, only the cut sets where one of these events is combined with event 8 are included in the Boolean. These lead to both T_1LD and T_1LP_1. Coupling of 14 or 16 with 15, 17, or 22 result in cut sets of two nonseismic events, which were screened out.

- Finally event 7 is combined with 15 to give sequence T_1LD. Event 7 leads to HPSI failure, and 15 to AFW failure.

4.3.2 Small LOCA Case

To determine the Boolean equation for the second event tree, sequences S_2D, S_2LP_2 and S_2LD were combined with an "OR" gate because occurrence of any one of these sequences will cause core melt. The reduced Boolean for this event tree consists only of single faults. This is because each of these faults will cause failure of HPSI, because this corresponds to sequence S_2D. No other system failures are required for core damage to occur. Essentially, the cut sets that would lead to the failures of AFW and one PORV, or AFW and HPSI, all fall out as nonminimal cut sets or consist only of nonseismic events and are screened out. The events which make up the small LOCA Boolean are 4, 7, and 20.

4.4 Plant-Level Boolean Equation

Because the small LOCA sensitive piping could not be inspected and its seismic capacity verified, the Boolean equations for small LOCA and no LOCA cases were combined in two different ways. Sensitivity analysis could then be used to estimate the plant HCLPF. The sensitivity studies were performed by the fragility analysis team.

The first method used split fractions to express the conditional probability of a seismic induced small LOCA given the seismic event. The two Boolean equations in Table 4-7 were combined using these split fractions, and sensitivity stúdies can be performed by varying the split fractions.

The second method used an additional term for the small LOCA Boolean equation that represented the HCLPF of the seismic induced small LOCA. Sensitivity studies can then be performed by varying the small LOCA HCLPF valve.

Table 4-1. Basic event descriptions.

Number	Event	Description	Nonseismic Probability
1	CCW-HTX-EQ-4B5A	Seismic failure of PCC/SCC coolers	SF
2	AFW-TNK-EQ-TK25	Seismic failure of air tank for AFW AOVs	SF
3	AFW-XHE-FO-EFWXX	Failure to align PWST to AFW MDPs	2.8E-01
4	ACP-TFM-EQ-57X68	Seismic failure of station transformers	SF
5	HPI-CCF-FC-HPSI	Nonseismic common cause HPSI failure	2.1E-03
6	PPS-XHE-FO-FDBLD	Failure to open PORVs for feed and bleed	5.0E-02
7	HPI-TNK-EQ-RWST	Seismic failure of the RWST	SF
8	AFW-TNK-EQ-DWST	Seismic failure of the DWST	SF
9	PPS-CCF-FC-PORVS	Nonseismic common cause PORV failure	8.0E-04
10	PCC-XHE-FO-ISOL	Failure to close PCC isolation valves	1.0E-01
11	OEP-PSF-FC-DG1B	Random failure of diesel generator 1B	6.7E-02
12	OEP-PSF-FC-DG1A	Random failure of diesel generator 1A	6.7E-02
13	AFW-XHE-FO-TRBLO	Failure to start AFW TDP locally	1.8E-01
14	OEP-CCF-FC-DGN	Nonseismic common cause DG failure	2.6E-03
15	AFW-CCF-FC-AFW	Nonseismic common cause AFW failure	2.0E-04
16	OEP-XHE-FO-FUEL	Failure to refill DG fuel tanks	1.0E-2
17	AFW-XHE-FO-TRBMC	Failure to start AFW TDP from MCR	5.0E-02
18	OEP-TNK-EQ-TK62X	Seismic failure of DG day tanks	SF
19	CCW-ACX-EQ-CHILL	Seismic failure of air cond. chillers	SF
20	SWS-BKW-EQ-CIRC	Seismic failure of circ. water pump house	SF
21	AFW-TNK-EQ-PWST	Seismic failure of the PWST	SF
22	AFW-TDP-LF-P25B	Random failure of the AFW TDP	5.0E-02

SF indicates that the event is a seismic related failure, and a HCLPF will be determined by the fragility analysis team.

Table 4-2. Sequence T_1LP_1 cut sets.

No Single Faults

Ranked Double Faults		Nonseismic Probability
ACP-TFM-EQ-57X68	AFW-TNK-EQ-DWST	SF
CCW-HTX-EQ-4B5A	AFW-TNK-EQ-DWST	SF
CCW-ACX-EQ-CHILL	AFW-TNK-EQ-DWST	SF
SWS-BKW-EQ-CIRC	AFW-TNK-EQ-DWST	SF
OEP-TNK-EQ-TK62X	AFW-TNK-EQ-DWST	SF
ACP-TFM-EQ-57X68	AFW-TDP-LF-P25B	5.0E-02
AFW-XHE-FO-TRBMC	ACP-TFM-EQ-57X68	5.0E-02
CCW-HTX-EQ-4B5A	AFW-TDP-LF-P25B	5.0E-02
CCW-ACX-EQ-CHILL	AFW-TDP-LF-P25B	5.0E-02
SWS-BKW-EQ-CIRC	AFW-TDP-LF-P25B	5.0E-02
OEP-TNK-EQ-TK62X	AFW-TDP-LF-P25B	5.0E-02
CCW-HTX-EQ-4B5A	AFW-XHE-FO-TRBMC	5.0E-02
CCW-ACX-EQ-CHILL	AFW-XHE-FO-TRBMC	5.0E-02
SWS-BKW-EQ-CIRC	AFW-XHE-FO-TRBMC	5.0E-02
OEP-TNK-EQ-TK62X	AFW-XHE-FO-TRBMC	5.0E-02
OEP-XHE-FO-FUEL	AFW-TNK-EQ-DWST	1.0E-02
OEP-CCF-FC-DGN	AFW-TNK-EQ-DWST	2.6E-03
OEP-XHE-FO-FUEL	AFW-TDP-LF-P25B	5.0E-04
AFW-XHE-FO-TRBMC	OEP-XHE-FO-FUEL	5.0E-04
CCW-HTX-EQ-4B5A	AFW-CCF-FC-AFW	2.0E-04
CCW-ACX-EQ-CHILL	AFW-CCF-FC-AFW	2.0E-04
SWS-BKW-EQ-CIRC	AFW-CCF-FC-AFW	2.0E-04
OEP-TNK-EQ-TK62X	AFW-CCF-FC-AFW	2.0E-04
ACP-TFM-EQ-57X68	AFW-CCF-FC-AFW	2.0E-04
OEP-CCF-FC-DGN	AFW-TDP-LF-P25B	1.3E-04
AFW-XHE-FO-TRBMC	OEP-CCF-FC-DGN	1.3E-04
PCC-XHE-FO-ISOL	AFW-CCF-FC-AFW	2.0E-05
OEP-PSF-FC-DG1B	AFW-CCF-FC-AFW	1.3E-05
OEP-PSF-FC-DG1A	AFW-CCF-FC-AFW	1.3E-05
PPS-XHE-FO-FDBLD	AFW-CCF-FC-AFW	1.0E-05
OEP-XHE-FO-FUEL	AFW-CCF-FC-AFW	2.0E-06
OEP-CCF-FC-DGN	AFW-CCF-FC-AFW	5.2E-07
PPS-CCF-FC-PORVS	AFW-CCF-FC-AFW	1.6E-07

SF indicates that the cut set consists entirely of seismic related failures for which HCLPFs will be determined.

Table 4-3. Sequence T_1LD cut sets.

No Single Faults		
Ranked Double Faults		**Nonseismic Probability**
ACP-TFM-EQ-57X68	AFW-TNK-EQ-DWST	SF
CCW-HTX-EQ-4B5A	AFW-TNK-EQ-DWST	SF
CCW-ACX-EQ-CHILL	AFW-TNK-EQ-DWST	SF
SWS-BKW-EQ-CIRC	AFW-TNK-EQ-DWST	SF
OEP-TNK-EQ-TK62X	AFW-TNK-EQ-DWST	SF
ACP-TFM-EQ-57X68	AFW-TDP-LF-P25B	5.0E-02
AFW-XHE-FO-TRBMC	ACP-TFM-EQ-57X68	5.0E-02
CCW-HTX-EQ-4B5A	AFW-TDP-LF-P25B	5.0E-02
CCW-ACX-EQ-CHILL	AFW-TDP-LF-P25B	5.0E-02
SWS-BKW-EQ-CIRC	AFW-TDP-LF-P25B	5.0E-02
OEP-TNK-EQ-TK62X	AFW-TDP-LF-P25B	5.0E-02
CCW-HTX-EQ-4B5A	AFW-XHE-FO-TRBMC	5.0E-02
CCW-ACX-EQ-CHILL	AFW-XHE-FO-TRBMC	5.0E-02
SWS-BKW-EQ-CIRC	AFW-XHE-FO-TRBMC	5.0E-02
OEP-TNK-EQ-TK62X	AFW-XHE-FO-TRBMC	5.0E-02
OEP-XHE-FO-FUEL	AFW-TNK-EQ-DWST	1.0E-02
OEP-CCF-FC-DGN	AFW-TNK-EQ-DWST	2.6E-03
OEP-XHE-FO-FUEL	AFW-TDP-LF-P25B	5.0E-04
AFW-XHE-FO-TRBMC	OEP-XHE-FO-FUEL	5.0E-04
ACP-TFM-EQ-57X68	AFW-CCF-FC-AFW	2.0E-04
CCW-HTX-EQ-4B5A	AFW-CCF-FC-AFW	2.0E-04
CCW-ACX-EQ-CHILL	AFW-CCF-FC-AFW	2.0E-04
CCW-BKW-EQ-CIRC	AFW-CCF-FC-AFW	2.0E-04
OEP-TNK-EQ-TK62X	AFW-CCF-FC-AFW	2.0E-04
HPI-TNK-EQ-RWST	AFW-CCF-FC-AFW	2.0E-04
OEP-CCF-FC-DGN	AFW-TDP-LF-P25B	1.3E-04
OEP-CCF-FC-DGN	AFW-XHE-FO-TRBMC	1.3E-04
AFW-CCF-FC-AFW	OEP-XHE-FO-FUEL	2.0E-06
AFW-CCF-FC-AFW	OEP-CCF-FC-DGN	5.2E-07
HPI-CCF-FC-HPSI	AFW-CCF-FC-AFW	4.2E-07

SF indicates that the cut set consists entirely of seismic related failures for which HCLPFs will be determined.

Table 4-4. Sequence S_2D cut sets.

Ranked Single Faults	Nonseismic Probability
ACP-TFM-EQ-57X68	SF
HPI-TNK-EQ-RWST	SF
CCW-HTX-EQ-4B5A	SF
CCW-ACX-EQ-CHILL	SF
SWS-BKW-EQ-CIRC	SF
OEP-TNK-EQ-TK62X	SF
OEP-XHE-FO-FUEL	1.0E-02
OEP-CCF-FC-DGN	2.6E-03
HPI-CCF-FC-HPSI	2.1E-03

Ranked Double Faults		Nonseismic Probability
PCC-XHE-FO-ISOL	OEP-PSF-FC-DG1B	6.7E-03
OEP-PSF-FC-DG1B	OEP-PSF-FC-DG1A	4.5E-03

SF indicates that the cut set consists entirely of seismic related failures for which HCLPFs will be determined.

Table 4-5. Sequence S_2LP_2 cut sets.

No Single Faults

Ranked Double Faults		Nonseismic Probability
ACP-TFM-EQ-57X68	AFW-TNK-EQ-DWST	SF
CCW-HTX-EQ-4B5A	AFW-TNK-EQ-DWST	SF
CCW-ACX-EQ-CHILL	AFW-TNK-EQ-DWST	SF
SWS-BKW-EQ-CIRC	AFW-TNK-EQ-DWST	SF
OEP-TNK-EQ-TK62X	AFW-TNK-EQ-DWST	SF
ACP-TFM-EQ-57X68	AFW-TDP-LF-P25B	5.0E-02
AFW-XHE-FO-TRBMC	ACP-TFM-EQ-57X68	5.0E-02
CCW-HTX-EQ-4B5A	AFW-TDP-LF-P25B	5.0E-02
CCW-ACX-EQ-CHILL	AFW-TDP-LF-P25B	5.0E-02
SWS-BKW-EQ-CIRC	AFW-TDP-LF-P25B	5.0E-02
OEP-TNK-EQ-TK62X	AFW-TDP-LF-P25B	5.0E-02
CCW-HTX-EQ-4B5A	AFW-XHE-FO-TRBMC	5.0E-02
CCW-ACX-EQ-CHILL	AFW-XHE-FO-TRBMC	5.0E-02
SWS-BKW-EQ-CIRC	AFW-XHE-FO-TRBMC	5.0E-02
OEP-TNK-EQ-TK62X	AFW-XHE-FO-TRBMC	5.0E-02
OEP-XHE-FO-FUEL	AFW-TNK-EQ-DWST	1.0E-02
OEP-CCF-FC-DGN	AFW-TNK-EQ-DWST	2.6E-03
OEP-XHE-FO-FUEL	AFW-TDP-LF-P25B	5.0E-04
AFW-XHE-FO-TRBMC	OEP-XHE-FO-FUEL	5.0E-04
CCW-HTX-EQ-4B5A	AFW-CCF-FC-AFW	2.0E-04
CCW-ACX-EQ-CHILL	AFW-CCF-FC-AFW	2.0E-04
CCW-BKW-EQ-CIRC	AFW-CCF-FC-AFW	2.0E-04
OEP-TNK-EQ-TK62X	AFW-CCF-FC-AFW	2.0E-04
ACP-TFM-EQ-57X68	AFW-CCF-FC-AFW	2.0E-04
OEP-CCF-FC-DGN	AFW-TDP-LF-P25B	1.3E-04
AFW-XHE-FO-TRBMC	OEP-CCF-FC-DGN	1.3E-04
PPS-XHE-FO-FDBLD	AFW-CCF-FC-AFW	1.0E-05
OEP-XHE-FO-FUEL	AFW-CCF-FC-AFW	2.0E-06
OEP-CCF-FC-DGN	AFW-CCF-FC-AFW	5.2E-07
PPS-CCF-FC-PORVS	AFW-CCF-FC-AFW	1.6E-07

SF indicates that the cut set consists entirely of seismic related failures for which HCLPFs will be determined.

Table 4-6. Sequence S_2LD cut sets.

Ranked Double Faults		Nonseismic Probability
No Single Faults		
ACP-TFM-EQ-57X68	AFW-TNK-EQ-DWST	SF
CCW-HTX-EQ-4B5A	AFW-TNK-EQ-DWST	SF
CCW-ACX-EQ-CHILL	AFW-TNK-EQ-DWST	SF
SWS-BKW-EQ-CIRC	AFW-TNK-EQ-DWST	SF
OEP-TNK-EQ-TK62X	AFW-TNK-EQ-DWST	SF
ACP-TFM-EQ-57X68	AFW-TDP-LF-P25B	5.0E-02
AFW-XHE-FO-TRBMC	ACP-TFM-EQ-57X68	5.0E-02
CCW-HTX-EQ-4B5A	AFW-TDP-LF-P25B	5.0E-02
CCW-ACX-EQ-CHILL	AFW-TDP-LF-P25B	5.0E-02
SWS-BKW-EQ-CIRC	AFW-TDP-LF-P25B	5.0E-02
OEP-TNK-EQ-TK62X	AFW-TDP-LF-P25B	5.0E-02
CCW-HTX-EQ-4B5A	AFW-XHE-FO-TRBMC	5.0E-02
CCW-ACX-EQ-CHILL	AFW-XHE-FO-TRBMC	5.0E-02
SWS-BKW-EQ-CIRC	AFW-XHE-FO-TRBMC	5.0E-02
OEP-TNK-EQ-TK62X	AFW-XHE-FO-TRBMC	5.0E-02
OEP-XHE-FO-FUEL	AFW-TNK-EQ-DWST	1.0E-02
OEP-CCF-FC-DGN	AFW-TNK-EQ-DWST	2.6E-03
OEP-XHE-FO-FUEL	AFW-TDP-LF-P25B	5.0E-04
AFW-XHE-FO-TRBMC	OEP-XHE-FO-FUEL	5.0E-04
ACP-TFM-EQ-57X68	AFW-CCF-FC-AFW	2.0E-04
CCW-HTX-EQ-4B5A	AFW-CCF-FC-AFW	2.0E-04
CCW-ACX-EQ-CHILL	AFW-CCF-FC-AFW	2.0E-04
CCW-BKW-EQ-CIRC	AFW-CCF-FC-AFW	2.0E-04
OEP-TNK-EQ-TK62X	AFW-CCF-FC-AFW	2.0E-04
HPI-TNK-EQ-RWST	AFW-CCF-FC-AFW	2.0E-04
OEP-CCF-FC-DGN	AFW-TDP-LF-P25B	1.3E-04
OEP-CCF-FC-DGN	AFW-XHE-FO-TRBMC	1.3E-04
AFW-CCF-FC-AFW	OEP-XHE-FO-FUEL	2.0E-06
AFW-CCF-FC-AFW	OEP-CCF-FO-DGN	5.2E-07
HPI-CCF-FC-HPSI	AFW-CCF-FC-AFW	4.2E-07

SF indicates that the cut set consists entirely of seismic related failures for which HCLPFs will be determined.

Table 4-7. Boolean equations for no LOCA and small LOCA accident sequences.

No LOCA Case
Core Damage = (4+20) * (8+15 + 17+22) + 8 * (14+16) + 7 * 15

Small LOCA Case
Core Damage = 4 + 7 + 20

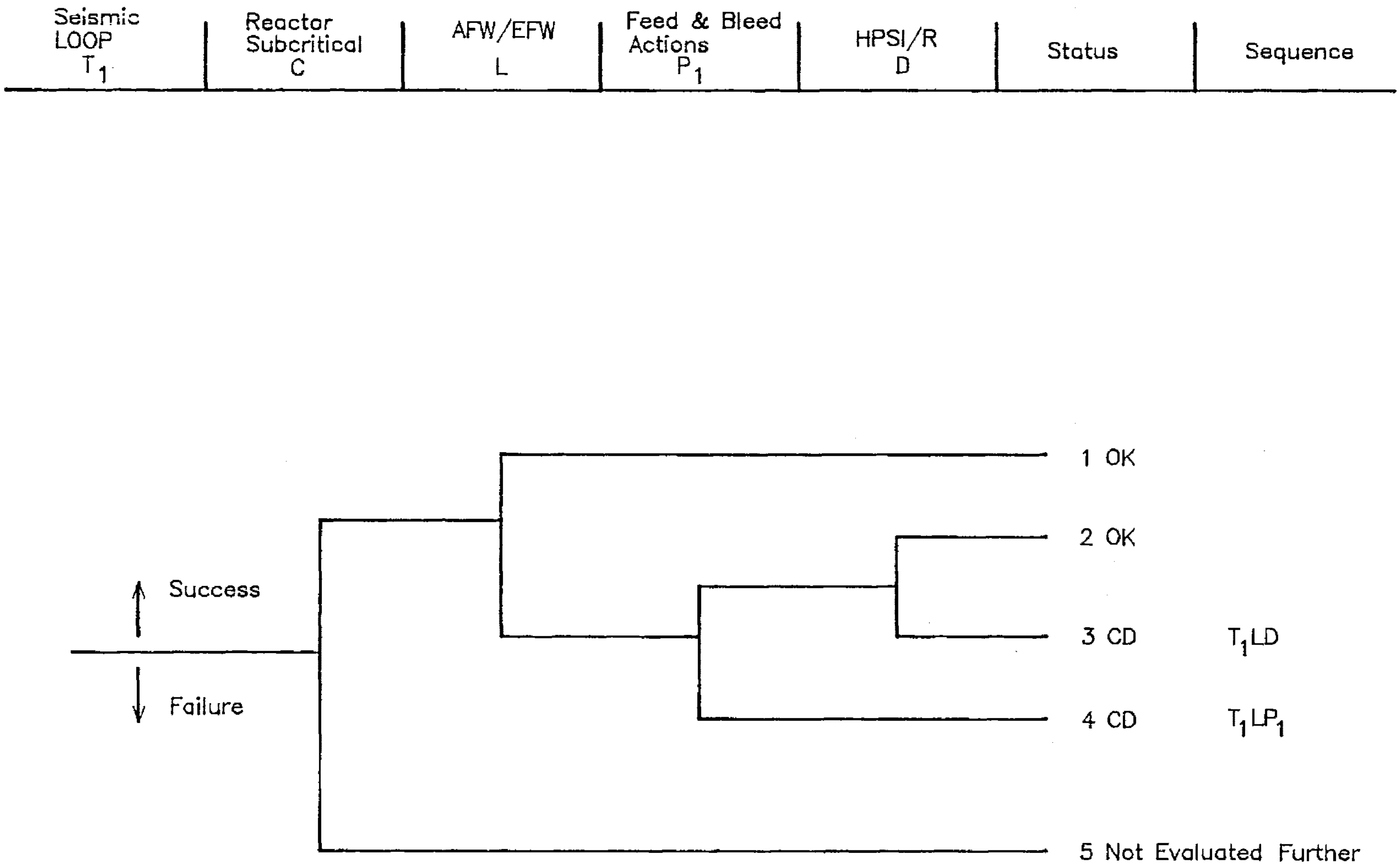


Figure 4-1 Seismic Event, LOOP Event Tree

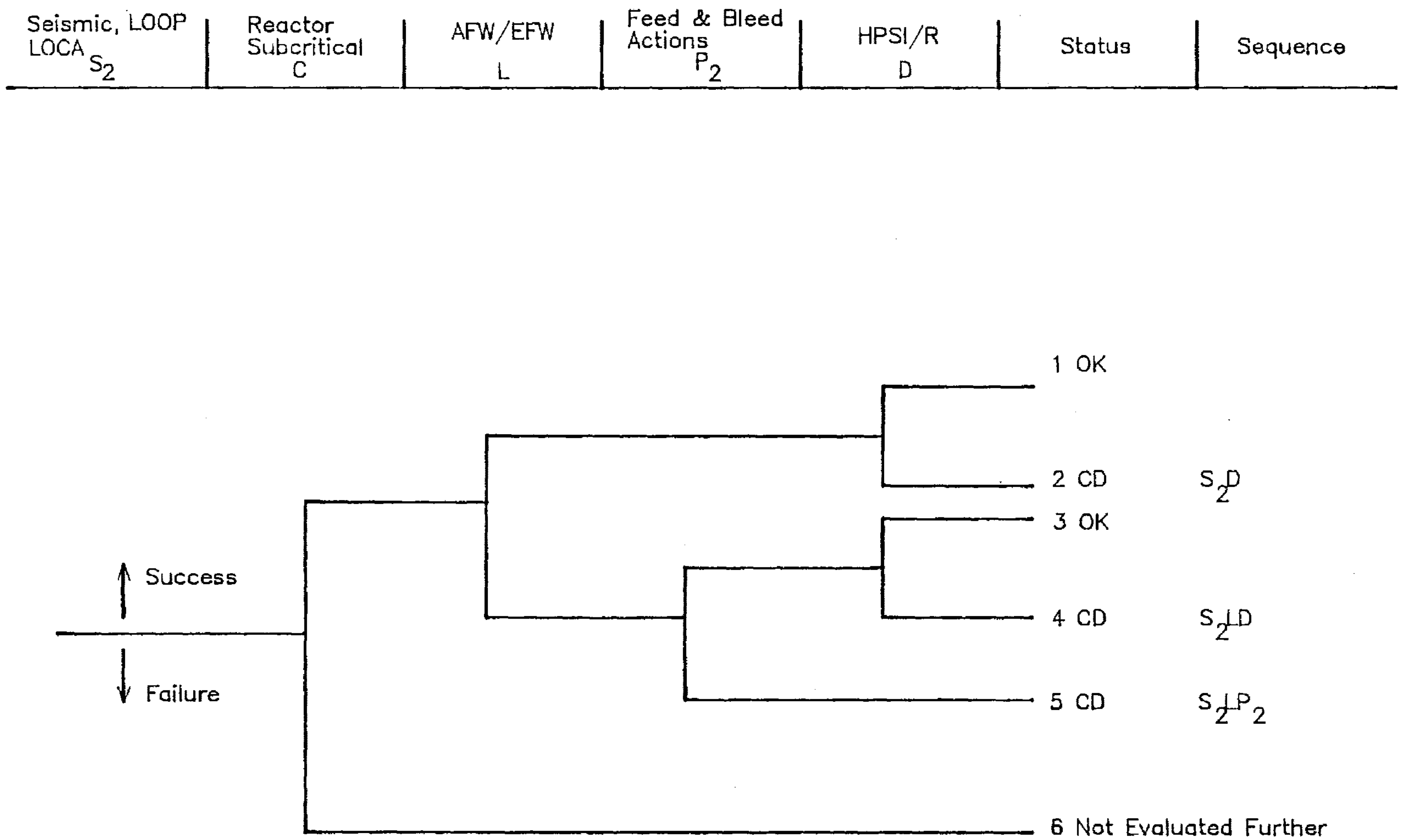


Figure 4–2 Seismic Event, LOOP Concurrent with Small LOCA Event Tree

5. ENGINEERING AND METHODOLOGY INSIGHTS

Based on the trial plant review, several insights into seismic margin studies have been gained. Section 5.1 discusses engineering and operational insights, and Section 5.2 discusses insights on the methodology and execution.

5.1 Engineering and Operational Insights

One of the most valuable results of a seismic margins review is finding and strengthening components and structures that may contribute to lower seismic capacity. Most of the items discussed below were noted during the two walkdowns or during the systems analysis, although a few, like the installation of new station batteries, were already planned. These changes have either already been performed or will be accomplished during the next refueling outage in March 1987.

1. The station service transformers, X-507 and X-608, transform power from the 4160-V emergency buses to the 480-V emergency buses. As discussed in Chapter 4, failure of these transformers could lead to core damage in the case of a small LOCA, and contribute to core damage in the non-LOCA case. Because of their low seismic capacity, they are being upgraded during the outage.

2. Station batteries 1 and 3 are being replaced with new lead calcium batteries that have high seismic capacity. If the older lead-antimony batteries had failed due to a seismic event, core damage could have resulted.

3. Anchorages on the containment spray pump area fans FN-44A and B will be strengthened. Failure of these fans could have led to long term heat-up and failure of the containment spray pumps, with subsequent failure of high pressure safety recirculation if recovery actions were not effective.

4. Block wall VE 21-1 will be strengthened to prevent its potential collapse from failing the containment spray area fans FN-44A and B discussed above.

5. Block wall VE 21-3 will be strengthened to prevent its potential collapse from rupturing a SCC pipe, and thereby failing SCC. This failure by itself would not have caused core damage.

6. The anchorages of the chillers for the computer room air conditioners AC-1A and B and the lab air conditioner AC-2 will be strengthened. Failure of the heat exchangers on these chillers could have failed the pressure boundary integrity of the SCC and PCC, and resulted in core damage upon loss of component cooling water.

7. A procedure was developed to isolate nonessential PCC lines and heat exchangers inside containment following a large earthquake if the PCC surge tank low level alarm annunciates. Although the PCC was designed to seismic standards, the project team could not enter

containment to verify the capacity of the components. Isolating the PCC lines provides assurance that any potential failure of the PCC pressure boundary integrity inside containment will not fail the entire PCC system. After examining the effect of the isolation procedure on the system fault trees and Boolean equation, the minimal cut sets showed that a single failure still existed due to selection of an isolation valve powered by the opposite redundant train. Failure of that one power train would have failed both the SCC and the isolation of PCC. The isolation procedure was revised accordingly to remove this potential single failure. This is one example of the ability of the fault tree logic modeling process to find small details that could affect seismic capacity, and to assist in formulating emergency procedures.

8. An unanchored monitor in the main control room panel was anchored. Its impact on other components and the seismic capacity of the plant therefore did not have to be evaluated.

9. The emergency lights were strapped and anchored.

10. A missing bolt on the anchorage of some level transmitters for the RWST was replaced.

11. Loose pressurized gas cylinders, a welding machine, and some heavy parts near the containment spray pump area fans were moved or tied securely.

12. Anchorages for the DG day tanks were strengthened

As can be seen, some of these changes had a major impact on the plant seismic capacity, while others probably did not. However, the changes brought about by the formal exercise of performing walkdowns and developing and solving the system fault trees demonstrates the power of the techniques, and the usefulness of a seismic margins review.

5.2 Insights on the Methodology and Execution

5.2.1 System Classification and Screening Guidelines

Overall, the guidance given in NUREG/CR-4334 and -4482 was very helpful in classifying the systems according to "Group A" and "Not-A." However, there are three areas where the trial plant review has provided more insight.

The first is with respect to the safety function of reactor subcriticality. This function can be performed by either the insertion of the control rods, or by boron injection using the boric acid transfer (BAT) system. Although the control rod drive mechanisms are screened out in Table 5-1 of NUREG/CR-4334 for review earthquake levels of 0.3 g, the reactor internals are not screened out based on insufficient information. This meant that both the reactor internals and the BAT system components were included in the information gathering process before and during the first walkdown. The BAT system is fairly complex, with numerous components that are not initially screened out. Appreciable resources were expended in gathering information concerning the BAT system. This was unnecessary after the seismic capacity review of the reactor internals. It would probably be more efficient in future seismic margin reviews if initial effort is placed in verifying that the reactor internals have high capacity, and only look at alternate means of subcriticality, such as the BAT system, if this is not the case. If necessary, this examination of alternate systems could be accomplished during the second walkdown.

The other two insights concern the safety function of emergency core cooling (early). Guidance in NUREG/CR-4334 and -4482, based on evaluation of previous PRAs for PWR plants, states that the emergency core cooling (early) function is included in Group A, while the emergency core cooling (late) function is in Group Not-A, and therefore is screened out of the analysis. There is the caveat that this screening is conditional on not finding any extremely gross plant-specific differences.

The systems analysis team therefore included the initial switchover phase from emergency core cooling injection (early) to emergency core cooling recirculation (late) as a screening verification step in the first plant walkdown. While the guidelines are ambiguous, this screening verification included long-term area cooling for the recirculation systems. It was determined that the containment spray pump area cooling fans FN-44A and B, and a block wall near the fans, VE-21, could not be screened out based on the first walkdown. Based on the plant Boolean equation for small LOCA, both of these items were single failures resulting in core damage in the long term if not recovered. HCLPFs for these items had to be calculated in order to determine plant seismic capacity. As noted above, the utility will make changes to these items to increase their capacity so that they do not impact overall plant capacity. Based on these findings, guidance in NUREG/CR-4334 and -4482 should be revised to insure that potential failures such as these are explicitly evaluated during a seismic margins review.

For the transient case (no LOCA), the emergency core cooling (early) function is defined in NUREG/CR-4334 and 4482 as achievement of residual heat removal. The AFW or EFW system at Maine Yankee or other PWR plants will achieve this balance within the first hour. For most PWR plants, irrecoverable failure of the emergency ac power system (station blackout) will not prevent the turbine-driven AFW train from performing early residual heat removal, and therefore satisfy the emergency core cooling (early) function. However, in the longer term without ac power, the station batteries would deplete, resulting in loss of instrumentation and AFW control power. Core damage could occur if dc power is not restored, and manual control of the turbine-driven AFW train or other feedwater source fails. Based on the guidelines, this long-term failure of AFW was screened out of the analysis. If it were assumed that battery depletion and loss of instrumentation and control power results in loss of AFW and other feedwater sources, then the Boolean expression for the no LOCA case would be dominated by the seismic failure singletons that result in station blackout:

- Failure of the SCC and PCC heat exchangers E-5A and 4B
- Failure of the station service transformers X-507 and 608
- Failure of the DG day tanks TK-62A and B
- Rupture of SCC and PCC because of chiller heat exchanger failure for the air conditioners AC-1A, 1B, and 2
- Structural failure of the circulating water pump house failing the SWS

Explicit guidance on the treatment of these long-term battery depletion sequences would be helpful.

5.2.2 Preparation for Walkdowns and Documentation

Based on the trial review, there are two areas where the project team experience could be helpful to future efforts. The first involves the identification of components to be evaluated during the first walkdown. The identification of the front-line system components was relatively straightforward because of the detailed nature of the available information, and the small number of components. However, identification of support system components was more difficult. This is because these systems are generally more complex, have many more components and branches, and are not generally documented as well. In addition, the references concerning interfaces between the front-line systems and support systems are often ambiguous. Finally, the actual physical nature and location of some items, such as distribution cabinets and panels, is not shown on plant drawings or documentation. Based on this experience, there are two recommendations. First, when reviewing the plant information, emphasize the interfaces with support systems such as ac and dc power, cooling water systems, HVAC systems, and instrument air systems. Document the ambiguities for later clarification. Second, plan to spend considerable effort tracing down these support system components during the first walkdown, and be prepared to make substantial revisions to the component list.

The second insight concerns documentation and information transfer between the systems analysis team and the fragilities team. Many of the components identified by the systems analysis team for HCLPF screening or evaluation were selected because of the potential for component rupture to cause flow diversion and consequent system failure. The component itself was not needed to fulfill a safety function, but the integrity of the component pressure boundary had to be assured for overall system success. The common example was heat exchangers for nonessential equipment whose seismic rupture would fail a necessary cooling water system. Since it can make a difference to the HCLPF assessment, the systems team must make a clear differentiation between components that are required to function for system success, and components that are required only to maintain pressure boundary integrity.

5.2.3 Systems Analysis and Pruning Process

There are a number of insights concerning the systems analysis (fault tree) and pruning process which may be helpful to future seismic margin reviews.

1. The procedure developed to isolate the PCC lines and components inside containment, and the automatic rupture isolation system on the SCC greatly reduced the number of components that had to be considered for HCLPF evaluation. Both the systems analysis and fragilities analysis efforts would have been larger, and eventually a containment walkdown might have been necessary.

2. As discussed above, verifying the seismic capacity of the reactor internals to allow subcriticality decreased the effort which would have been required for the BAT system evaluation.

3. Early evaluation and screening out of potential recovery actions and alternate systems, such as the small positive displacement pump for core cooling injection, reduced the number of components that required systems and fragility evaluations.

4. Being able to define all the components on one skid as one supercomponent, such as the DGs, reduces the systems analysis effort, but the evaluation for the fragility team may not be reduced.

5. As the fault trees are developed, it is useful to keep a list of the failure modes which should be considered for each component. For example, the pump failure modes include fail to start, fail to run, and test or maintenance outage, as well as seismic failure, but the pump appears only once on the initial fault tree. This information is needed later for the quantification process.

6. In the initial trees it is necessary to include all the components that require support systems, including those components such as motor-operated or air-operated valves that will likely be screened out later because of their high HCLPF and low nonseismic unavailability. Otherwise, if they are pruned from the initial trees, their dependency on support systems may be overlooked in the rest of the analysis. Also, since physical interactions between the component and structures, such as block walls or restraints, must be checked, it is better to include the component in the initial fault trees.

7. Although a few seismic interactions, such as the possible impacts of potential seismically induced fires, were not evaluated, the potential for firewater piping ruptures to damage equipment was reviewed. The DG control panels and distribution panels were located under firewater piping that had considerable lateral sway. Upon investigation, however, it was determined that the piping was dry, and two signals would be required to fill the piping. The probability of inadvertent actuation was therefore negligible. The PCC and SCC pumps were also located under sprinkler nozzles, but the motor housings for these pumps were designed to prevent water from entering. Therefore, ruptures of firewater piping were not evaluated to impact seismic capacity.

5.2.4 Minimal Cut Set Evaluation Process

Minimal cut sets were developed at four stages in this project.

1. Front-line system level cut sets using partially pruned fault trees, including their support systems, were developed just before the second walkdown to provide some guidance to the fragility team. These pointed out some potentially important system minimal cut sets. Although plant or sequence level cut sets could have been of additional assistance, because the fault trees were still fairly large the number of minimal cut sets would have been large as well. The additional effort to develop plant level cut sets before the second walkdown is not judged to be an effective allocation of resources, but the effort to develop system cut sets is effective.

2. Immediately following the second walkdown, the fault trees were pruned based on the information gathered. Sequence and plant level minimal cut sets were then developed and transferred to the fragility team for use in calculating the preliminary plant HCLPF. Because some component HCLPFs were not yet calculated, these cut sets were still considered preliminary.

3. The plant Boolean equations containing the minimal cut sets were revised by hand just prior to the draft report to the Peer Review Group to include additional fragility information.

4. Some final plant Boolean equations were developed by computer for this report.

Although this is more effort than originally planned, it is probably similar to that required in future seismic margin reviews.

5.2.5 Schedule and Resources

The schedule for the systems analysis tasks was judged to be adequate. The resources expended for each task are presented in Table 5-1. these resources were adequate for an experienced systems analysis team. About 10 percent of these manhours are support and clerical resources.

Table 5-1. Systems analysis resource expenditure.

TASK		MANHOURS
1	COLLECT INFORMATION	
1.1	First Round Information	20
1.2	Additional Specific Information	34
1.3	Visit AE	8
2	REVIEW PLANT INFORMATION	
2.1	Review EQ Level	0
2.2	Initial Systems Review	60
2.3	Indentify Components for Group A	36
2.4	Initial Screening of Components	50
2.5	Design Analysis/Seismic Reports	0
3	PLANT WALKDOWNS	
3.1	Target Areas for First Walkdown	50
3.2	Perform First Walkdown	84
3.3	Simplified Analysis	36
3.4	Document First Walkdown	34
3.5	Perform Second Walkdown	26
4	SYSTEMS MODELING	
4.1	Develop Event Trees and Fault Trees	539
4.2	Derive Accident Sequences	70
4.3	Boolean Equation/Minimal Cut Sets	50
5	SEISMIC MARGIN EVALUATION	
5.1	HCLPF - Components	
5.1.1	HCLPF - CDFM	0
5.1.2	HCLPF - FAM	0
5.2	HCLPF - Plant Capacity	10
6	REPORTING	
6.1	Internal Review of Report	22
6.2	Letter and Draft Final Report	253
7	MEETINGS	
7.1	Project Team Meetings	104
7.2	Peer Review Group Meetings	22
7.3	NRC/ACRS/Expert Panel Meetings	24
	TOTAL	1532

REFERENCES

Battle, R.E., Emergency Diesel Generator Operating Experience, 1981-1983, NUREG/CR-4347, Oak Ridge National Laboratory, Oak Ridge, Tennessee, July, 1985.

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

Carlson, D.D., Interim Reliability Evaluation Program Procedures Guide, NUREG/CR-2728, SAND82-1100, Sandia National Laboratories, Albuquerque, New Mexico, January, 1983.

Fleming, K.N. et al., Classification and Analysis of Reactor Operating Experience Involving Dependent Events, EPRI-NP-3967, Electric Power Research Institute, Palo Alto, California, June, 1985.

Fletcher, D.C., Accident Mitigation Following a Small Break with Coincident Failure of Charging and High Pressure Injection for the Westinghouse Zion PWR, EGG-CAAD-5428, EG&G Idaho Inc., Idaho Falls, Idaho, April 1981.

Kolb, G.J., Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One - Unit 1 Nuclear Power Plant, NUREG/CR-2787, Sandia National Laboratories, Albuquerque, New Mexico, June, 1982.

Nuclear Regulatory Commission, PRA Procedures Guide, NUREG/CR-2300, Washington, D.C., January, 1983.

Prassinos, P.G., Ravindra, M.K., and Savy, J.B., Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants, NUREG/CR-4482, UCID-20579, Lawrence Livermore National Laboratory, Livermore, California, March, 1986.

APPENDIX A

IDENTIFIERS AND SYMBOLS

Table A-1. Fault tree symbols.

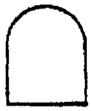
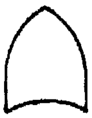
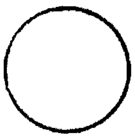
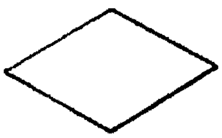
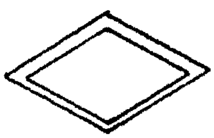
	AND GATE (AG)	- Output fault occurs if all of the input faults occur.
	OR GATE (OG)	- Output fault occurs if at least one of the input faults occur.
	BASIC EVENT (BE)	- An initiating fault requiring no further development.
	UNDEVELOPED EVENT (UE)	- An event which is not developed further, because it is of insufficient consequence or information is unavailable.
	DEVELOPED EVENT (DE)	- An event that could be further developed or is developed elsewhere but is treated here as a primary event.
	DESCRIPTION	- Contains the description of an event.
	TRANSFER IN	- Indicates that the tree is developed further at the occurrence of the corresponding TRANSFER OUT.
	TRANSFER OUT	- Indicates that this portion of the tree must be attached at the corresponding TRANSFER IN.

Table A-2. Fault tree event identifiers.

All basic, developed, undeveloped events and tab-or gates are to be coded with the following format.

XXX-YYY-ZZ-AAAAA

where

XXX	=	System Identifier
YYY	=	Event & Component Type Identifier
ZZ	=	Failure Mode Code
AAAAA	=	Selected by analyst, try to indicate what the failure involves (e.g., rather than PS10, use PTRNA)

Gates should be labeled with an alphanumeric ID, e.g., AFW1, HPI5, etc.

Table A-2 (Cont'd)

System Identifiers	
ACP	AC Power System
AFW	Auxiliary Feedwater System
CIS	Containment Isolation System
CSS	Containment Spray System
DCP	DC Power System
HPI	High Pressure Safety Injection System
MCR	Main Control Room
OEP	Onsite Electric Power System
PCC	Primary Component Cooling Water System
PPS	Primary Pressure Relief System
RAS	Recirculation Actuation System
RPS	Reactor Protection System
SCC	Secondary Component Cooling Water System
SIS	Safety Injection Actuation System
SWS	Service Water System

Table A-2 (Cont'd)

Event and Component Type Identifiers	
Air Cooling Heat Exchanger	ACX
Sensor/Transmitter Units:	
Flow	FST
Level	LST
Physical Position	ZST
Pressure	PST
Radiation	RST
Temperature	TST
Flux	NST
Circuit Breaker	BKR
Calculational Unit	CAL
Electrical Cable	CBL
Signal Conditioner	CND
Control Rods:	
Hydraulically-Driven	CRH
Motor-Driven	CRM
Ducting	DCT
Motor-Driven Compressor	MDC
Motor-Driven Fan	FAN
Fuse	FUS
Diesel Generator	DGN
Hydrogen Recombiner Unit	HRU
Heat Exchanger	HTX
Inverter	INV
Electrical Isolation Device	ISO
Air Cleaning Unit	ACU
Load/Relay Unit	RLY
Logic Unit	LOG
Local Power Supply	LPS

Table A-2 (Cont'd)

Event and Component Type Identifiers	
Motor-Generator Unit	MGN
Motor-Operated Damper	MOD
Solenoid-Operator Damper	SOD
Pumps:	
Engine-Driven	EDP
Motor-Driven	MDP
Turbine-Driven	TDP
Manual Control Switch	XSW
Rectifier	REC
Transfer Switch	TSW
Transformer	TFM
Tank	TNK
Bistable Trip Unit	TXX
Air Heating Unit	AHU
Electrical Bus - DC	BDC
Electrical Bus - AC	BAC
Manual Damper	XDM
Pneumatic/Hydraulic Damper	PND
Battery	BAT
Valves:	
Check Valve	CKV
Hydraulic Valve	HDV
Safety/Relief Valve	SRV
Solenoid-Operated Valve	SOV
Motor-Operated Valve	MOV
Manual Valve	XVM
Air-Operated Valve	AOV
Testable Check Valve	TCV
Explosive Valve	EPV

Table A-2 (Cont'd)

Event and Component Type Identifiers	
Filter	FLT
Instrumentation and Control Circuit	ICC
Strainer	STR
Heater Element	HTR
Pipe Segment	PSF
Pipe Train	PTF
Actuation Segment	ACS
Actuation Train	ACT
AC Electrical Train	TAC
DC Electrical Train	TDC
Block Wall	BKW
Operator Action	XHE
Common Cause Event	CCF
Miscellaneous Aggregation of Events	VFC
Phenomenological Events	PHN

Table A-2 (Cont'd)

Failure Mode Codes*	
Valves, Contacts, Dampers	
Fail to Transfer	FT
Normally Open, Fail Open	OO
Normally Open, Fail Closed (Position)	OC
Normally Closed, Fail Closed	CC
Normally Closed, Fail Open	CO
Valves, Filters, Orifices, Nozzles	
Plugged	PG
Pumps, Motors, Diesels, Turbines, Fans, Compressors	
Fail to Start	FS
Fail to Continue Running	FR
Sensors, Signal Conditioners, Bistable	
Fail High	HI
Fail Low	LO
No Output	NO
Segments, Trains, and Miscellaneous Agglomerations	
Loss of Flow, No Flow	LF
Loss of Function	FC
Actuation Fails	FA
No Power, Loss of Power	LP
Failure (for miscellaneous fault agglomerations not based on segments or trains)	VF
Hardware	HW

* Grouping of failure modes by events or components are only suggestions. The failure modes listed may be used for any applicable event or component type.

Table A-2 (Cont'd)

Failure Mode Codes*	
Battery, Bus, Transformer	
No Power, Loss of Power	LP
Short	ST
Open	OP
Tank, Pipes, Seals, Tubes, Walls	
Leak	LK
Rupture	RP
Seismic Failure	EQ
Human Errors	
Fail to Operate	FO
Miscalibrate	MC
Fail to Restore from Test or Maintenance	RE
Normal Operations (unavailable due to planned activity)	
Maintenance	MA
Test	TE
Test and Maintenance	TM
Non-Specified Failure	XX

* Grouping of failure modes by events or components are only suggestions. The failure modes listed may be used for any applicable event or component type.

APPENDIX B

AUXILIARY FEEDWATER SYSTEM

Table B-1. Auxiliary feedwater (AFW).

Safety Function:	Supply water to the steam generators to remove reactor decay heat when main feedwater is not available.	
System Components:		
Tanks:	TK-21	Demineralized Water Storage Tank
	TK-16	Primary Water Storage Tank
Pumps:	P-25A	Emergency Feed Pump
	P-25B	Auxiliary Feed Pump
	P-25C	Emergency Feed Pump
Turbines:	T-1	Turbine for P-25B (Powered from Main Steam)
Valves:	Refer to Valve Table	
Support Systems:		
AC Power:	P-25A	4160V Emergency Bus 6
	P-25C	4160V Emergency Bus 5
DC Power:	P-25A	125V DC Distribution Cabinet 3
	P-25C	125V DC Distribution Cabinet 1
Air:	TK-111	Control AOVs (AFW-A-101,201,301)
	TK-123	Isolation AOVs (AFW-A-338,339,340)
	TK-25	Turbine Steam Control Valves (MS-P-168, MS-T-163)
HVAC:	Pump Room	
Pump Cooling:	E-86A	P-25A Discharge Recirculation
	E-86B	P-25B Discharge Recirculation
	E-86C	P-25C Discharge Recirculation
Actuation:	Steam Generator Low Level	
Instrumentation:	Feedwater Control System (SG Level, Pressure)	

Table B-2. AFW valve table.

Valve	Description	Power (SOV)	Normal Position	Operating Position (Actuation)	Fail Position
AFW-A-101	Flow control to SG E-1-1	120VAC 1A (1201A1)	0	0	0
AFW-A-201	Flow control to SG E-1-2	120VAC 1A (1201B1)	0	0	0
AFW-A-301	Flow control to SG E-1-3	120VAC 1A (1201C1)	0	0	0
AFW-A-338	Flow control isolation valve (AFW-A-101)	120VAC 3A (1205A)	0	0	0
AFW-A-339	Flow control isolation valve (AFW-A-201)	120VAC 3A (1205B)	0	0	0
AFW-A-340	Flow control isolation valve (AFW-A-301)	120VAC 3A (1205C)	0	0	0
MS-A-173	AFW pump B turbine trip and throttle valve	mechanical	0	latched open	C
MS-P-168	Turbine steam supply pressure control	120 VAC Bus 4 (1106)	0	0	0
MS-T-163	Turbine steam supply control	125 VDC Batt 3 (1102)	0	C(SIAS/CIS)	C*

* Fails open on loss of solenoid power.

Table B-3. AFW cooling requirements.

P-25A	• Oil cooler E-86A • Recirculation to DWST
P-25B	• Oil cooler E-86B • Recirculation to DWST • Designed to operate under elevated temperature conditions
P-25C	• Oil cooler E-86C • Recirculation to DWST
T-1	• Does not have a high temperature interlock
Pump Room	• Open door, portable fan

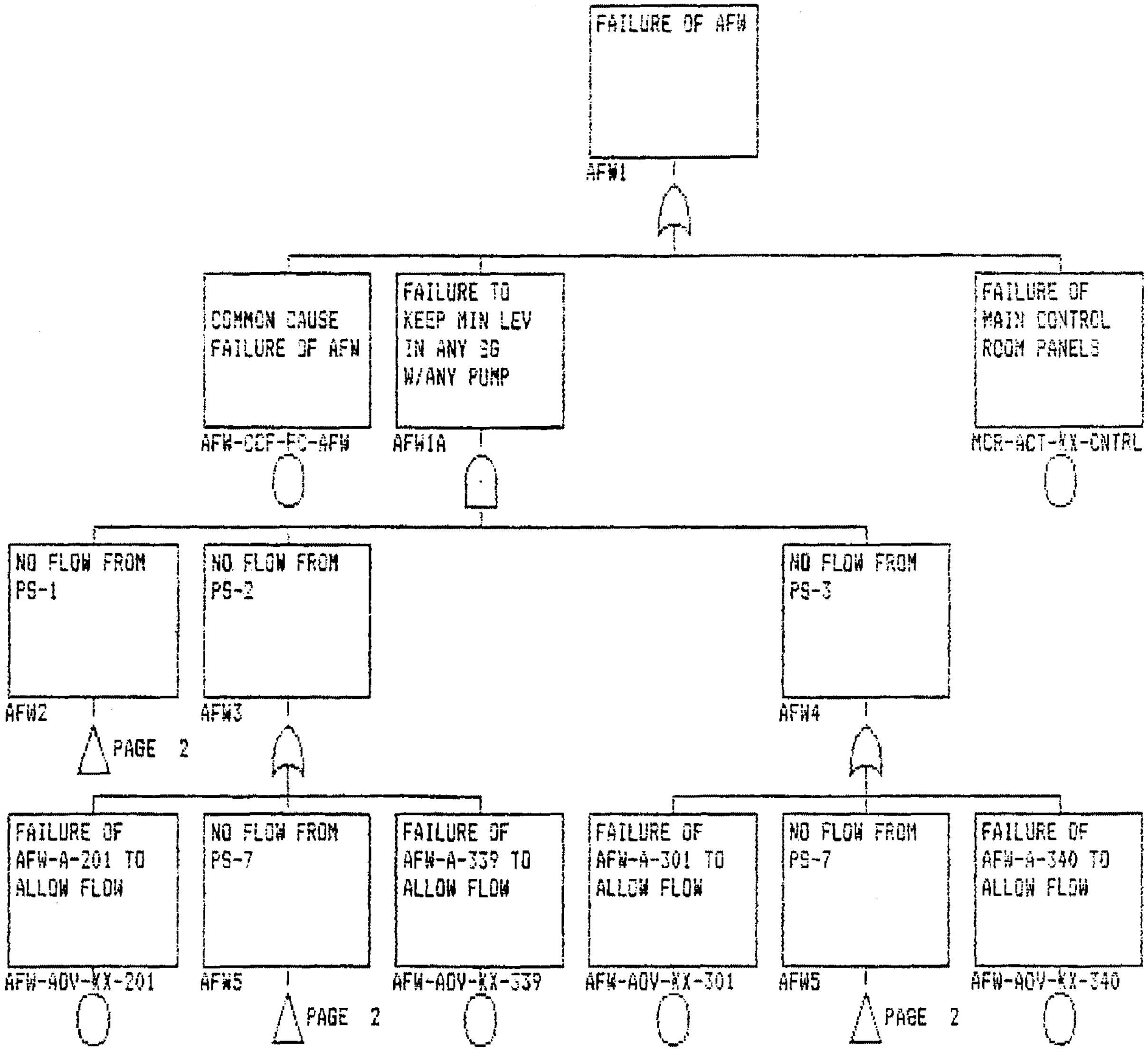


Figure B-1 Auxiliary Feedwater System Fault Tree.

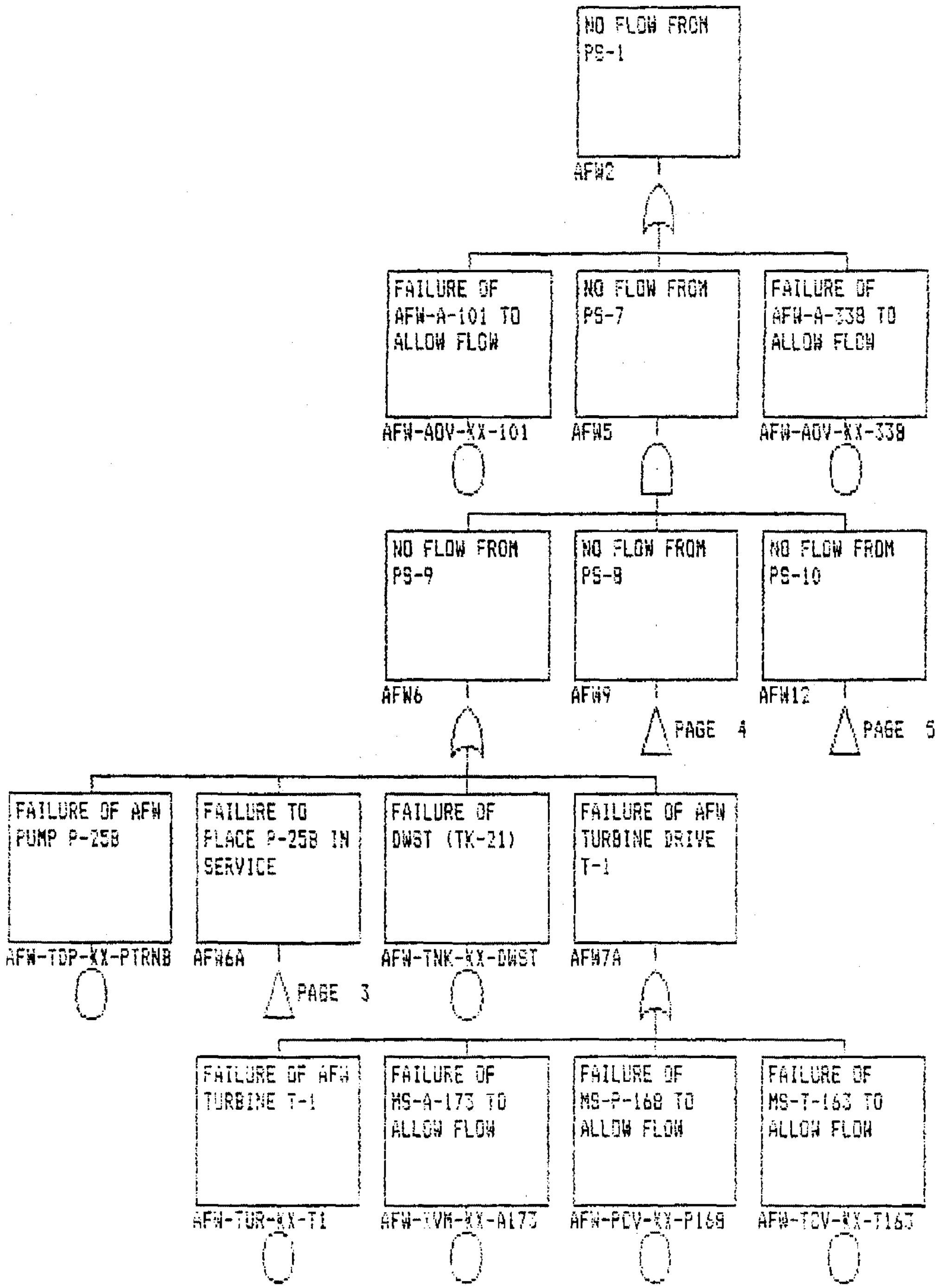


Figure B-1 (cont.)

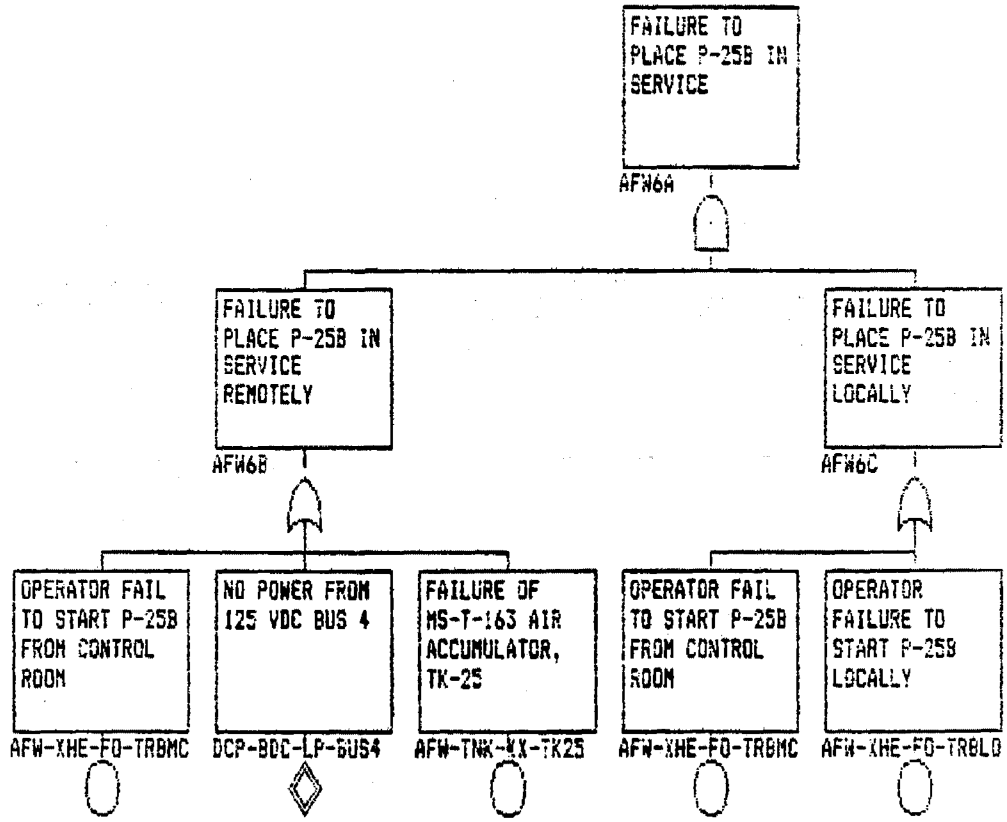


Figure B-1 (cont.)

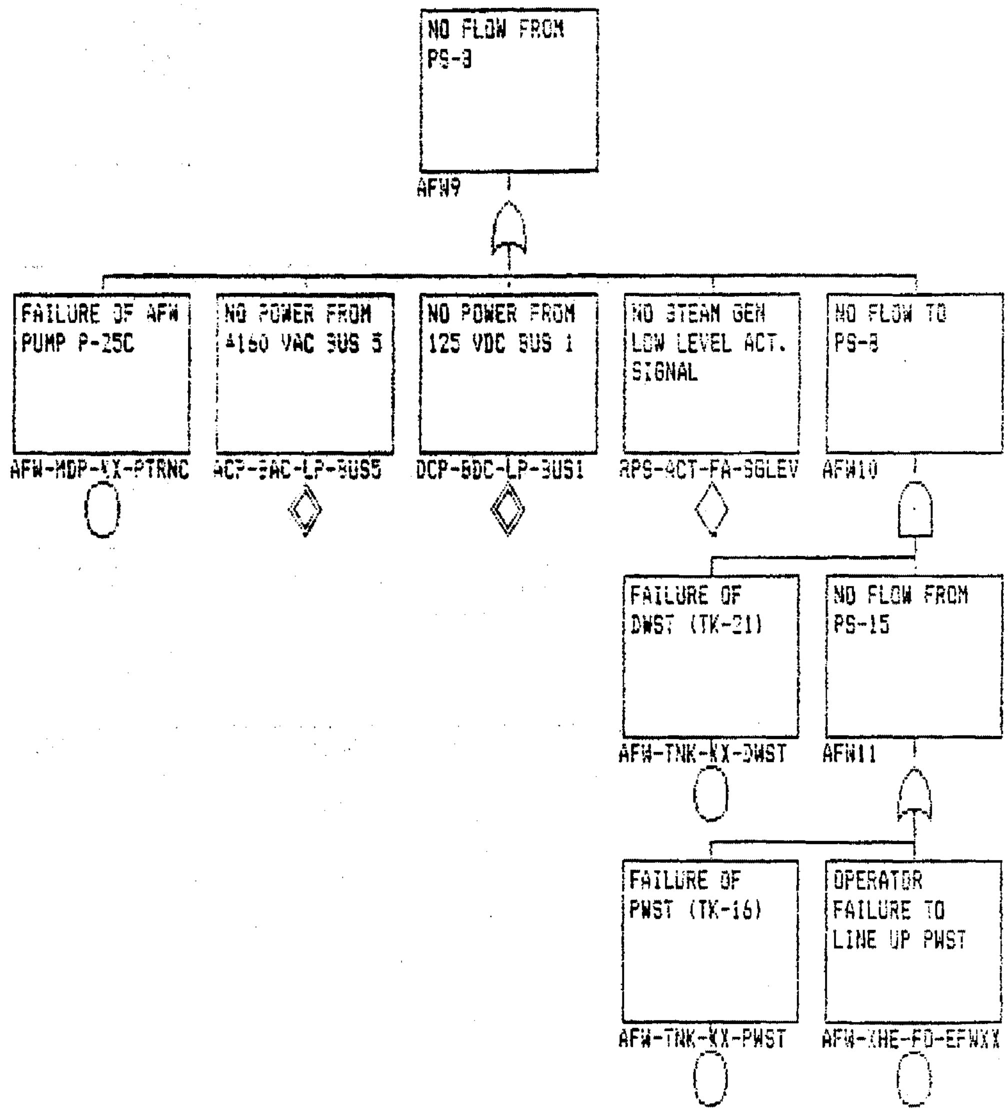


Figure B-1 (cont.)

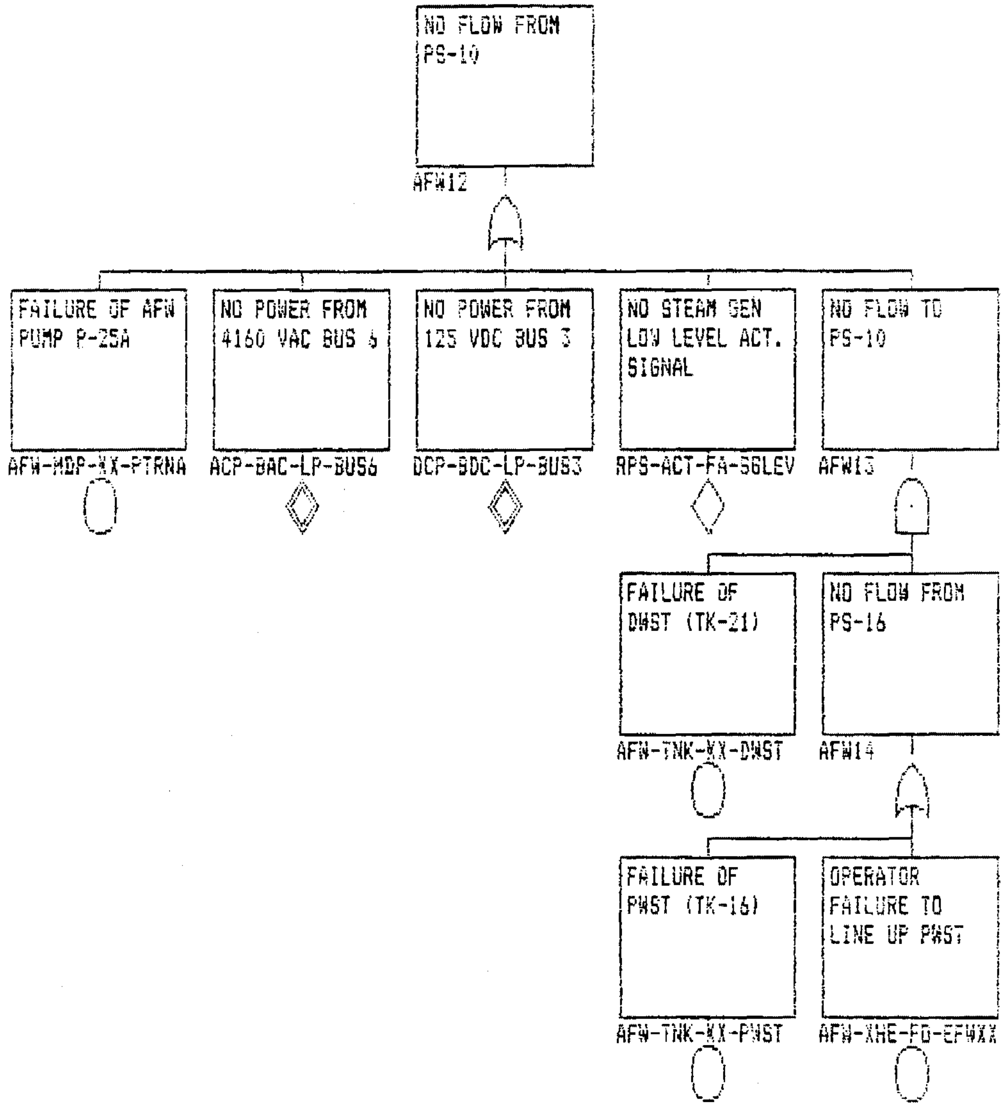


Figure B-1 (cont.)

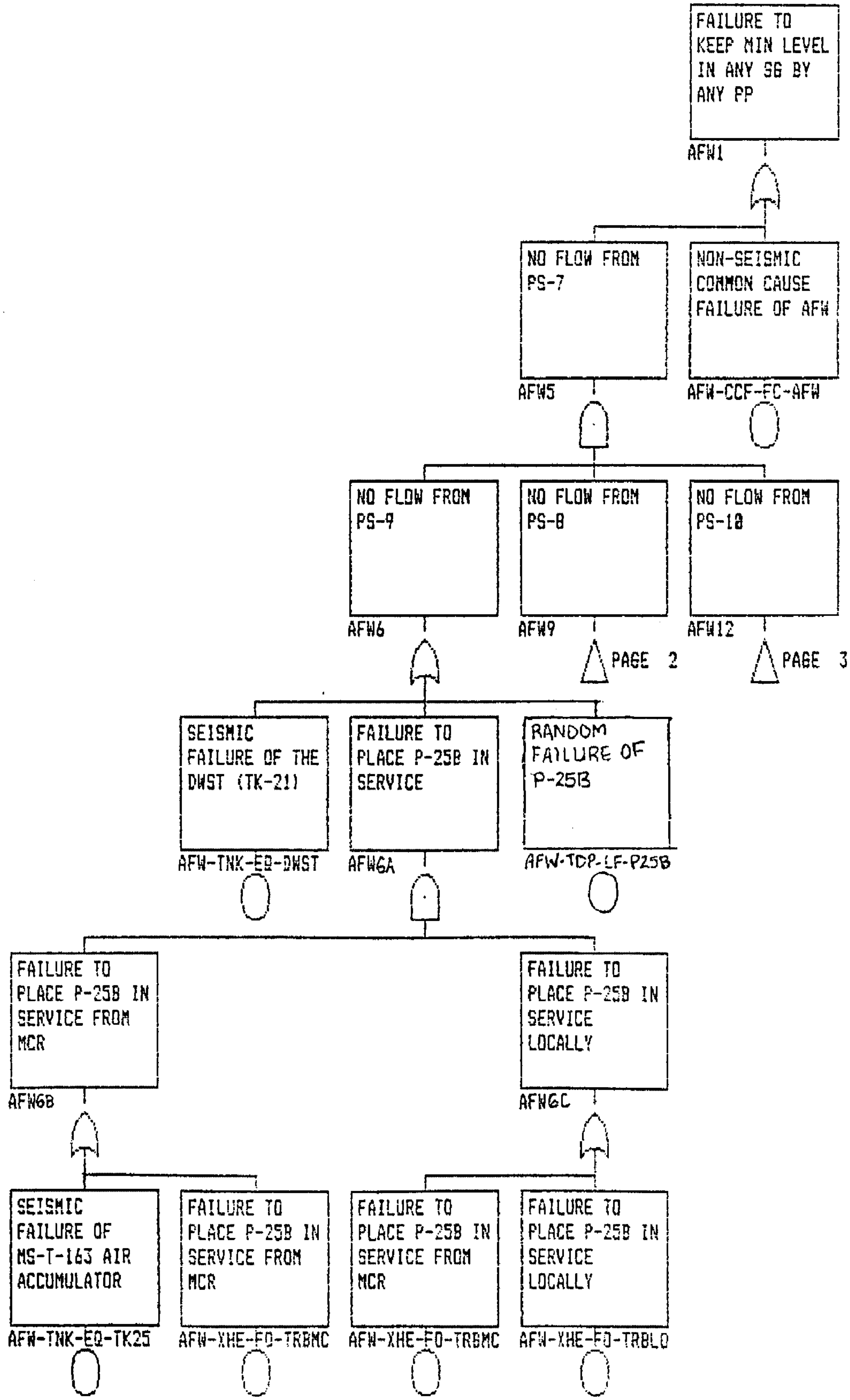


Figure B-2 Auxiliary Feedwater System Fault Tree, Pruned and Merged.

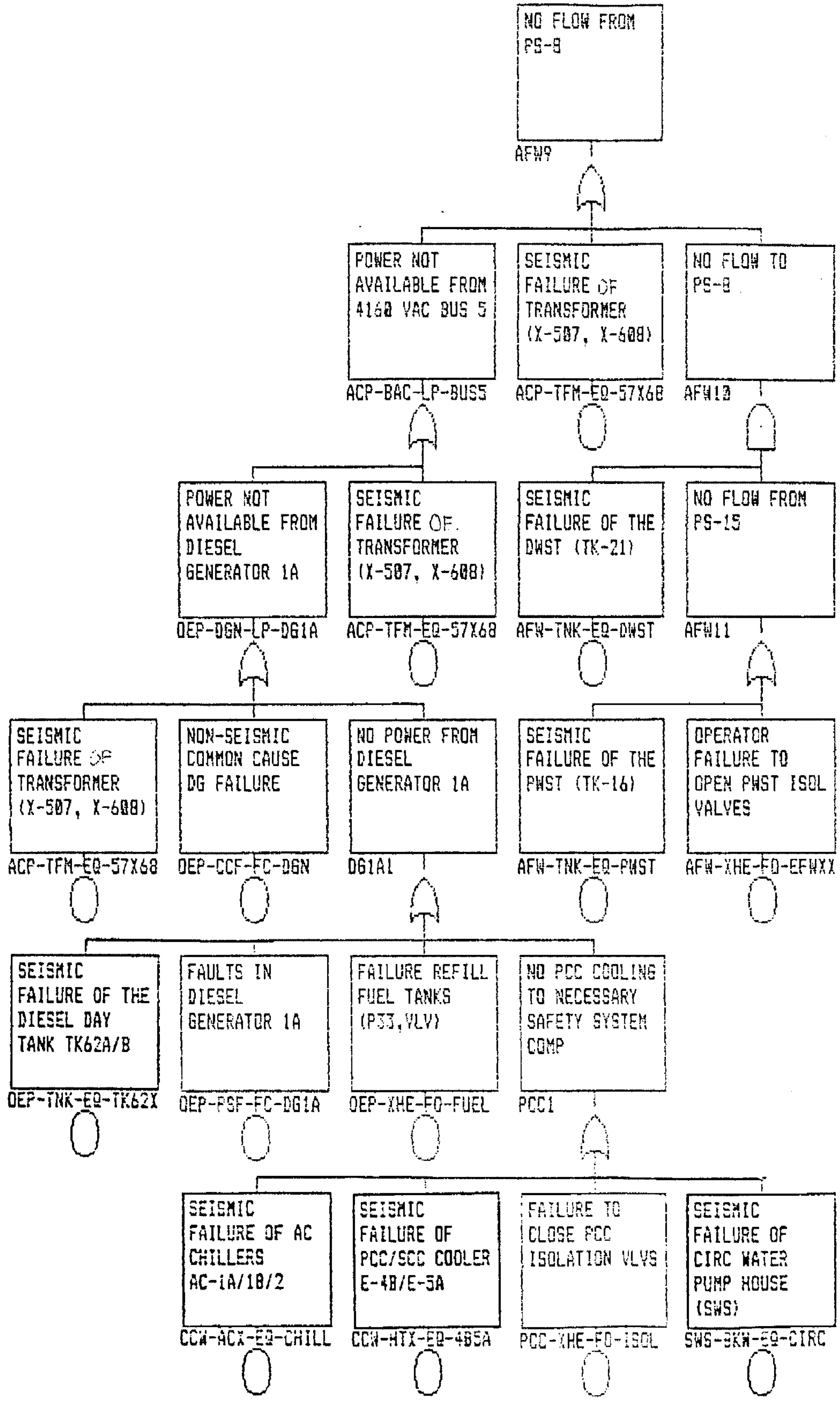


Figure B-2 (cont.)

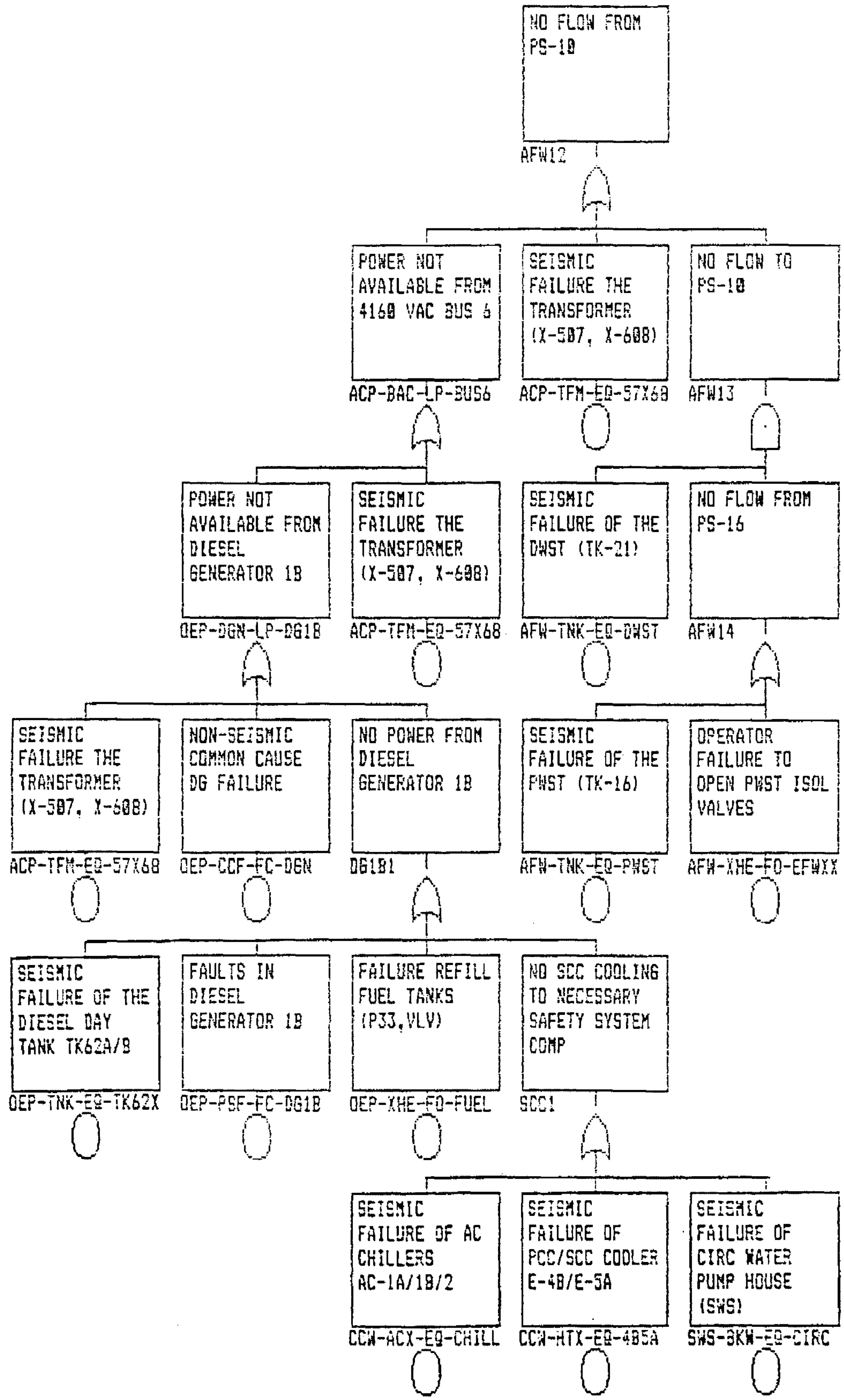


Figure B-2 (cont.)

APPENDIX C

HIGH PRESSURE SAFETY INJECTION SYSTEM

Table C-1. High pressure safety injection (HPSI) and containment spray pump area cooling (CSPPCL).

Safety Function:	Inject borated water into the reactor vessel immediately after a LOCA. Also for feed and bleed, post-accident core cooling and additional shutdown capability during rapid cooldown of RCS. Spray pump area cooling to ensure long-term availability of containment spray pumps.	
System Components:		
Tanks:	TK-4	Refueling Cavity Water Storage Tank
Pumps:	P-14A (N.O.)	Charging (HPSI) Pump
	P-14B (S)	Charging (HPSI) Pump
	P-14S (Spare)	Charging (HPSI) Pump
Fans:	FN-44A	Spray Pump Area Fan
	FN-44B	Spray Pump Area Fan
Valves:	Refer to Valve Table	
Support Systems:		
AC Power:	P-14A	4160V Emergency Bus 5
	P-14B	4160V Emergency Bus 6
	P-14S	4160V Emergency Bus 5/6
	FN-44A	480V Emergency MCC 7B
	FN-44B	480V Emergency MCC 8B
DC Power:	P-14A	125V DC Distribution Cabinet 1
	P-14B	125V DC Distribution Cabinet 3
	P-14S	125V DC Distribution Cabinet 1, 3
Pump Cooling:	P-14A	PCC
	P-14B	SCC
	P-14S	PCC
	Motors Air Cooled	
Actuation:	P-14A	SIAS A
	P-14B	SIAS B
	P-14S	SIAS A/B
	FN-44A	SIAS A
	FN-44B	SIAS B

Table C-1 (Cont'd)

Additional Components Whose Failure May Lead to HPSI Failure

E-34	Seal Water Heat Exchanger	Isolated by CH-A-32 and CH-A-33.
E-67	Reactor Coolant Regenerative Heat Exchanger	
E-96	Seal Water Heater	
FL-34B	Seal Water Supply Filter	
TK-54	Spray Chemical Addition Tank	Catastrophic failure may cause failure of RWST and/or interconnecting line.

Table C-2. HPSI/CSPPCL valve table.

Valve	Description	Power (SOV)	Normal Position	Operating Position (Actuation	Fail Position
CH-A-32	HPSI pump B discharge to charging header	125VDC DC/CE-1	O	C(SIAS-A)	C
CH-A-33	HPSI pump A discharge to charging header	125VDC DC/CE-1	O	C(SIAS-A)	C
CH-F-38	Inlet to charging header	125VDC DC/CE-2	O	C(SIAS-B)	C
HSI-M-11	HPSI train B to Loop 1 injection	MCC 8A	C	O(SIAS-B)	AI
HSI-M-12	HPSI train A to Loop 1 injection	MCC 7A	C	O(SIAS-A)	AI
HSI-M-21	HPSI train B to Loop 2 injection	MCC 8A	C	O(SIAS-B)	AI
HSI-M-22	HPSI train A to Loop 2 injection	MCC 7A	C	O(SIAS-A)	AI
HSI-M-31	HPSI train B to Loop 3 injection	MCC 8A	C	O(SIAS-B)	AI
HSI-M-32	HPSI train A to Loop 3 injection	MCC 7A	C	O(SIAS-A)	AI
HSI-M-40	HPSI train B discharge cross-connect to train A	MCC 8A	C	C*	AI
HSI-M-41	HPSI train A discharge	MCC 7A	C	O(SIAS-A)	AI
HSI-M-42	HPSI train B discharge	MCC 8A	C	O(SIAS-B)	AI
HSI-M-43	HPSI train A discharge cross-connect to train B	MCC 7A	C	C*	AI
HSI-M-50	RWST supply to HPSI pump A	MCC 7B1	C	O(SIAS-A) C(RAS-A)	AI

* Open for recovery action.

Table C-2 (Cont'd)

Valve	Description	Power (SOV)	Normal Position	Operating Position (Actuation)	Fail Position
HSI-M-51	RWST supply to HPSI pump B	MCC 8B1	C	O(SIAS-B) C(RAS)	AI
HSI-M-54	Recirculation supply to HPSI pump B	MCC 7B1	C	O(RAS-A)	AI
HSI-M-55	Recirculation supply to HPSI pump B	MCC 8B1	C	O(RAS-B)	AI
SL-P-3	RCP seal water inlet	125VDC DC/CE-2 (211)	O	C(SIAS-B)	O

Table C-3. HPSI cooling requirements.

P-14A	• Air-cooled motor • PCC cooled stuffing box (seals) • Shaft-mounted lube oil pump for bearings and gear (P-14A-2) • Electrical lube oil pump for standby cooling (P-14A-3)
P-14B	• Air-cooled motor • SCC cooled seals • Shaft-mounted lube oil pump (P-14B-2) • Electrical lube oil pump (P-14B-3)
P-14S	• Air-cooled motor • PCC or SCC cooled seals (PCC preferred) • Shaft-mounted lube oil pump (P-14C-2) • Electrical lube oil pump (P-14C-3)
Pump Cubicles	• Open area, natural circulation • Doors to fuel building, turbine building and the garage doors may be opened to increase flow.

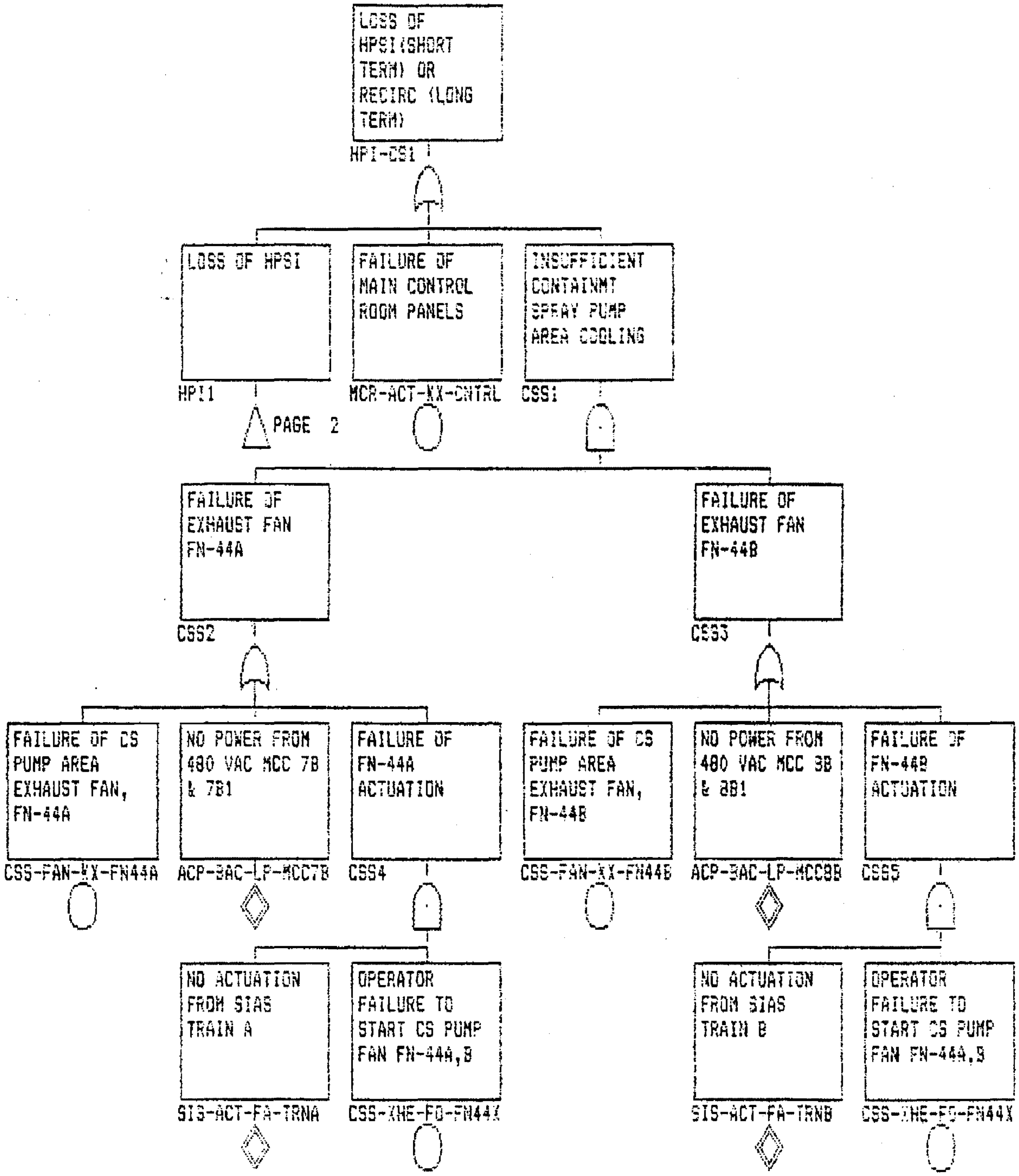


Figure C-1 High Pressure Safety Injection System Fault Tree.

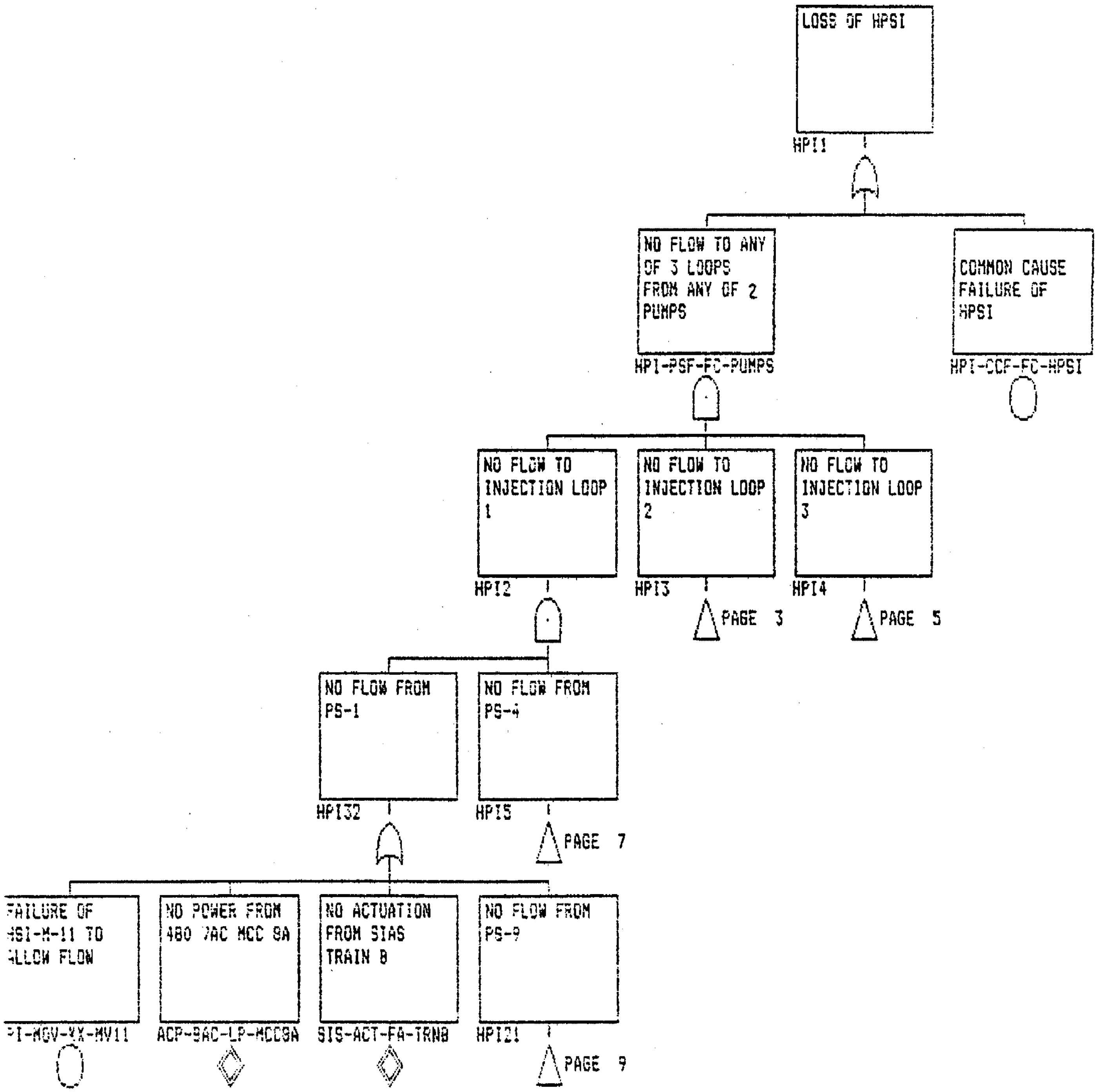


Figure C-1 (cont.)

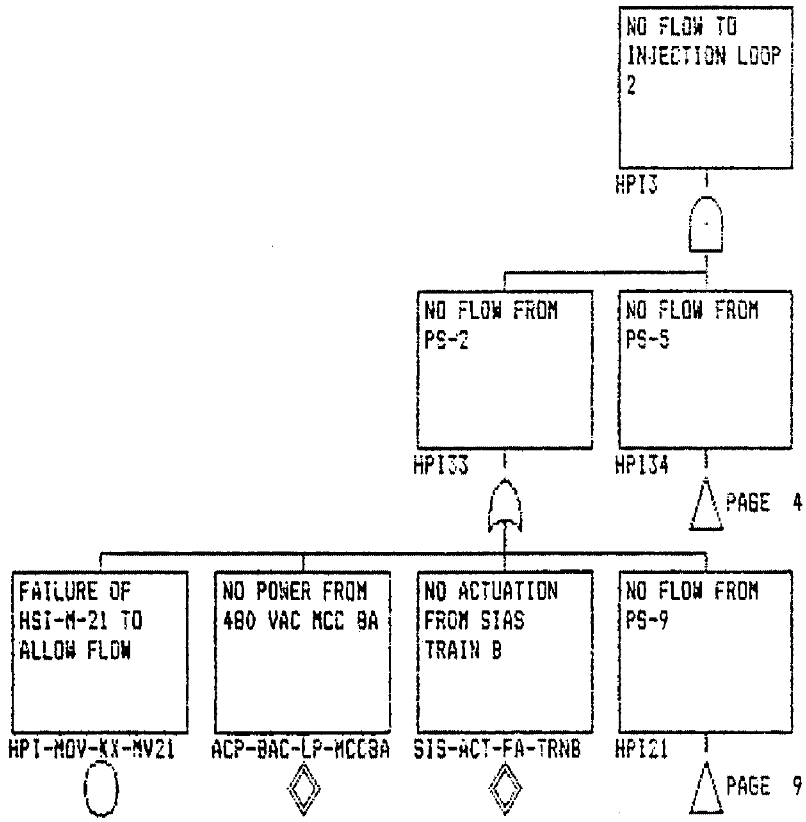


Figure C-1 (cont.)

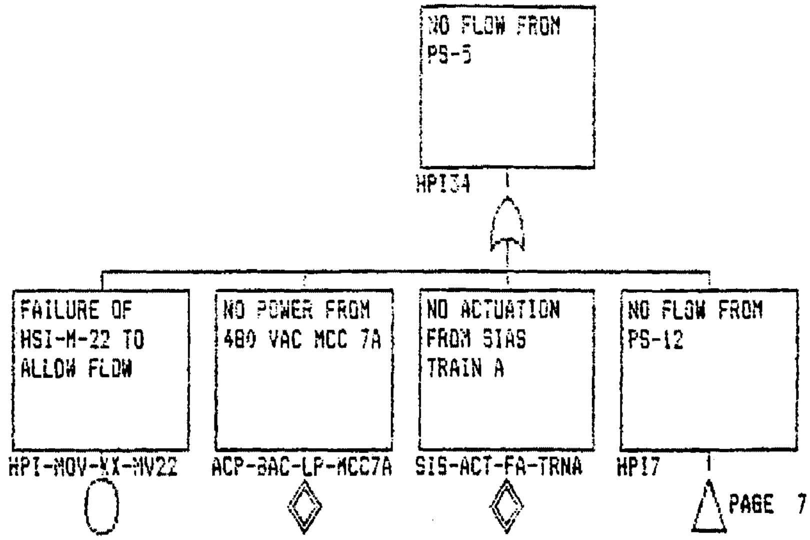


Figure C-1 (cont.)

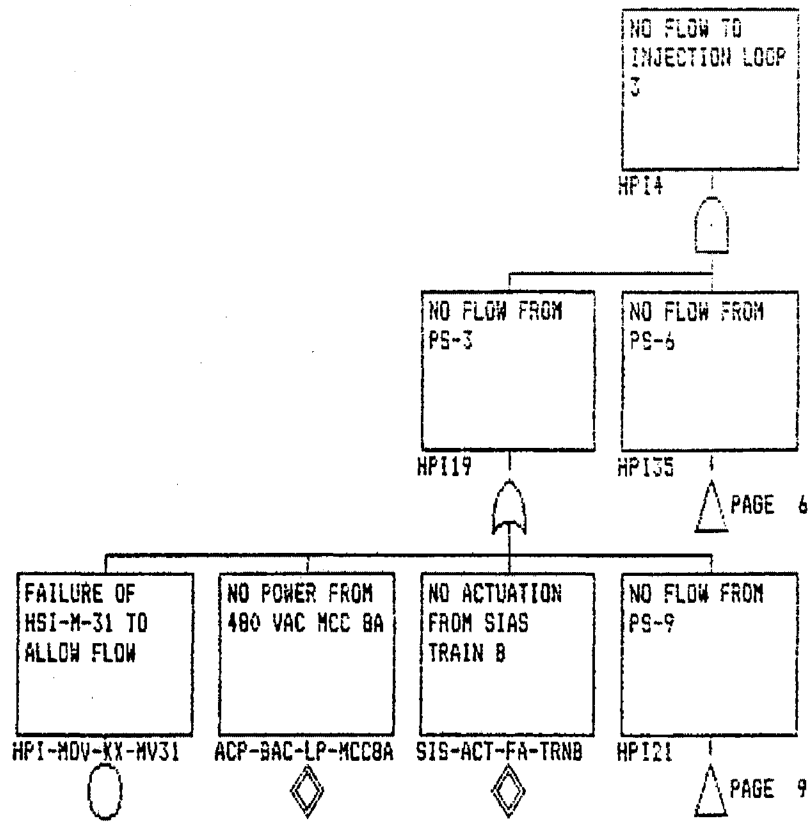


Figure C-1 (cont.)

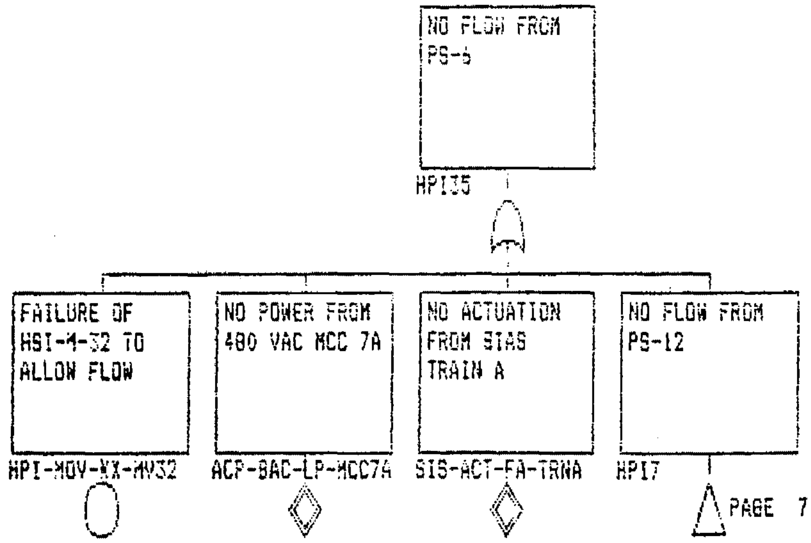


Figure C-1 (cont.)

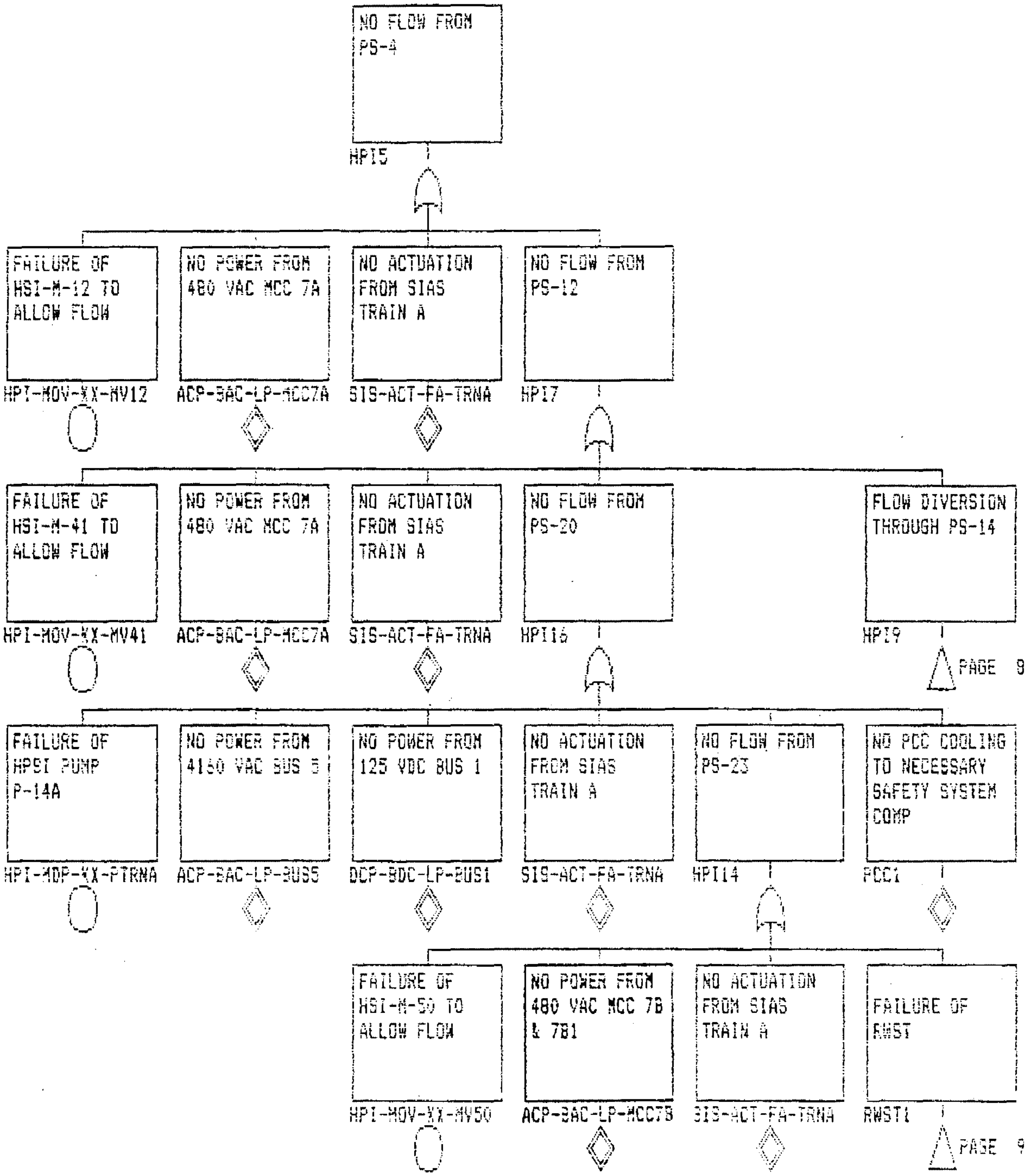


Figure C-1 (cont.)

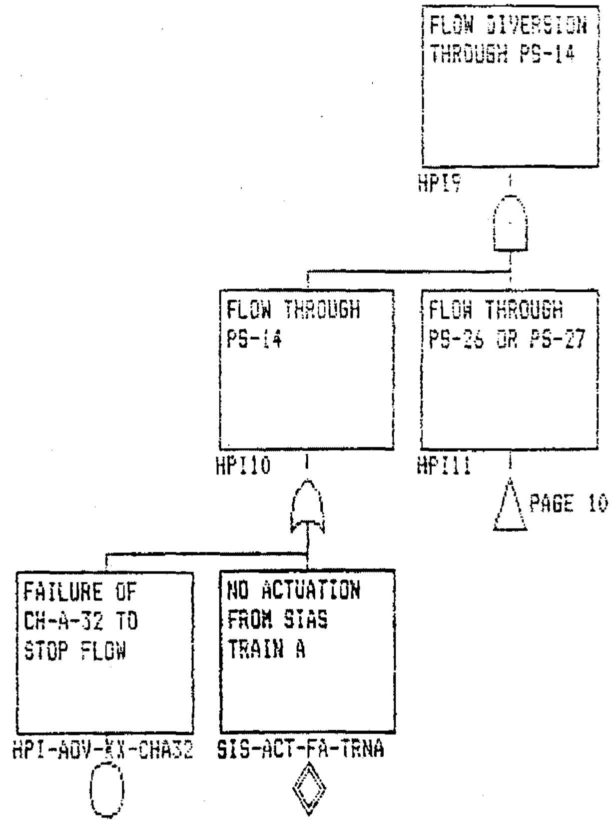


Figure C-1 (cont.)

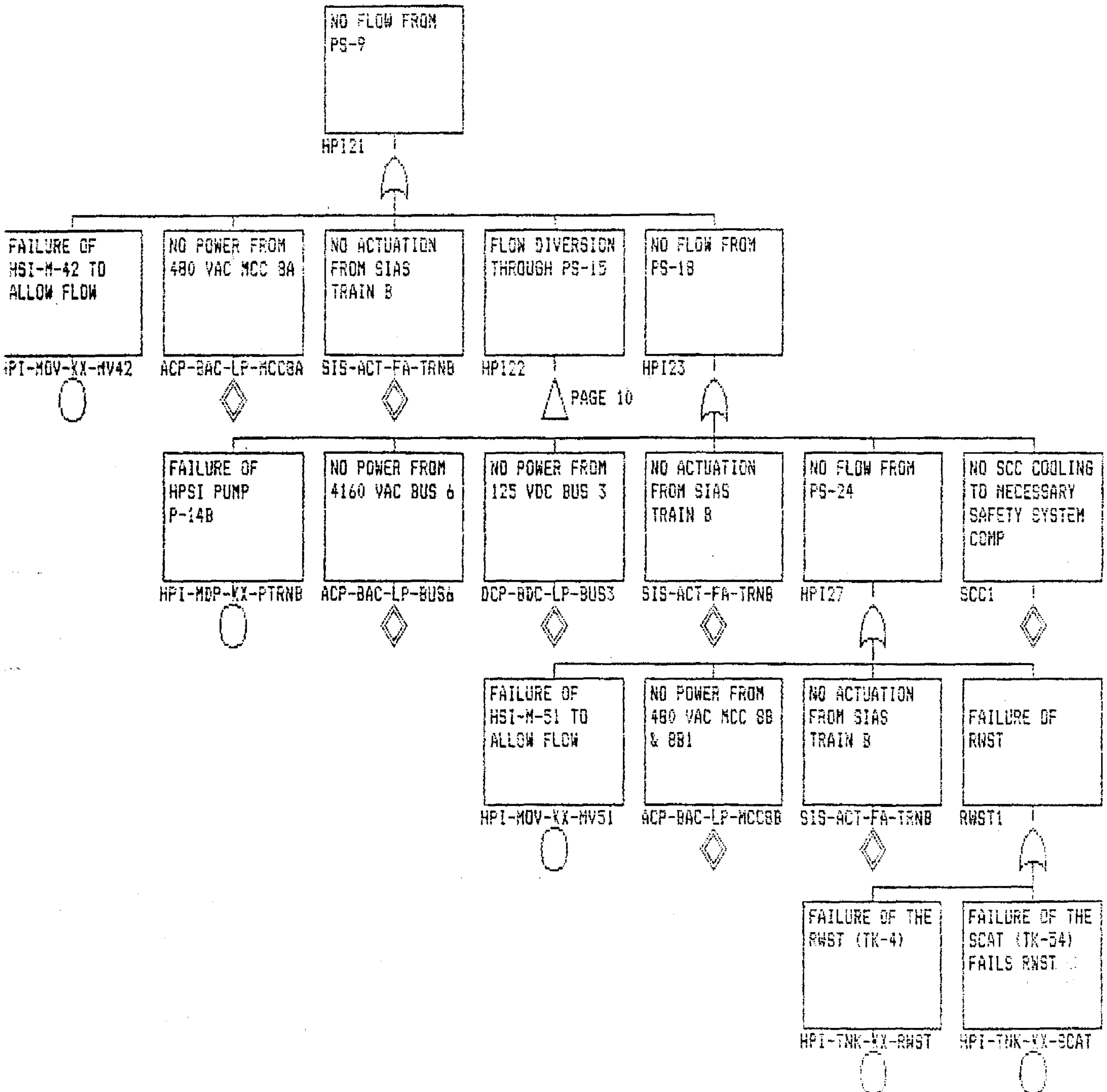


Figure C-1 (cont.)

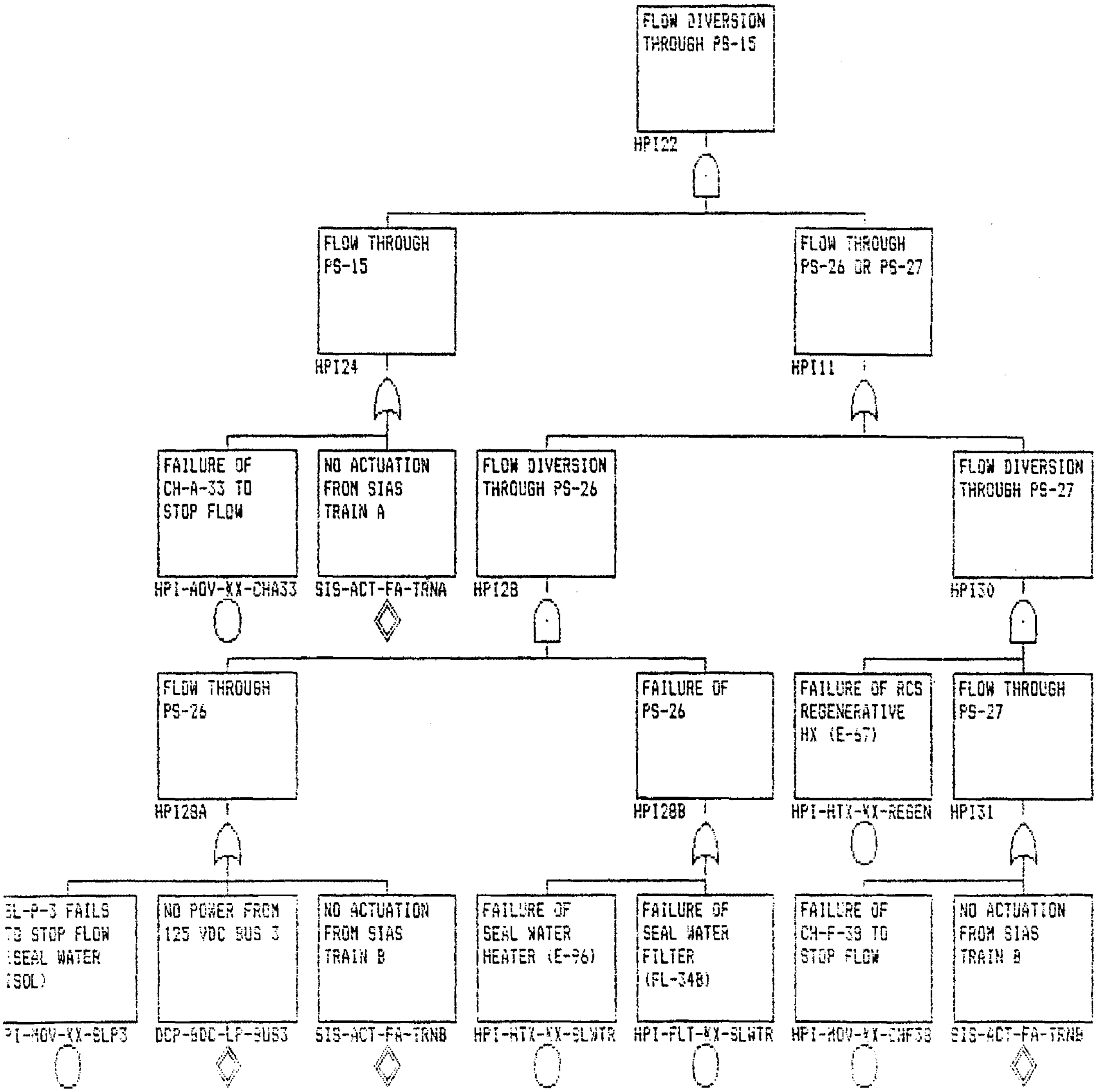


Figure C-1 (cont.)

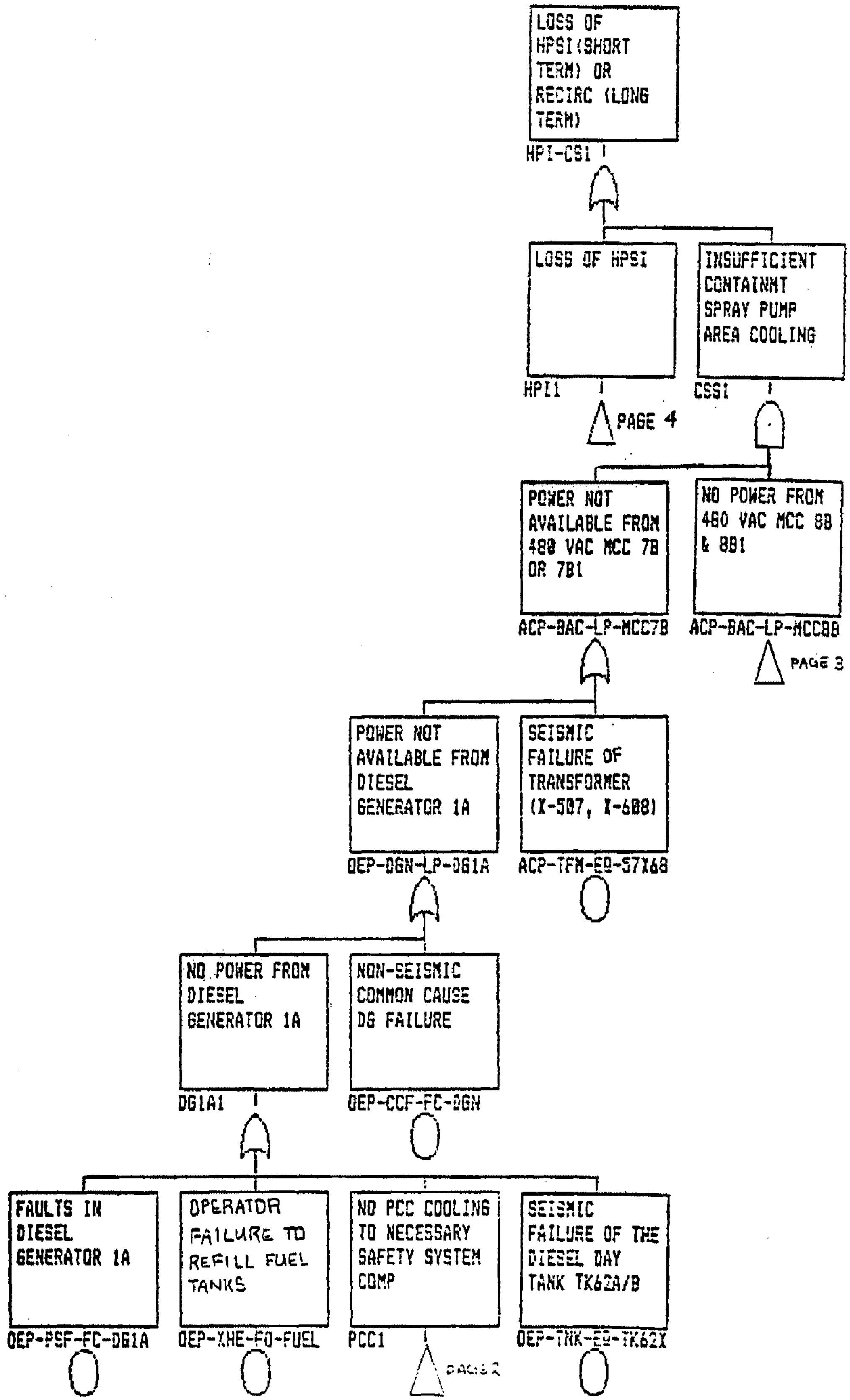


Figure C-2 High Pressure Safety Injection System System Fault Tree, Pruned and Merged.

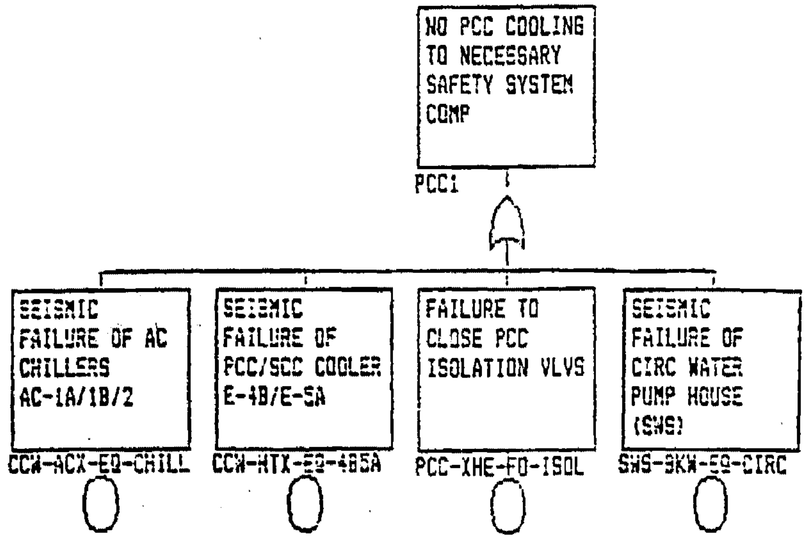


Figure C-2 (cont.)

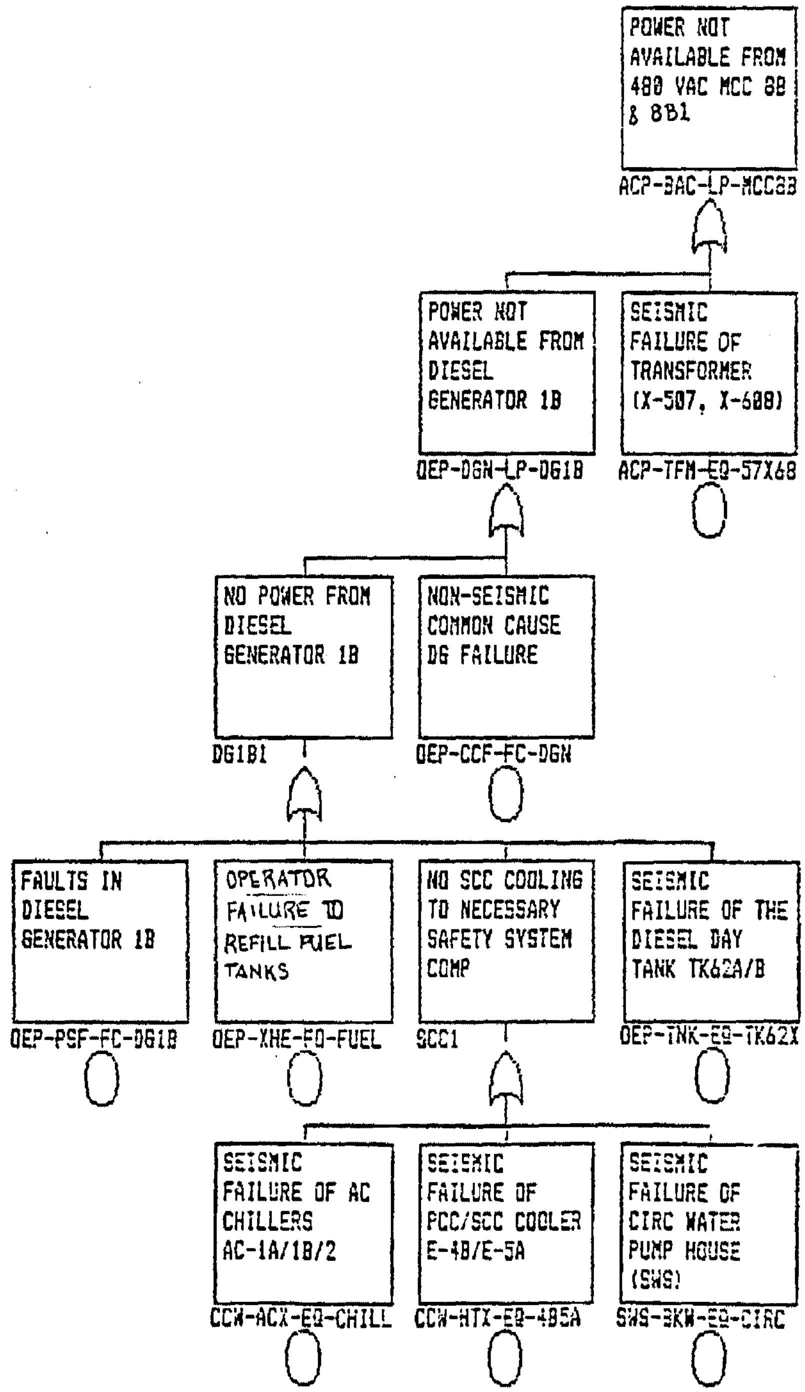


Figure C-2 (cont.)

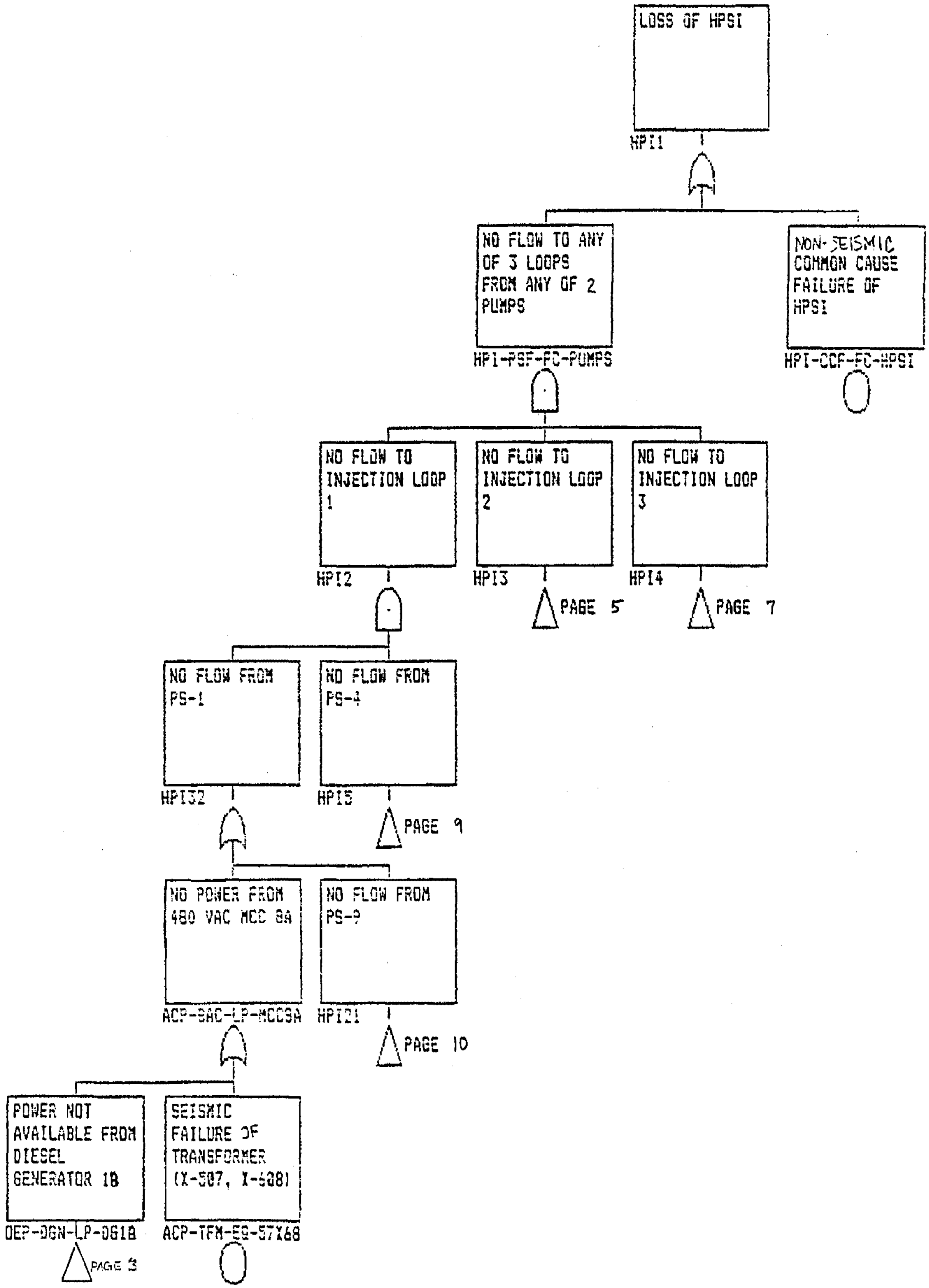


Figure C-2 (cont.)

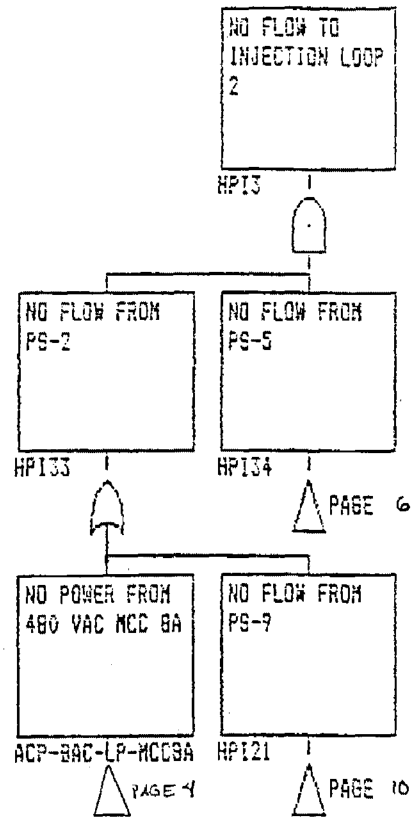


Figure C-2 (cont.)

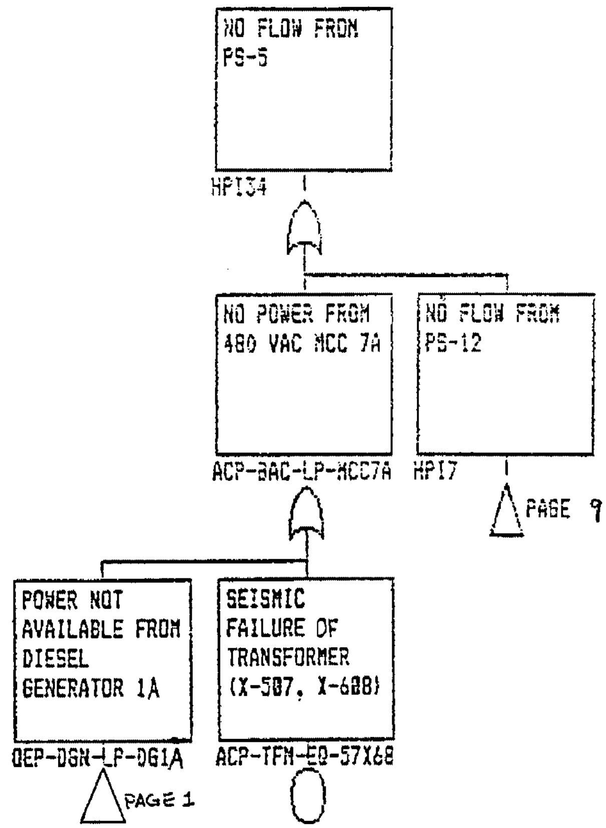


Figure C-2 (cont.)

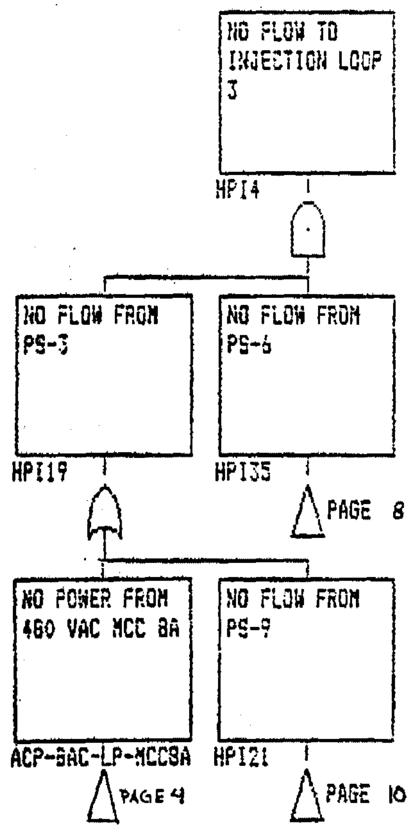


Figure C-2 (cont.)

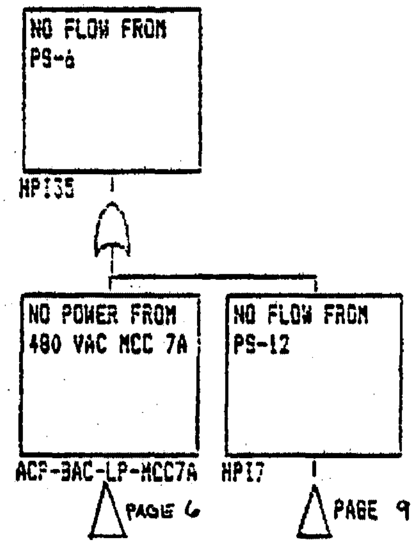


Figure C-2 (cont.)

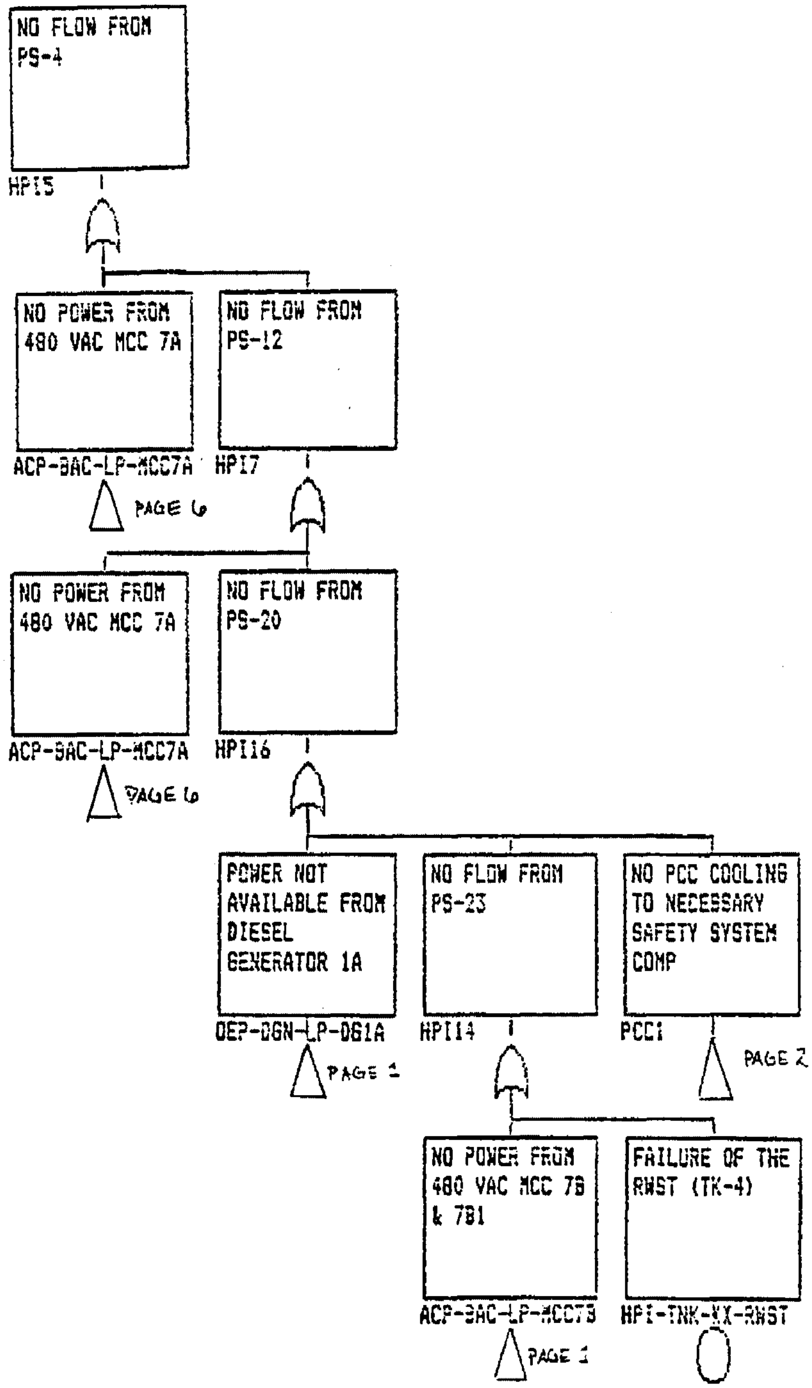


Figure C-2 (cont.)

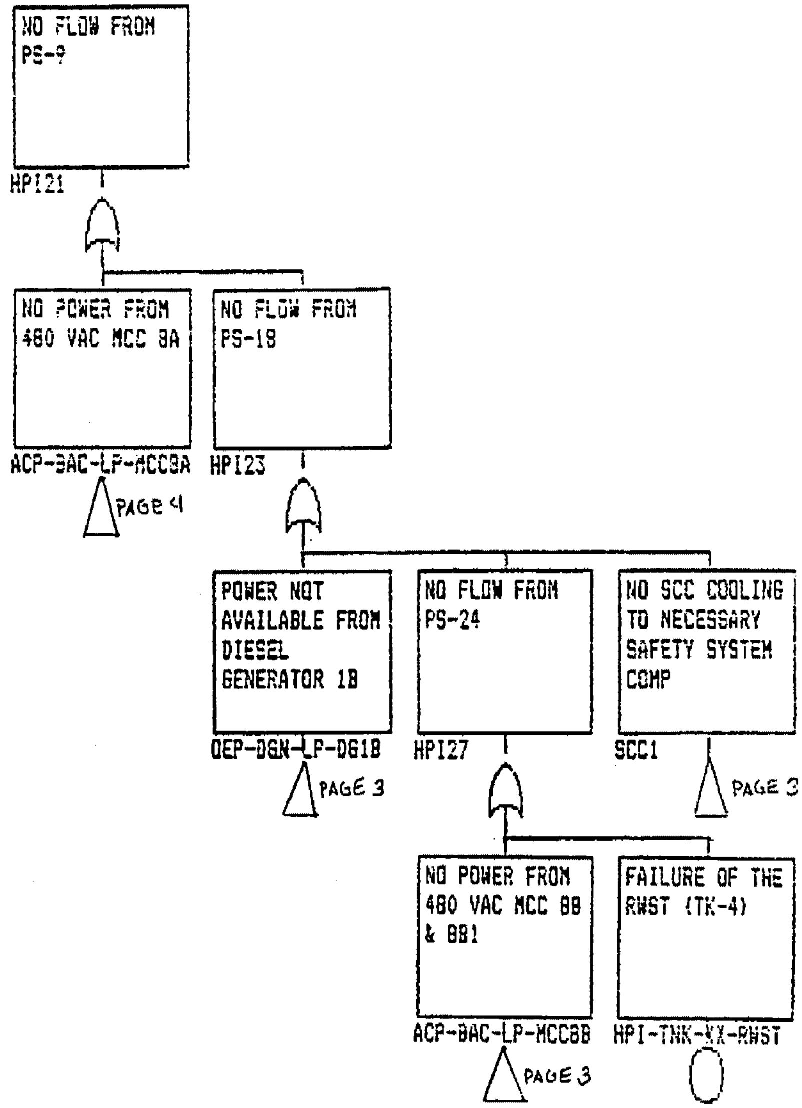


Figure C-2 (cont.)

APPENDIX D

PRIMARY PRESSURE RELIEF SYSTEM

Table D-1. Primary pressure relief system (PPS).

Safety Function:	To provide feed and bleed capability. Also provides reactor coolant system overpressure protection.	
System Components:		
Valves:	PR-S-14	Power-Operated Relief Valve
	PR-S-15	Power-Operated Relief Valve
	PR-M-16	PORV Isolation Valve
	PR-M-17	PORV Isolation Valve
Support Systems:		
AC Power:	PR-S-14	480V Emergency MCC 7B
	PR-M-16	480V Emergency MCC 7B
	PR-S-15	480V Emergency MCC 8B
	PR-M-17	480V Emergency MCC 8B
Actuation:	Actuated by operator for feed and bleed	

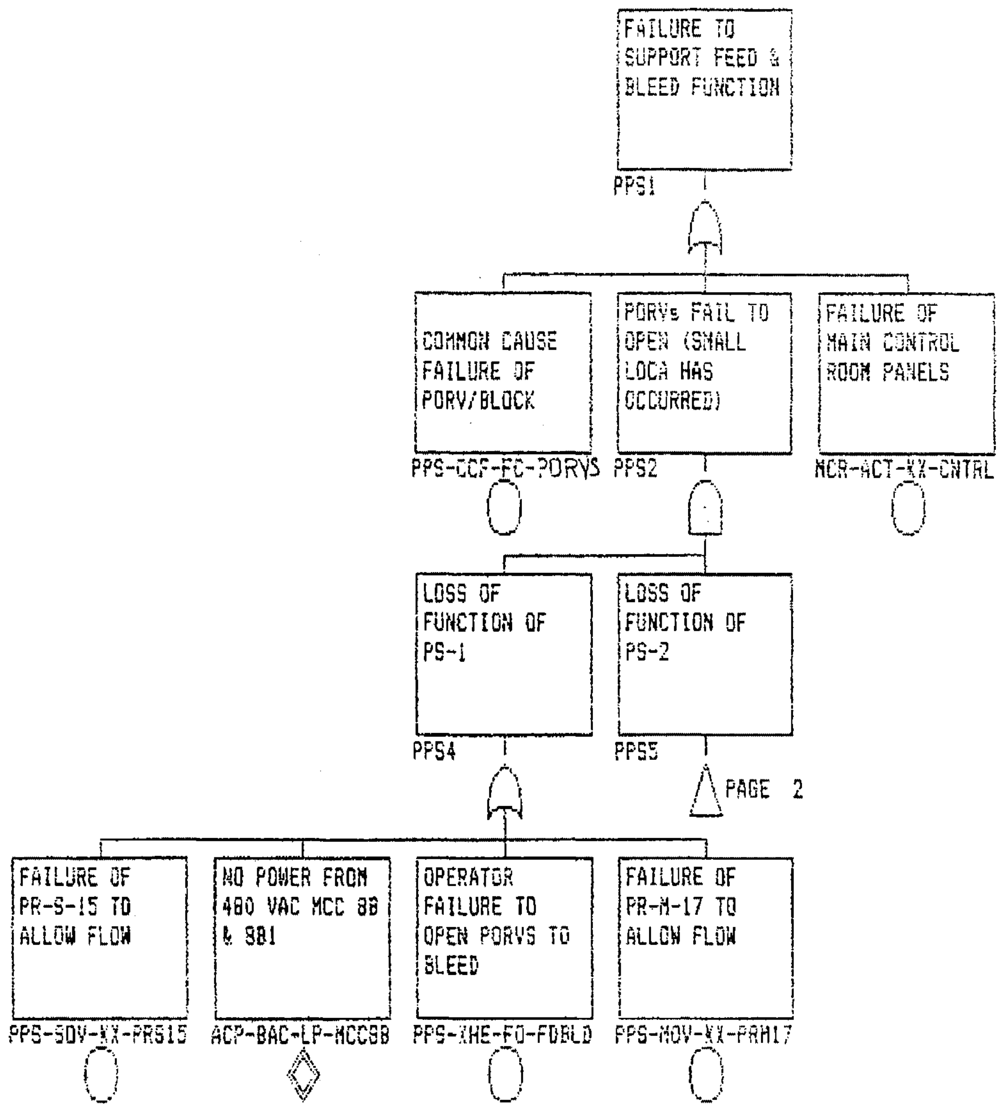


Figure D-1 Power-Operated Relief Valve Fault Tree.

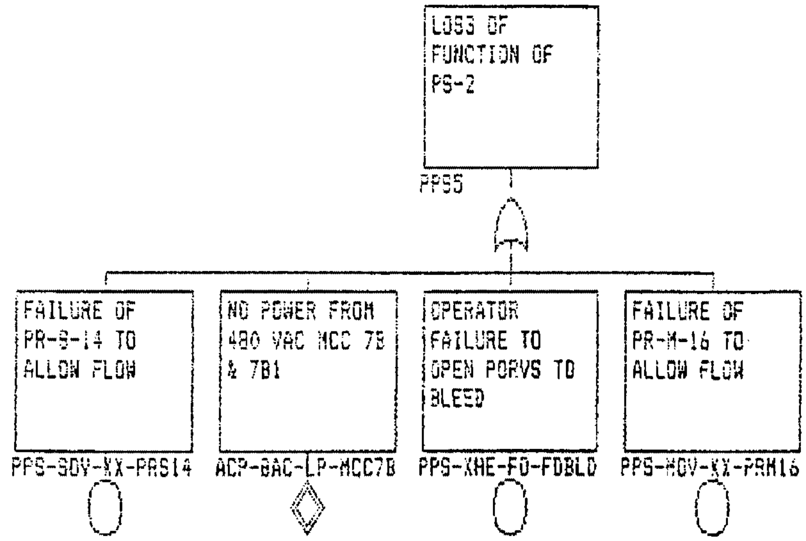


Figure D-1 (cont.)

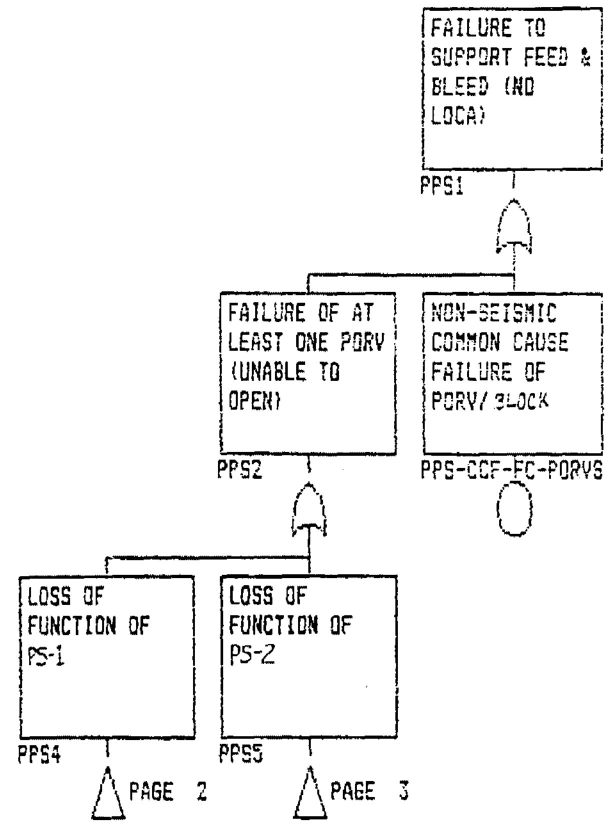


Figure D-2 Power-Operated Relief Valve Fault Tree (No LOCA), Pruned and Merged.

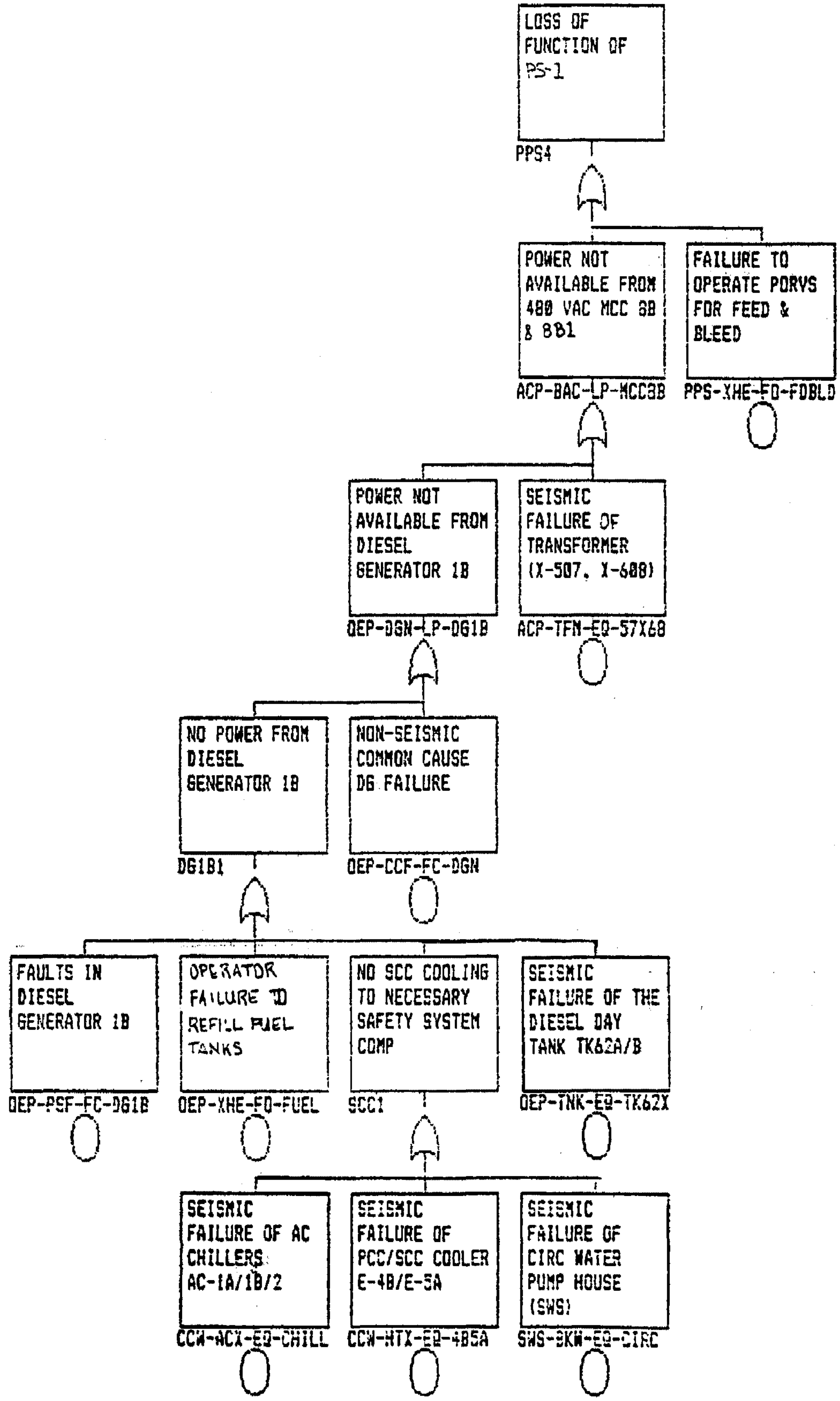


Figure D-2 (cont.)

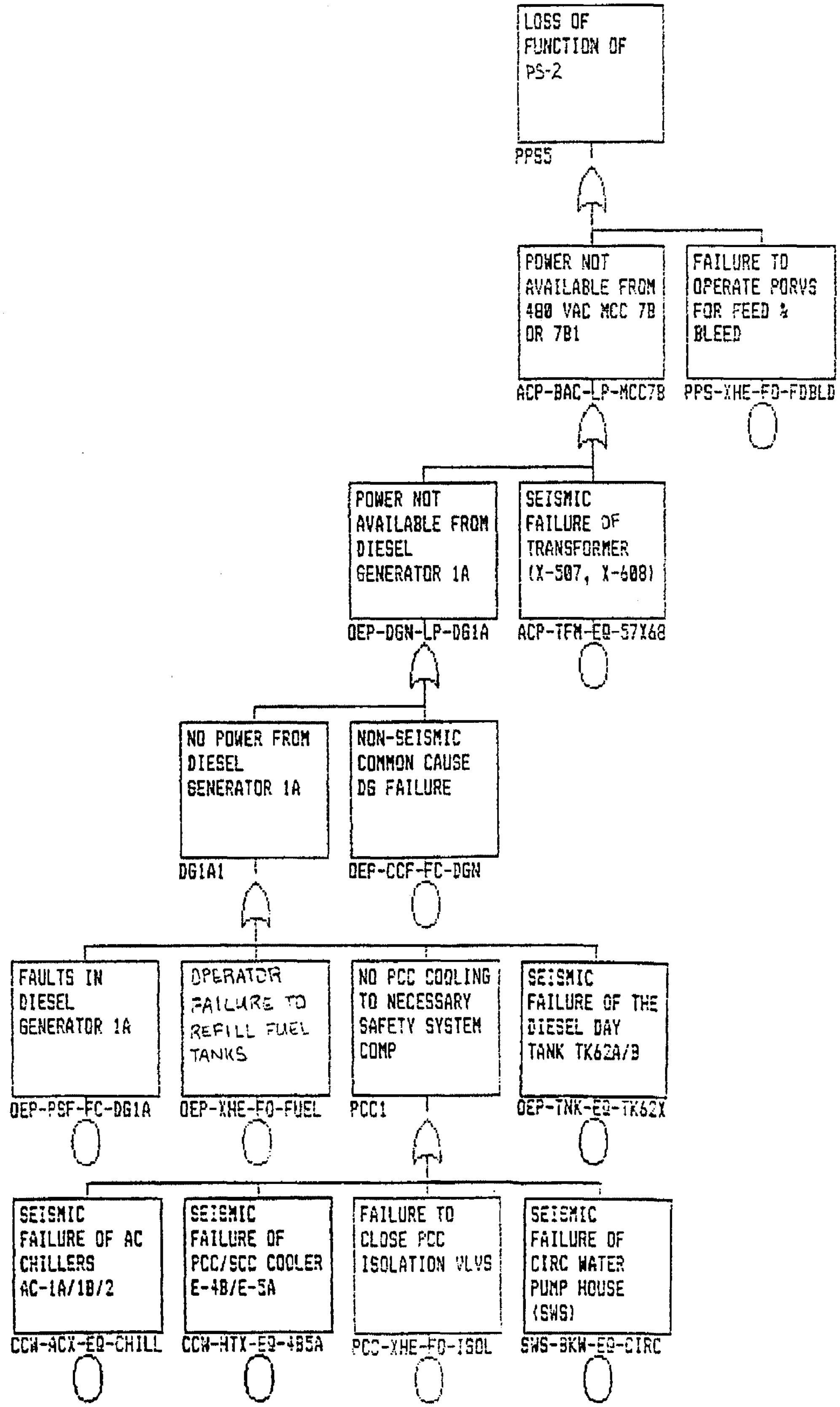


Figure D-2 (cont.)

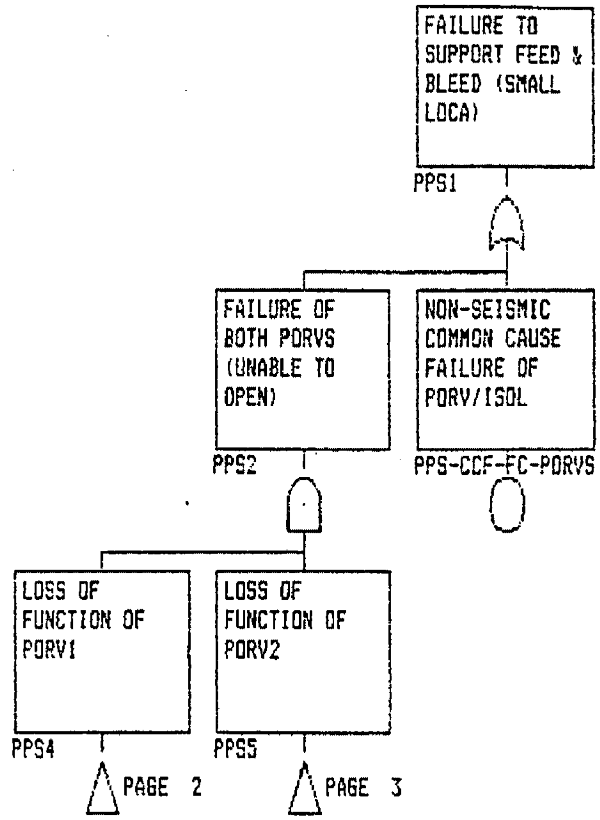


Figure D-3 Power-Operated Relief Valve Fault Tree (Small LOCA), Pruned and Merged.

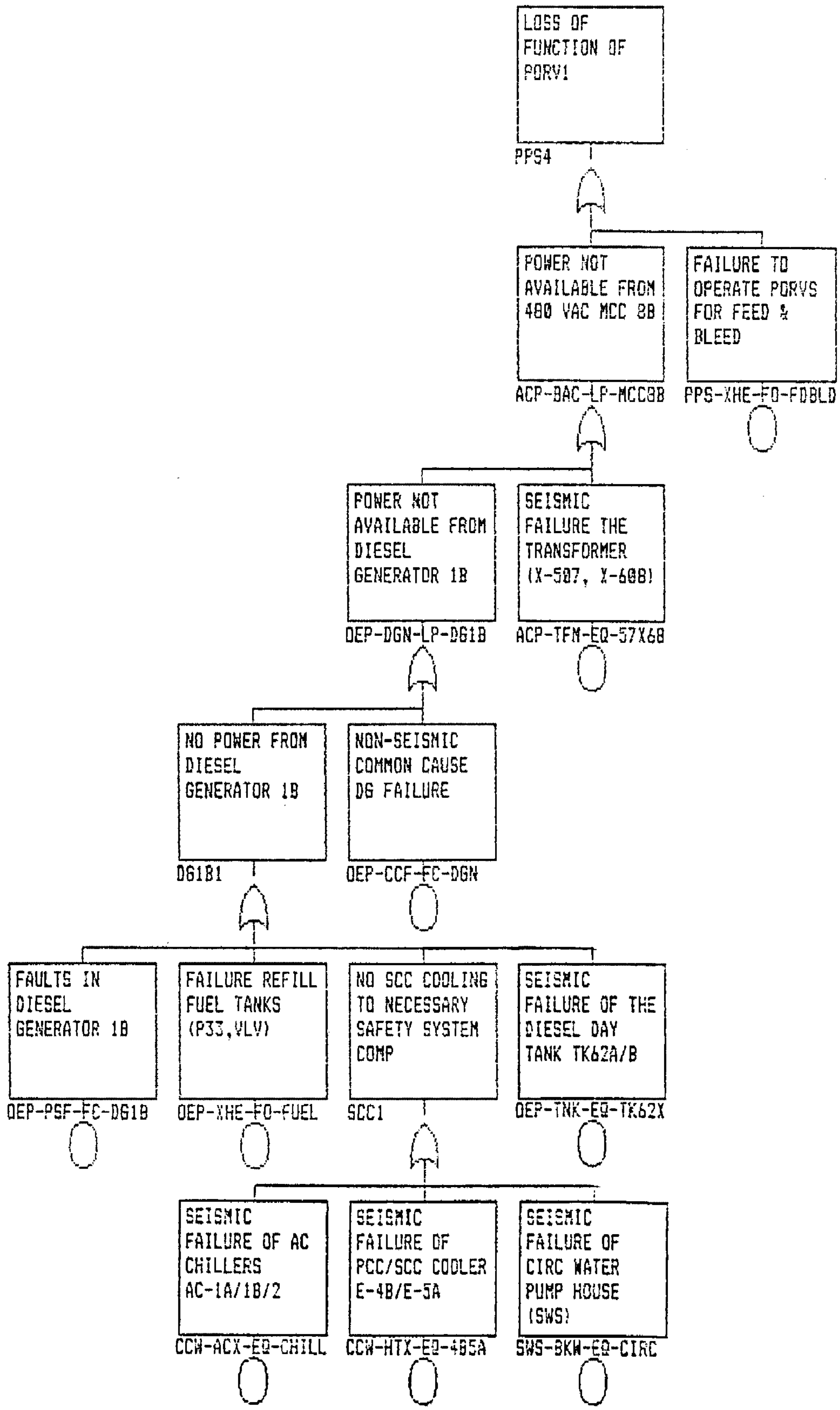


Figure D-3 (cont.)

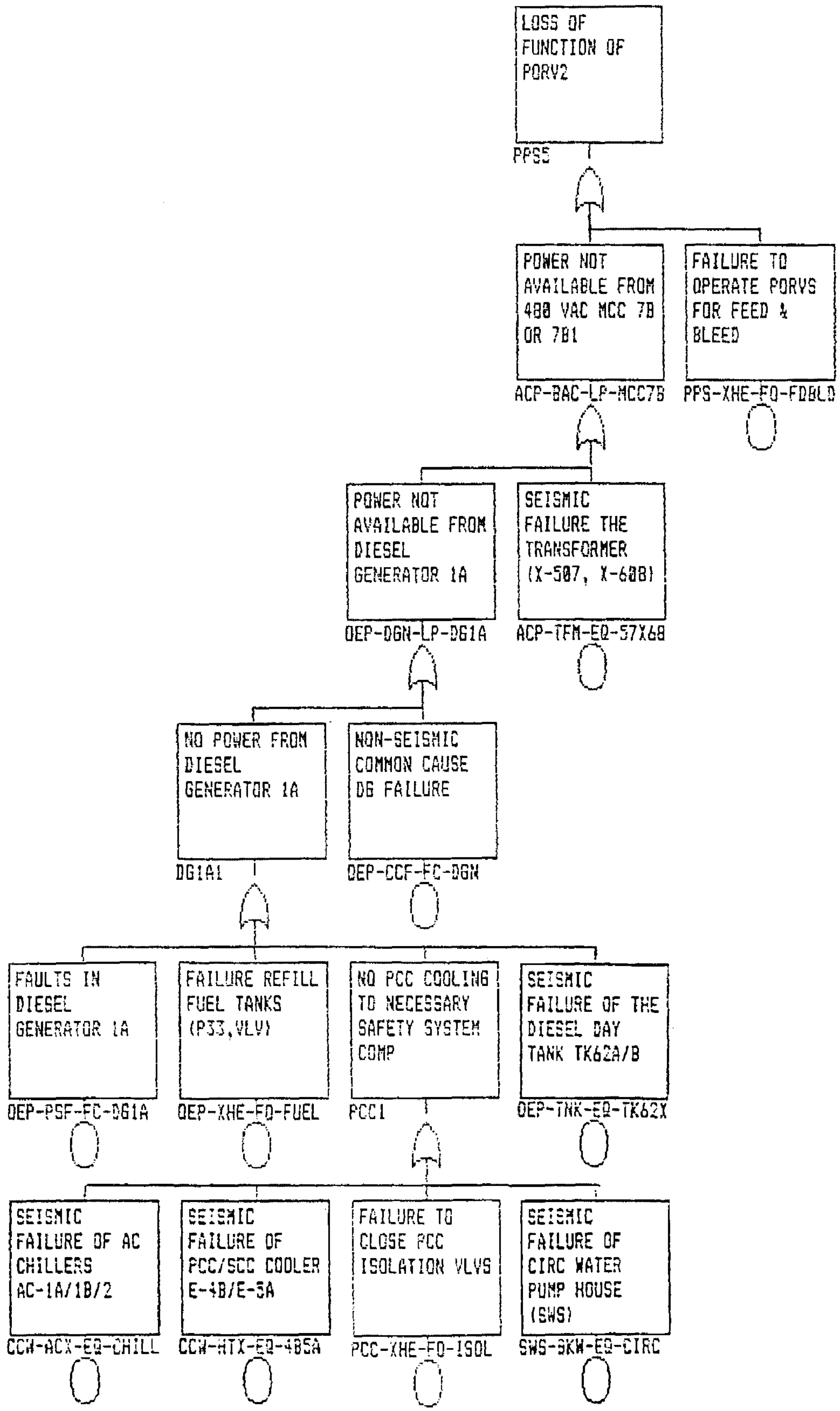


Figure D-3 (cont.)

APPENDIX E

PRIMARY COMPONENT COOLING WATER SYSTEM

Table E-1. Primary component cooling (PCC).

Safety Function:	Provide cooling water required by plant equipment for normal operation and decay heat removal during cooldown or accidents.	
System Components:		
Tanks:	TK-5	PCC Surge Tank
Pumps:	P-9A (N.O.)	PCC Pump
	P-9B (S)	PCC Pump
Heat Exchangers:	E-4A (S.)	PCC Cooler
	E-4B (N.O.)	PCC Cooler
Valves:	Refer to Valve Table	
Support Systems:		
AC Power:	P-9A	4160V Emergency Bus 5
	P-9B	4160V Emergency Bus 5
DC Power:	P-9A	125V DC Distribution Cabinet 1
	P-9B	125V DC Distribution Cabinet 1
HVAC:	Turbine Building	
Cooling:	P-9A	Oil Cooled
	P-9B	Oil Cooled
	Motors	Air Cooled
	E-4A	SWS
	E-4B	SWS
Actuation:	P-9B	Low PCC Header Pressure

Table E-2. PCC valve table.

Valve	Description	Power (SOV)	Normal Position	Operating Position (Actuation)	Fail Position
PCC-A-216	Return from penetration coolers	125 VDC DP/P (3413)	O	C(SIAS-A/CIS-A)	C**
PCC-A-238	Return from air recirc. coolers	125 VDC DP/BU (3412)	O	C(CSAS-B)	C**
PCC-A-268	Return from CEA air coolers	125 VDC BATT 2 (3416)	O	C(SIAS-B/CIS-B)	C**
PCC-A-493	DG-1A cooling water outlet	125VDC (1730A)	O		
PCC-M-43	PCCW outlet from RHR heat exchanger	MCC 7B1	C	O(RAS-A)	AI
PCC-M-90	PCCW isolation to auxiliary building	MCC 7A	O	C(RAS-A)	AI
PCC-M-150	PCCW isolation to letdown heat exchangers	MCC 7A	O	C(RAS-A)	AI
PCC-M-219	PCCW isolation to containment	MCC 7A	O	C(CIS-A)	AI
PCC-T-19	Cooler supply temperature control	pneumatic	O	O	O
PCC-T-20	Cooler bypass temperature control	pneumatic	C	O	C

** Fail open on loss of solenoid power

Table E-3. PCC cooling requirements.

P-9A	• Air-cooled motor • Oil-lubed bearings
P-9B	• Air-cooled motor • Oil-lubed bearings
Turbine Building	• Natural circulation

Table E-4. PCC cooling loads.

	Loads	Location
*AC-1B	Control Room Air Conditioner	Vent & AC Equip. Rm, El. 39'0"
C-3A	Waste Gas Compressor (E-88A)	
C-3B	Waste Gas Compressor (E-88B)	
*E-3A	Residual Heat Removal Exchanger	Cont. Spray Pump Area, El. 14'6"
*E-25	Fuel Pool Heat Exchanger	Fuel Bldg., El. 25'0"
E-29	Recovery Evaporator Distillate Condenser	
E-30	Recovery Evaporator Distillate Cooler	
E-31	Recovery Evaporator Bottoms Cooler	
E-34	Reactor Coolant Pump Seal Water Heat Exchanger	
E-35	High Pressure Drain Cooler	
E-39A	Neutron Shield Tank Cooler	
E-39B	Neutron Shield Tank Cooler	
E-44	Letdown Heat Exchanger	
E-45	Waste Evaporator Bottoms Cooler	
E-46	Waste Evaporator Distillate Cooler	
E-53-1	CEA Drive Mechanism Air Cooler	
E-53-2	CEA Drive Mechanism Air Cooler	
E-53-3	CEA Drive Mechanism Air Cooler	
*E-54-1	Reactor Containment Air Recirculation Cooler	Reactor Containment
*E-54-2	Reactor Containment Air Recirculation Cooler	Reactor Containment
*E-54-3	Reactor Containment Air Recirculation Cooler	Reactor Containment
*E-54-4	Reactor Containment Air Recirculation Cooler	Reactor Containment
*E-54-5	Reactor Containment Air Recirculation Cooler	Reactor Containment
*E-54-6	Reactor Containment Air Recirculation Cooler	Reactor Containment
E-70	Pressurizer Quench Tank Cooler	
E-71A	Degasifier Vent Condenser	
E-71B	Degasifier Vent Condenser	
E-72A	Degasifier Effluent Cooler	
E-72B	Degasifier Effluent Cooler	
E-75	Waste Evaporator Distillate Condenser	
E-77A	Reactor Coolant Sample Heat Exchanger	
E-77B	Reactor Coolant Sample Heat Exchanger	
E-81A	Secondary Sample Heat Exchanger	
E-81B	Secondary Sample Heat Exchanger	
E-81C	Secondary Sample Heat Exchanger	
*E-82A	DG-1A Cooler	
*E-91B	Safeguard (LPSI) Pumps Seal Leakage Cooler	
*E-92B	Charging Pump Seal Leakage Cooler	PAB, El. 13'6"
E-93A	Degasifier Vent Cooler	
E-93B	Degasifier Vent Cooler	
E-94	Waste Gas Compressors Aftercooler	
E-100	Blowdown Tank Cooler	

Table E-4 (Cont'd)

	Loads	Location
P-1-1	Reactor Coolant Pump	
P-1-2	Reactor Coolant Pump	
P-1-3	Reactor Coolant Pump	
*P-7	Auxiliary Charging Pump	PAB, El. 11'0"
P-11	Recovery Evaporator Reboiler Pump	
*P-12A	LPSI Pump	Cont. Spray Pump Area, El. 14'6"
*P-14A	Charging Pump	PAB, El. 21'0"
*P-14S	Charging Pump	PAB, El. 21'0"
P-19	Recovery Evaporator Distillate Pump	
P-20	Recovery Evaporator Bottoms Pump	
P-21	Waste Evaporator Reboiler Pump	
P-22	Waste Evaporator Distillate Pump	
P-65	Waste Evaporator Bottoms Pump	
P-66A	Degasifier Pump	
P-66B	Degasifier Pump	
*Containment Penetration Coolers	(Penetrations 9, 29, 30, 31, 32, 45, 46, 53, 54, 55, 62, 64, 65, 66)	

* = PCC Loads not isolated by PCC-M-90, -150 or -219, whose failure may lead to failure of the PCC.

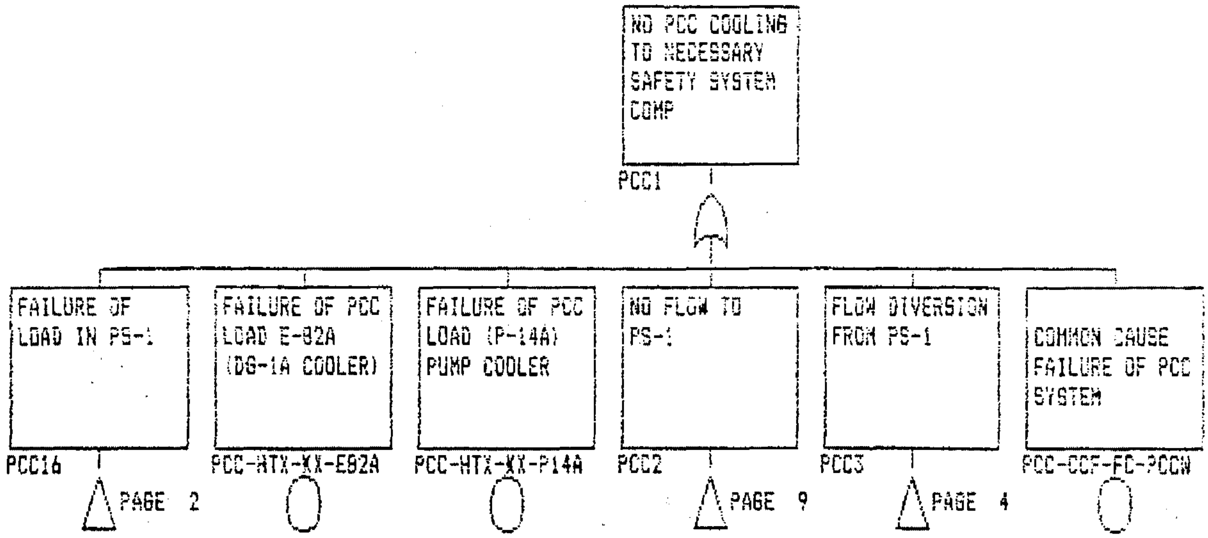


Figure E-1 Primary Component Cooling System Fault Tree.

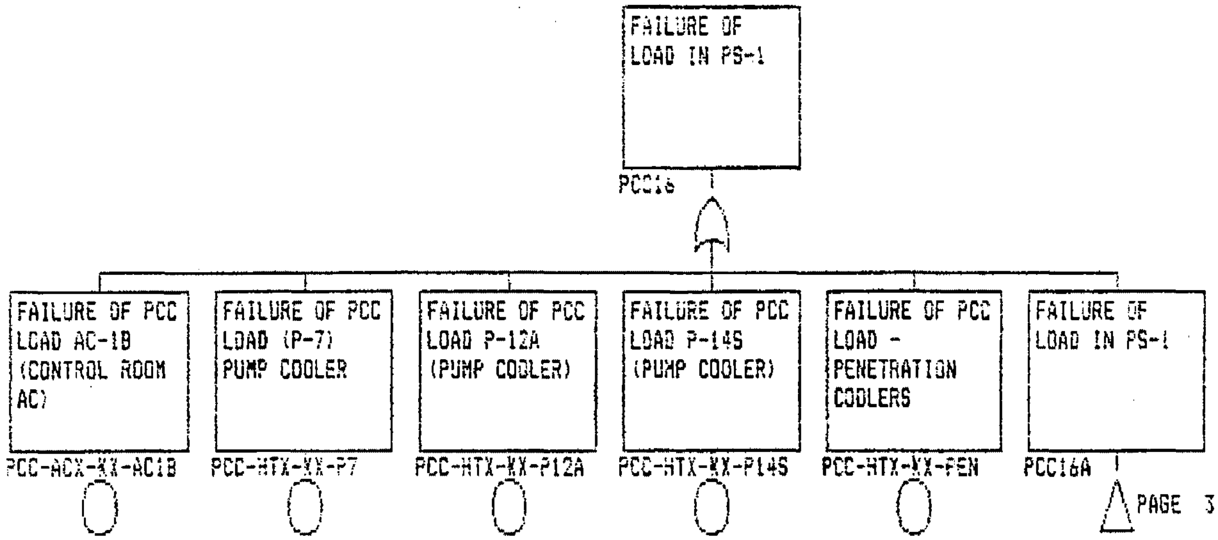


Figure E-1 (cont.)

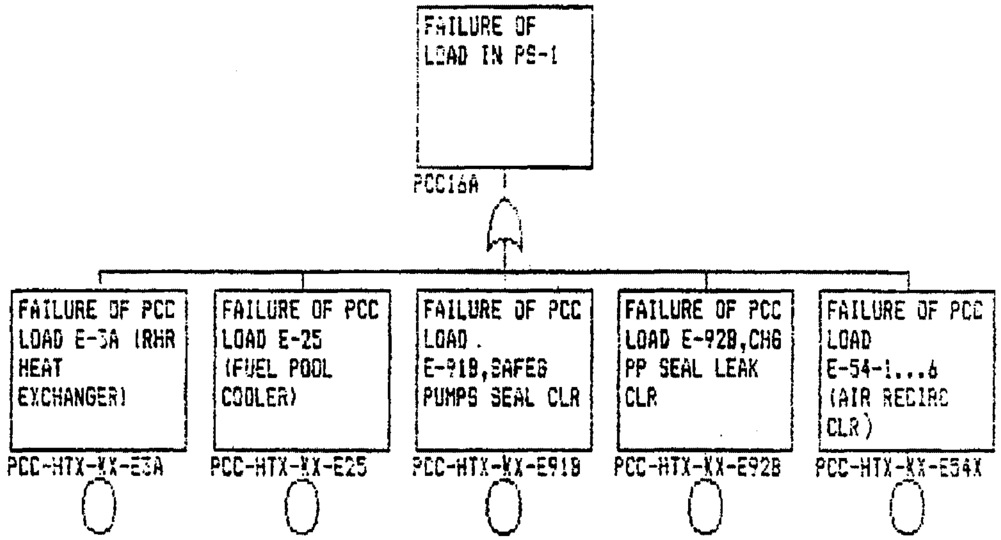


Figure E-1 (cont.)

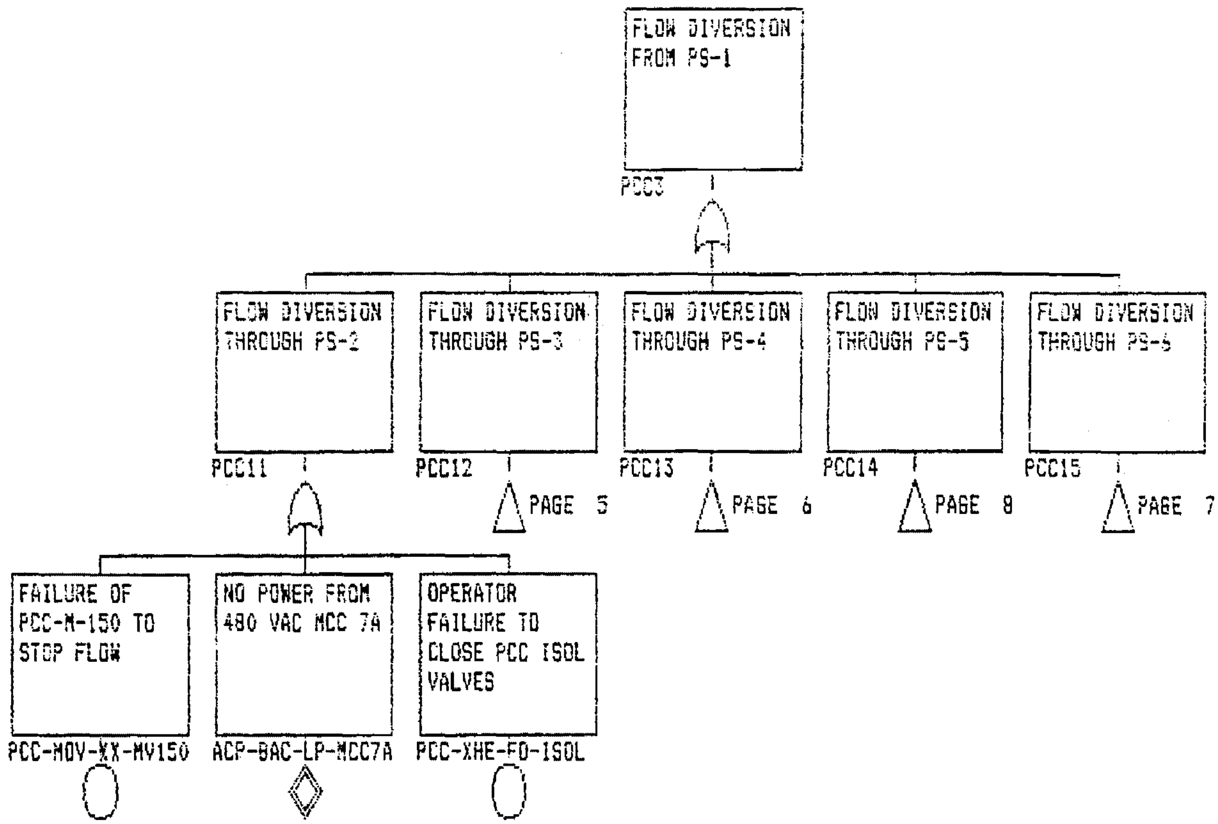


Figure E-1 (cont.)

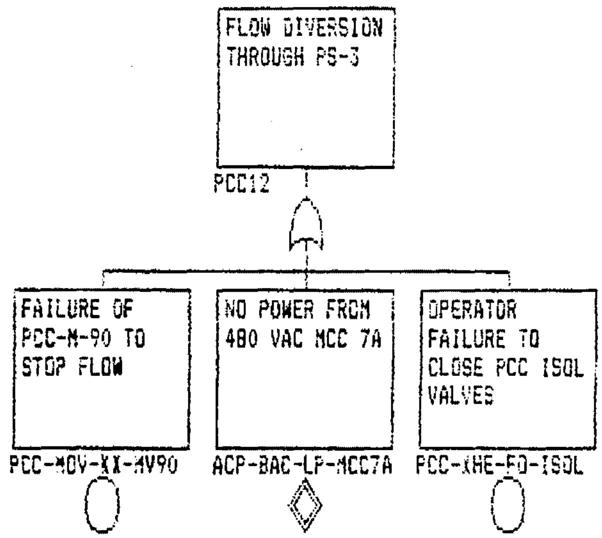


Figure E-1 (cont.)

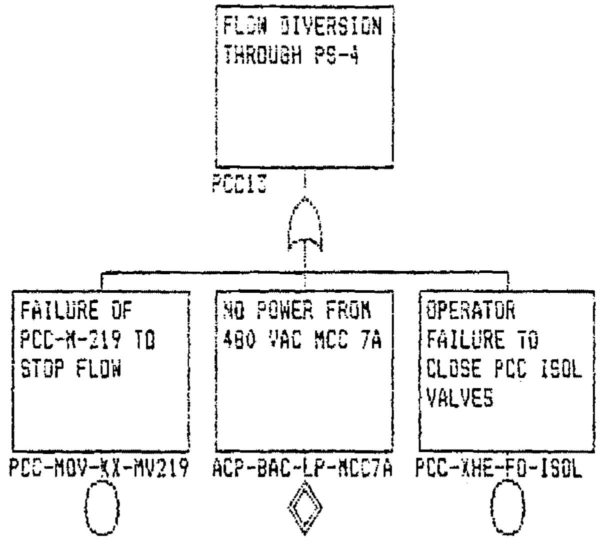


Figure E-1 (cont.)

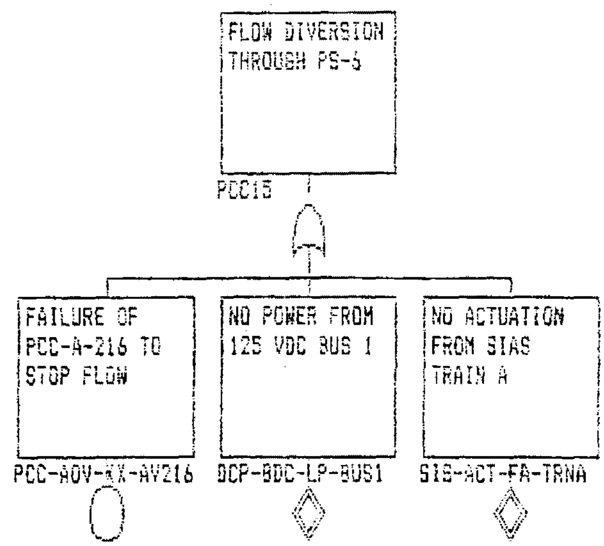


Figure E-1 (cont.)

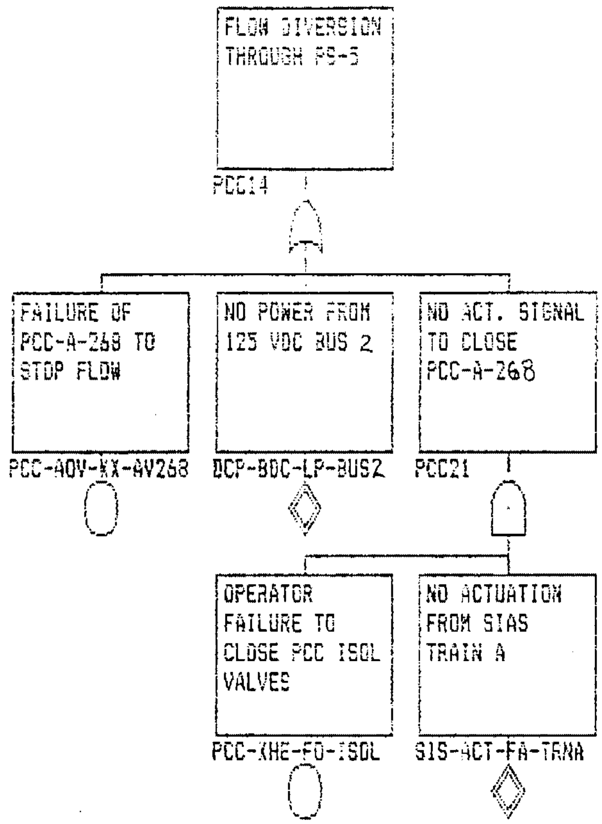


Figure E-1 (cont.)

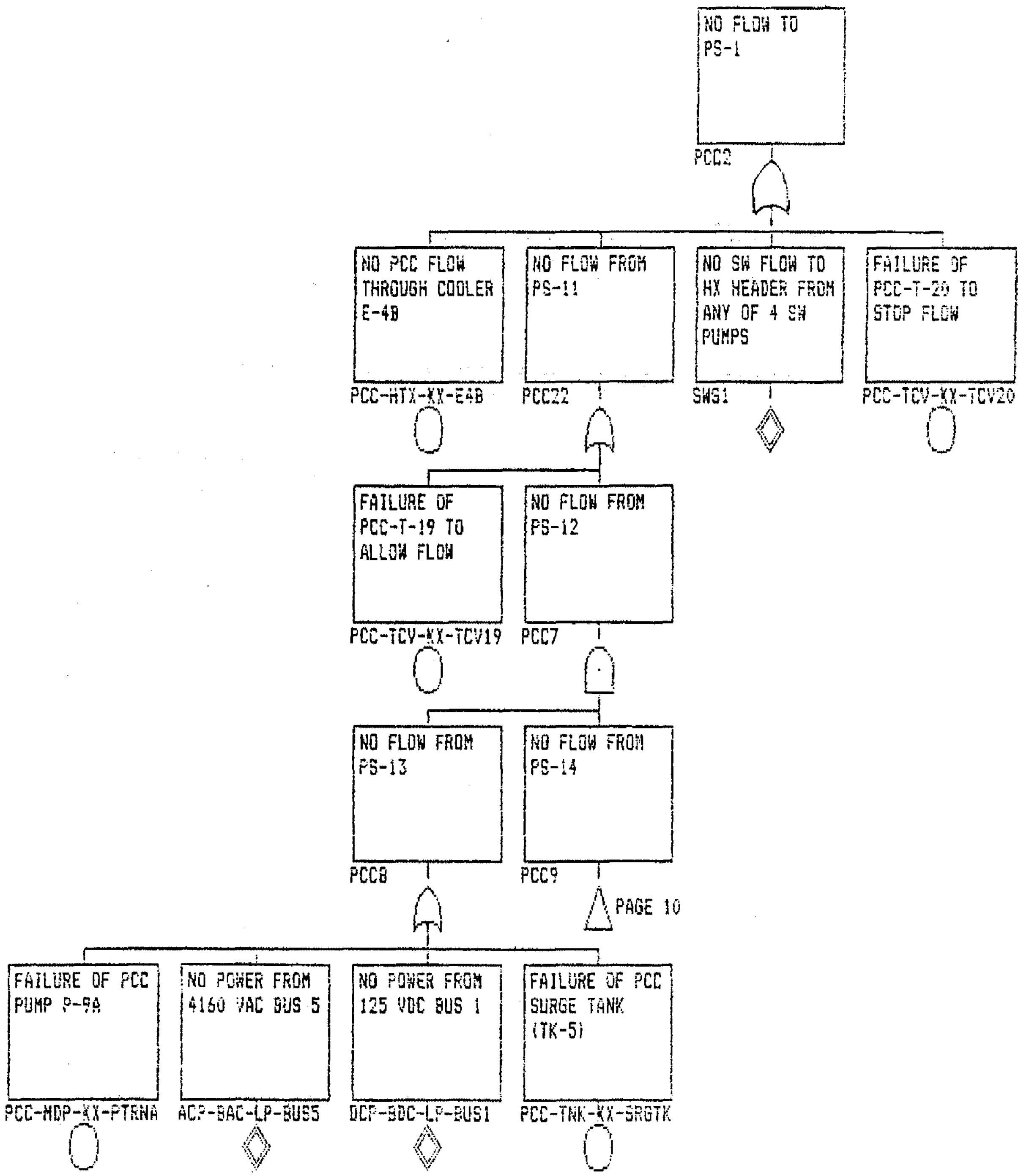


Figure E-1 (cont.)

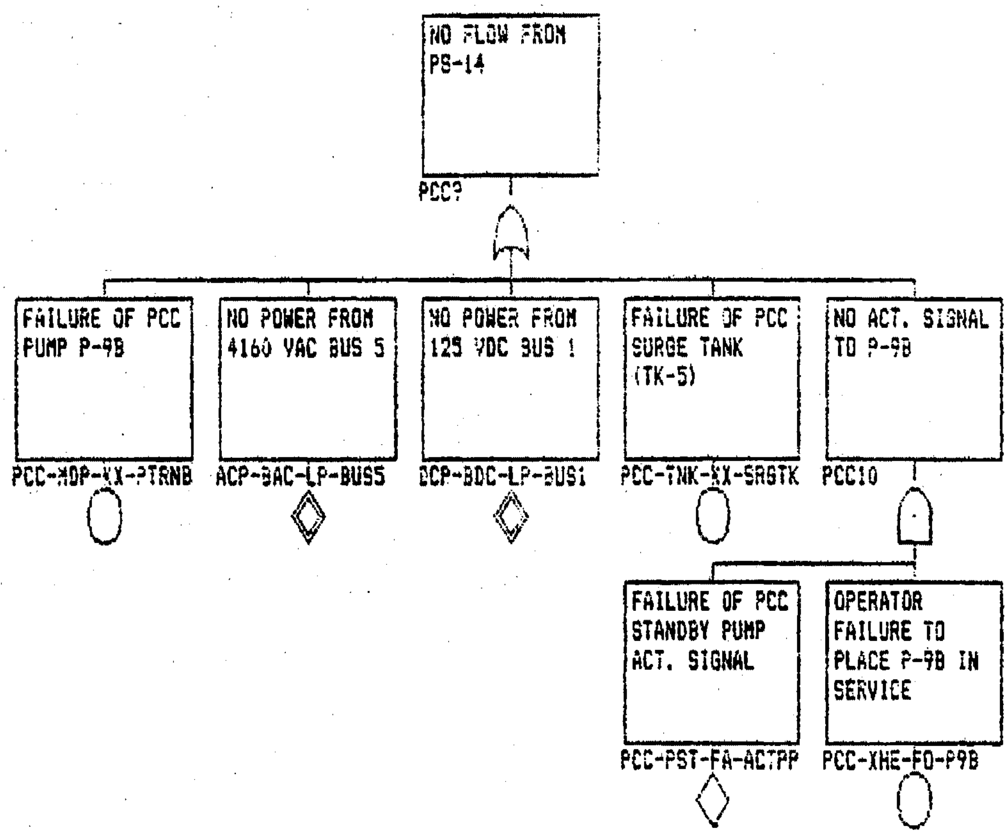


Figure E-1 (cont.)

APPENDIX F

SECONDARY COMPONENT COOLING WATER SYSTEM

Table F-1. Secondary component cooling (SCC).

Safety Function:	Provide cooling water required by plant equipment for normal operation and decay heat removal during cooldown or accidents.	
System Components:		
Tanks:	TK-59	SCC Surge Tank
Pumps:	P-10A (N.O.)	SCC Pump
	P-10B (S)	SCC Pump
Heat Exchangers:	E-5A (N.O.)	SCC Cooler
	E-5B (S)	SCC Cooler
Valves:	Refer to Valve Table	
Support Systems:		
AC Power:	P-10A	4160V Emergency Bus 6
	P-10B	4160V Emergency Bus 6
DC Power:	P-10A	125V DC Distribution Cabinet 3
	P-10B	125V DC Distribution Cabinet 3
Air:	TK-110	Isolation valves (SCC-A-460, 461)
HVAC:	Turbine Building	
Cooling:	P-10A	Oil Cooled
	P-10B	Oil Cooled
	Motors	Air-Cooled
	E-5A	SWS
	E-5B	SWS
Actuation:	P-10B	Low SCC Header Pressure
	SCC-A-460, 461	Low Suction Pressure

Table F-2. SCC valve table.

Valve	Description	Power (SOV)	Normal Position	Operating Position (Actuation)	Fail Position
SCC-A-460	Non-seismic supply header stop	125 VDC 3 DP/BU (1725A1 & A2)	0	C	0
SCC-A-461	Non-seismic supply header stop	125 VDC 3 DP/BU (1725B1 & B2)	0	C	0
SCC-T-23	Cooler bypass temperature control	pneumatic	C	0	C
SCC-T-24	Cooler supply temperature control	pneumatic	0	0	0
SCC-T-305	DG-1B cooler inlet temperature control	125 VDC (1730B)	0		

Table F-3. SCC cooling requirements.

P-10A	• Air-cooled motor • Oil-lubed bearings
P-10B	• Air-cooled motor • Oil-lubed bearings
Turbine Building	• Natural circulation

Table F-4. SCC cooling loads.

	Loads	Location
*AC-1A	Computer Room Air Conditioner	Vent & AC Equip. Rm, El.39'0"
*AC-2	Lab Air Conditioner	Vent & AC Equip. Rm, El.39'0"
AC-3	Office Area Air Conditioner	
C-1A	Control Air Compressor	
C-1B	Control Air Compressor	
C-1C	Control Air Compressor	
*E-3B	Residual Heat Exchanger	Cont. Spray Pump Area, El. 14'6"
E-6A	Generator Hydrogen Cooler	
E-6B	Generator Hydrogen Cooler	
E-6C	Generator Hydrogen Cooler	
E-6D	Generator Hydrogen Cooler	
E-7A	Turbine Oil Cooler	
E-7B	Turbine Oil Cooler	
E-8	Exciter Air Cooler	
E-19A	Generator Seal Oil Unit (Air Side)	
E-19B	Generator Seal Oil Unit (Hydrogen Side)	
E-20	Generator Leads Cooler	
E-21A	Control Air Compressor Aftercooler	
E-21B	Control Air Compressor Aftercooler	
E-21C	Control Air Compressor Aftercooler	
E-78A	Sample Cooler	
E-78B	Sample Cooler	
E-78C	Sample Cooler	
E-78D	Sample Cooler	
E-78E	Sample Cooler	
E-78F	Sample Cooler	
E-79A	Steam Generator Feed Pump Lube Oil Cooler	
E-79B	Steam Generator Feed Pump Lube Oil Cooler	
E-80A	Electro Hydraulic Governor Oil Cooler	
E-80B	Electro Hydraulic Governor Oil Cooler	
*E-82B	DG-1B Cooler	
*E-91A	Safeguards (LPSI) Pumps Seal Leakage Cooler	
*E-92A	Charging Pump Seal Leakage Cooler	PAB, El. 16'0"
E-101A	Turbine Drive Main Feed Pump	
E-101B	Lube Oil Cooler	
*P-12B	LPSI Pump	Cont. Spray Pump Area, El. 14'6"
*P-14B	Charging Pump	PAB, El. 21'0"
+P-14S	Charging Pump	
P-27A	Condensate Pump	

Table F-4 (Cont'd)

	Loads		Locations
P-27B	Condensate Pump		
P-27C	Condensate Pump		
P-62A	Heater Drain Pump		
P-62B	Heater Drain Pump		
*Containment	Penetrations Cooler		
	Penetrations	9,29,30,31	Containment Spray Pump Area
	Penetrations	32,45,46,47,62	Primary Aux. Tunnel Area
	Penetrations	53,54,55,64,65,66	Main Steam Valve Area

* Loads not isolated by non-seismic stop valves, whose failure may fail the SCC system.
+ PCC cooling perferred.

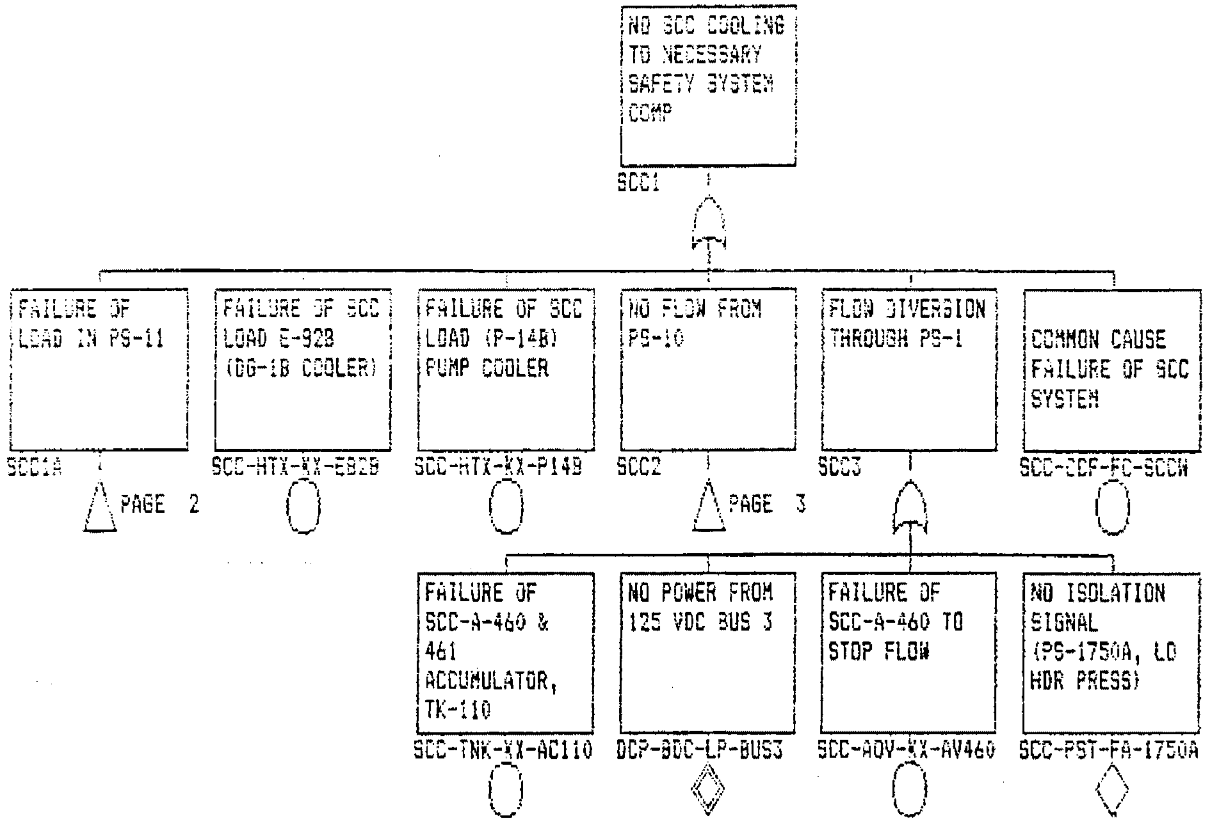


Figure F-1 Secondary Component Cooling System Fault Tree.

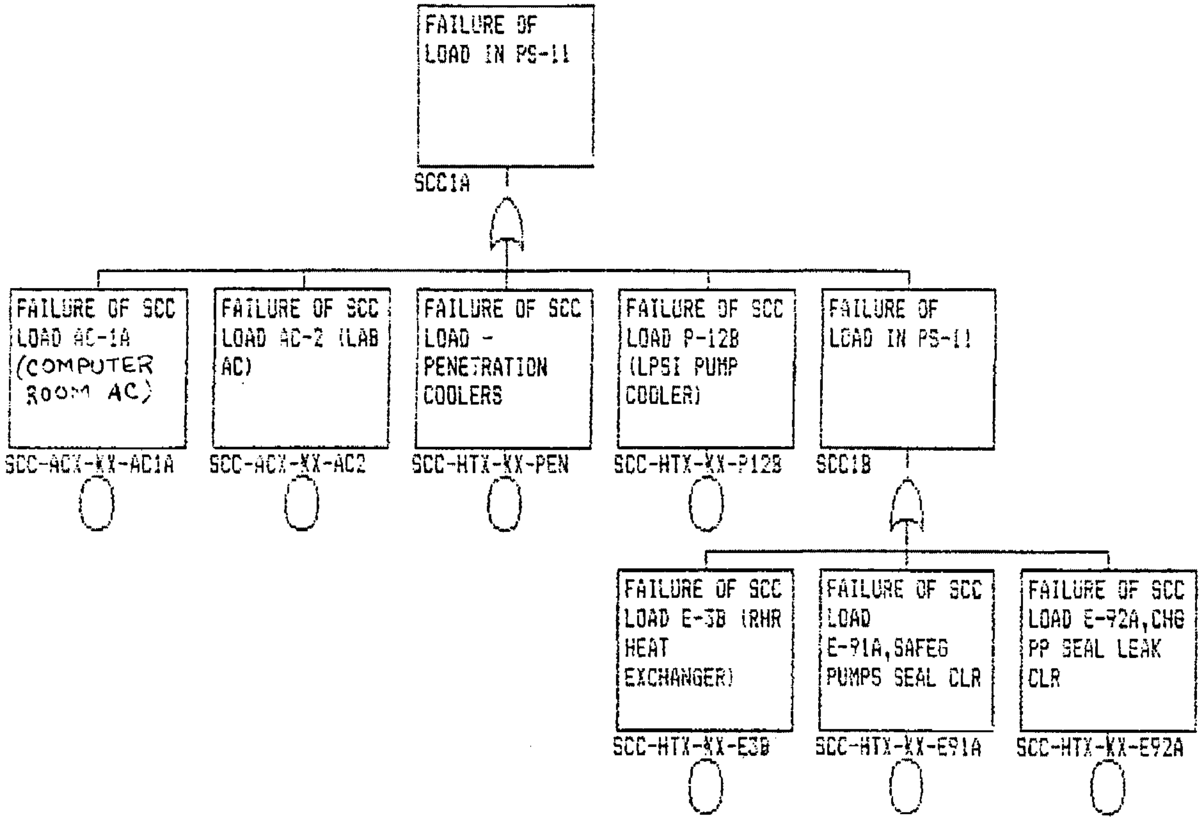


Figure F-1 (cont.)

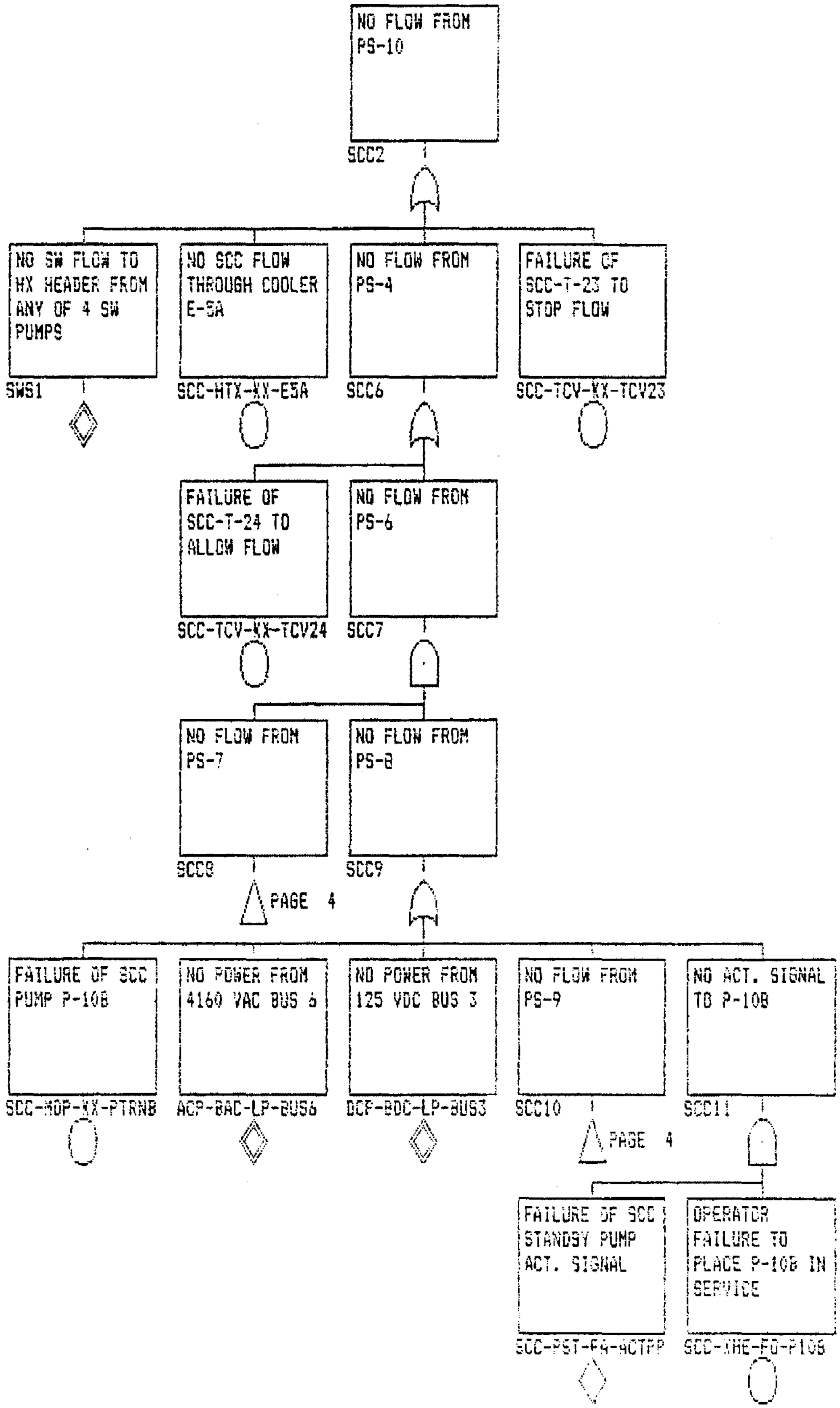


Figure F-1 (cont.)

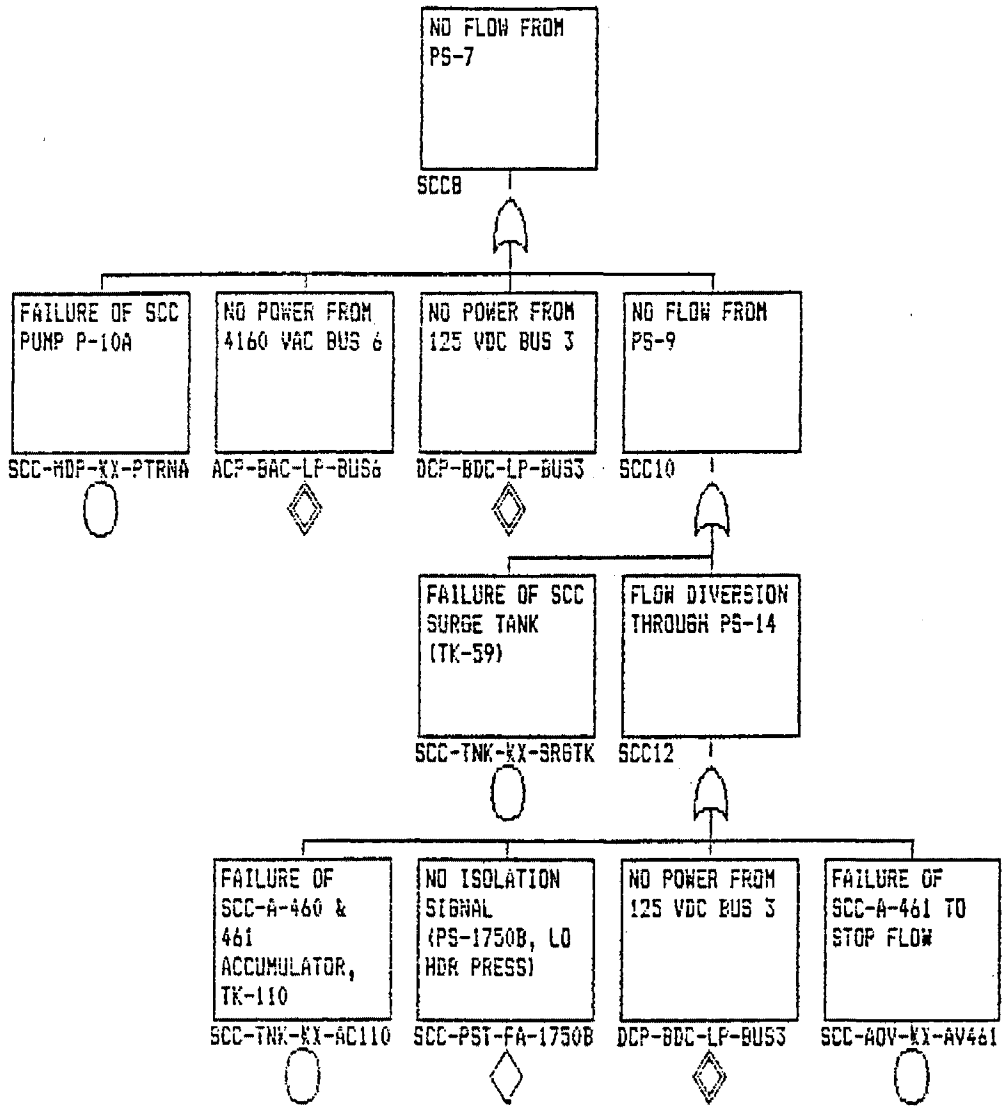


Figure F-1 (cont.)

APPENDIX G

SERVICE WATER SYSTEM

Table G-1. Service water system (SWS).

Safety Function:	Provide cooling for the PCC and SCC systems.	
System Components:		
Pumps:	P-29A (N.O.)	South SWS Pump (for SCC)
	P-29B (S)	South SWS Pump (for SCC)
	P-29C (S)	North SWS Pump (for PCC)
	P-29D (N.O.)	North SWS Pump (for PCC)
Heat Exchangers:	E-4A(S)	PCC Cooler
	E-4B(N.O.)	PCC Cooler
	E-5A(N.O.)	SCC Cooler
	E-5B(S)	SCC Cooler
Support Systems:		
AC Power:	P-29A	480V Emergency Bus 7
	P-29B	480V Emergency Bus 8
	P-29C	480V Emergency Bus 7
	P-29D	480V Emergency Bus 8
DC Power:	P-29A	125V DC Distribution Cabinet 1,
	P-29B	125V DC Distribution Cabinet 3,
	P-29C	125V DC Distribution Cabinet 1,
	P-29D	125V DC Distribution Cabinet 3,
HVAC:	Pump House	
Pump Cooling:	Pump Discharge	

Table G-2. SWS cooling requirements.

P-29A	• Pump discharge recirculation (safety class) • Raw water (preferred)
P-29B	• Pump discharge recirculation (safety class) • Raw water (preferred)
P-29C	• Pump discharge recirculation (safety class) • Raw water (preferred)
P-29D	• Pump discharge recirculation (safety class) • Raw water (preferred)
Circ. Water Pump House	• Natural circulation

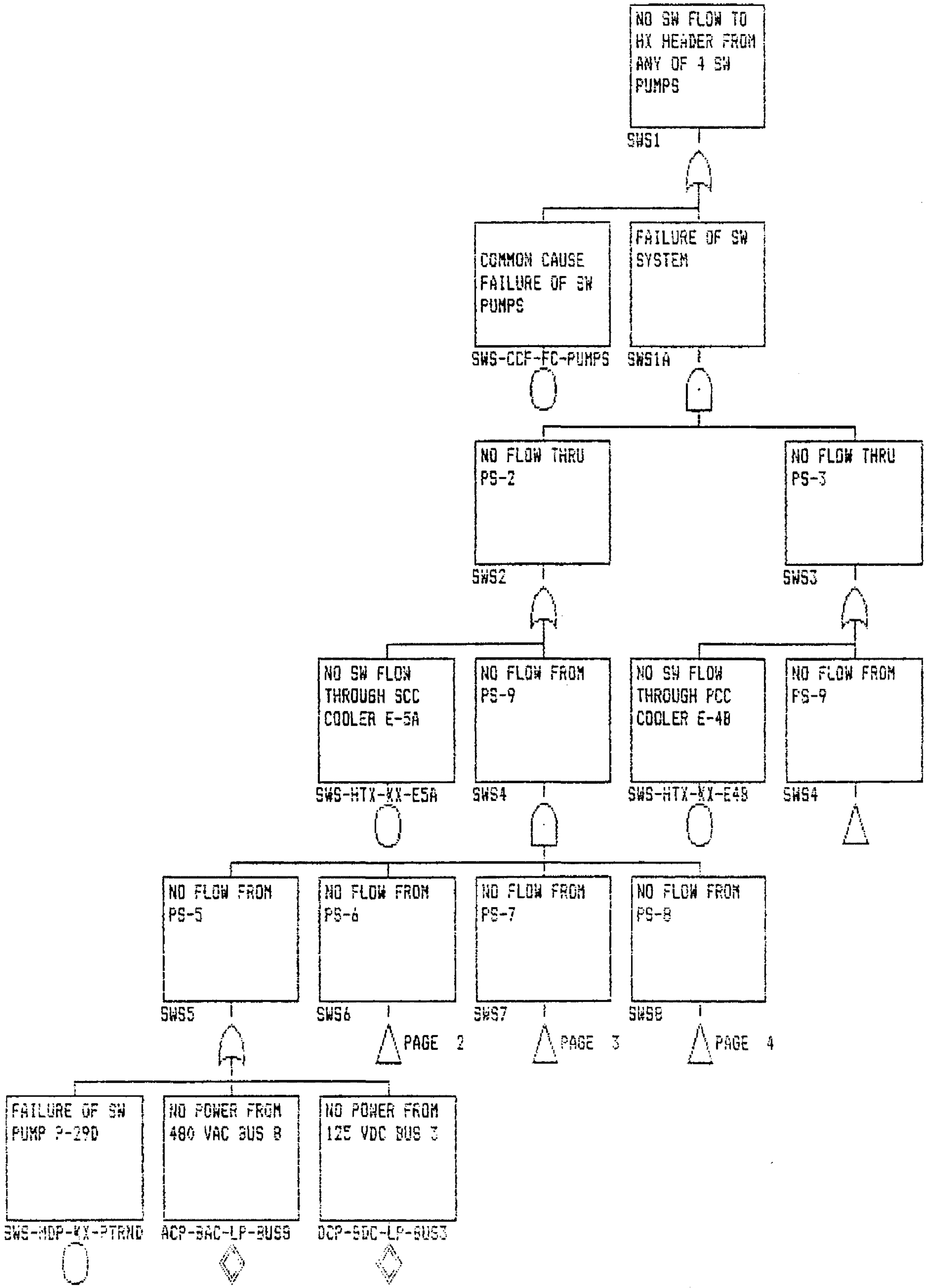


Figure G-1 Service Water System Fault Tree.

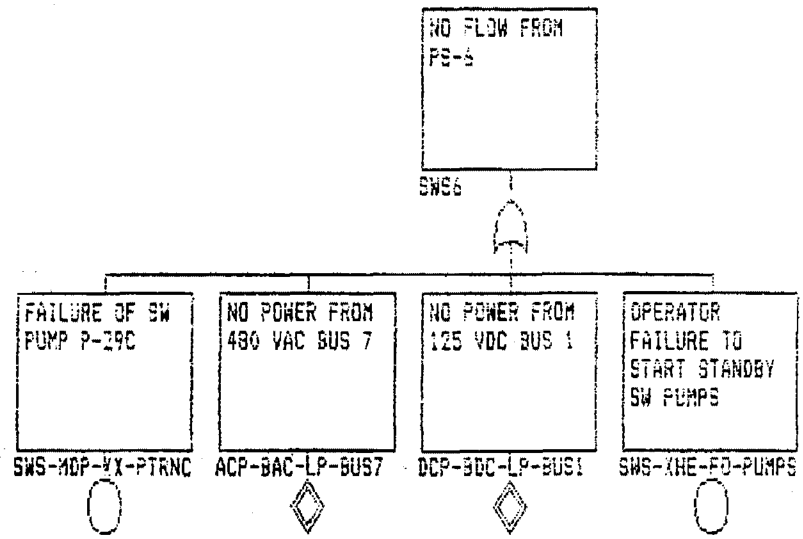


Figure G-1 (cont.)

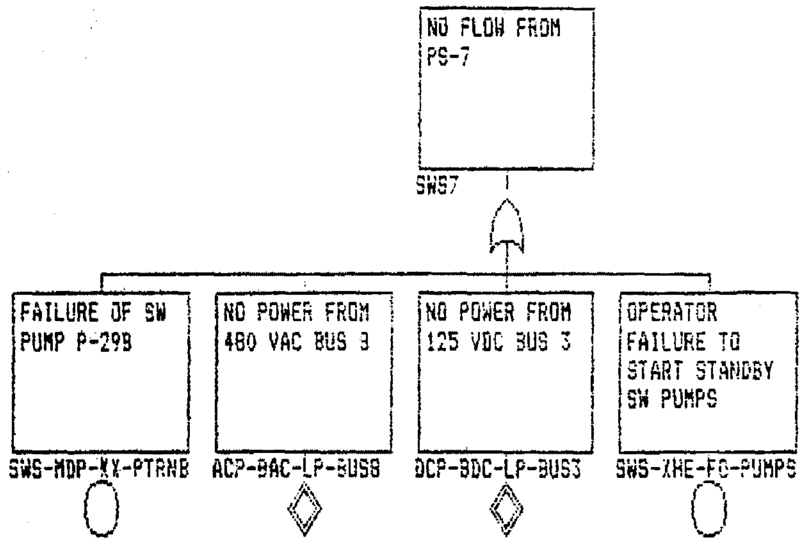


Figure G-1 (cont.)

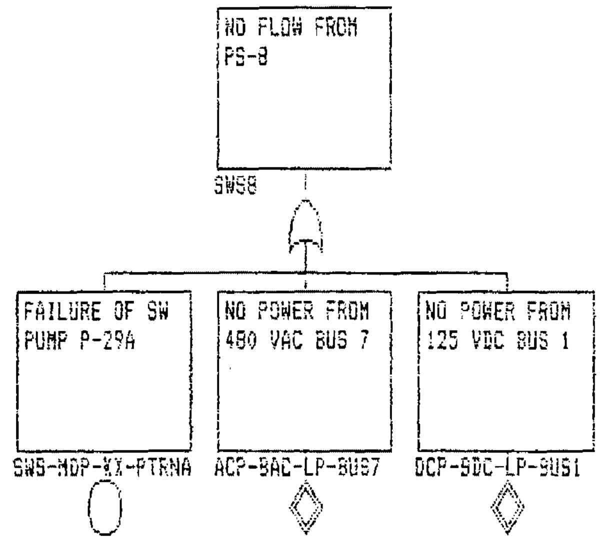


Figure G-1 (cont.)

APPENDIX H

ELECTRIC POWER SYSTEM

Table H-1. AC power.

Safety Function:	To provide operating power to plant equipment and 120V AC power to plant instrumentation.	
System Components:		
Buses:	Buses 5, 6	4160V Emergency Buses
	Buses 7, 8	480V Emergency Buses
	Buses 1-4, 1A-4A	120V Vital Buses
Transformers:	X-507	4160V Bus 5 to 480V Bus 7
	X-608	4160V Bus 6 to 480V Bus 8
Motor Control Centers:	MCC 7A, 7B and 7B1	480V Emergency MCCs
	MCC 8A, 8B and 8B1	480V Emergency MCCs
Inverters:	INVR-1	120VAC Inverter
	INVR-2	120VAC Inverter
	INVR-3	120VAC Inverter
	INVR-4	120VAC Inverter
Support Systems:		
AC Power:	DG-1A	Diesel Generator
	DG-1B	Diesel Generator
DC Power:	DC-1	125 VDC Bus
	DC-2	125 VDC Bus
	DC-3	125 VDC Bus
	DC-4	125 VDC Bus
HVAC:	FN-31	Switchgear Room Supply Fan
	FN-32	Switchgear Room Exhaust Fan

Table H-2. DC power.

Safety Function:	Provide DC power and control for switchgear, vital bus inverters, vital SOVs and instrumentation.	
System Components:		
Buses:	DC-1	125 VDC Bus
	DC-2	125 VDC Bus
	DC-3	125 VDC Bus
	DC-4	125 VDC Bus
Batteries:	BATT-1	60 Cell Station Battery
	BATT-2	60 Cell Station Battery
	BATT-3	60 Cell Station Battery
	BATT-4	60 Cell Station Battery
Battery Chargers:		
	BC-1	129V - 250A Charger
	BC-2	129V - 250A Charger
	BC-3	120V - 250A Charger
	BC-4	120V - 250A Charger
Distribution:		
	DC/CE-1	Distribution Cabinet
	DC/CE-2	Distribution Cabinet
	DP/P	Distribution Panel
	DP/BU	Distribution Panel
Support Systems:		
AC Power:	BC-1, 2	480V Emergency MCC 7
	BC-3, 4	480V Emergency MCC 8
	FN-31	480V Emergency MCC 7A
	FN-32	480V Emergency MCC 8A
HVAC:	FN-31	Protected Switchgear Room Supply Fan
	FN-32	Protected Switchgear Room Exhaust Fan

Table H-3. Diesel generators (DG).

Safety Function:	Provide electric power to the plant emergency buses when normal power is not available.	
System Components:		
Generators:	DG-1A(S)	Diesel Generator
	DG-1B(S)	Diesel Generator
Pumps:	P-33A	Auxiliary Fuel Oil Transfer Pump
	P-33B	Auxiliary Fuel Oil Transfer Pump
Heat Exchangers:	E-82A	DG-1A Cooler
	E-82B	DG-1B Cooler
Tanks:	TK-28A	Auxiliary Fuel Oil Supply Tank
	TK-28B	Auxiliary Fuel Oil Supply Tank
	TK-62A	Diesel Generator Day Tank
	TK-62B	Diesel Generator Day Tank
Support Systems:		
AC Power:	DG-1A Distribution Panel	480 VAC MCC 7A
	DG-1A Engine Control Panel	480 VAC MCC 7A
	DG-1B Distribution Panel	480 VAC MCC 8A
	DG-1B Engine Control Panel	480 VAC MCC 8A
	P-33A	480 VAC MCC 7A
	P-33B	480 VAC MCC 8A
DC Power:	DG-1A Start 1 & 2 Circuits and Control Power	125 VDC Battery 1
	DG-1B Start 1 & 2 Circuits and Control Power	125 VDC Battery 3
Air:	TK-76-A1,A2,A3	DG-1A Air Receiver Bank
	TK-76-A4,A5,A6	DG-1A Air Receiver Bank
	TK-76-B1,B2,B3	DG-1B Air Receiver Bank
	TK-76-B4,B5,B6	DG-1B Air Receiver Bank
HVAC:	FN-20A	DG-1A Room Exhaust Fan
	FN-20B	DG-1B Room Exhaust Fan
		Air Intake and Exhaust Dampers
DG Cooling:	E-82A	PCC
	E-82B	SCC
Actuation:	DG Start and Load Shed/Sequencer	

Table H-4. DG cooling requirements.

DG-1A	• The engine and turbocharger aftercoolers are water cooled by the PCC (or Fire Protection System), via E-82A and two cooling water pumps. • The generator is air cooled by a blower driven by a gear off the engine camshaft.
DG-1B	• The engine and turbocharger aftercoolers are water cooled by the SCC (or Fire Protection System), via E-82B and two cooling water pumps. • The generator is air cooled by a blower driven by a gear off the engine camshaft.
Diesel Rooms	• Combustion air via normally closed intake and exhaust dampers. • Exhaust FN-20A and FN-20B.

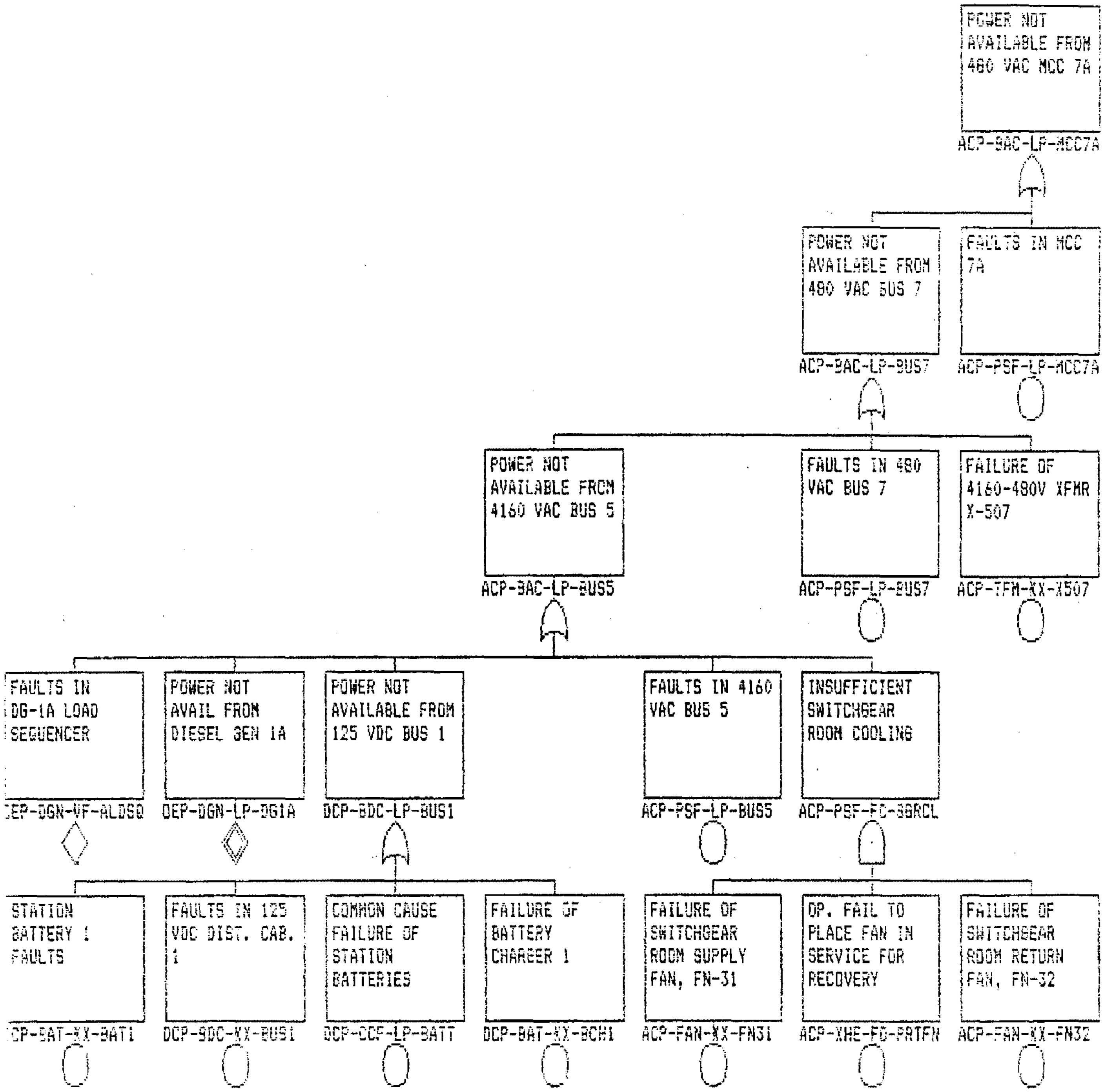


Figure H-1 AC Power (MCC 7A) Fault Tree.

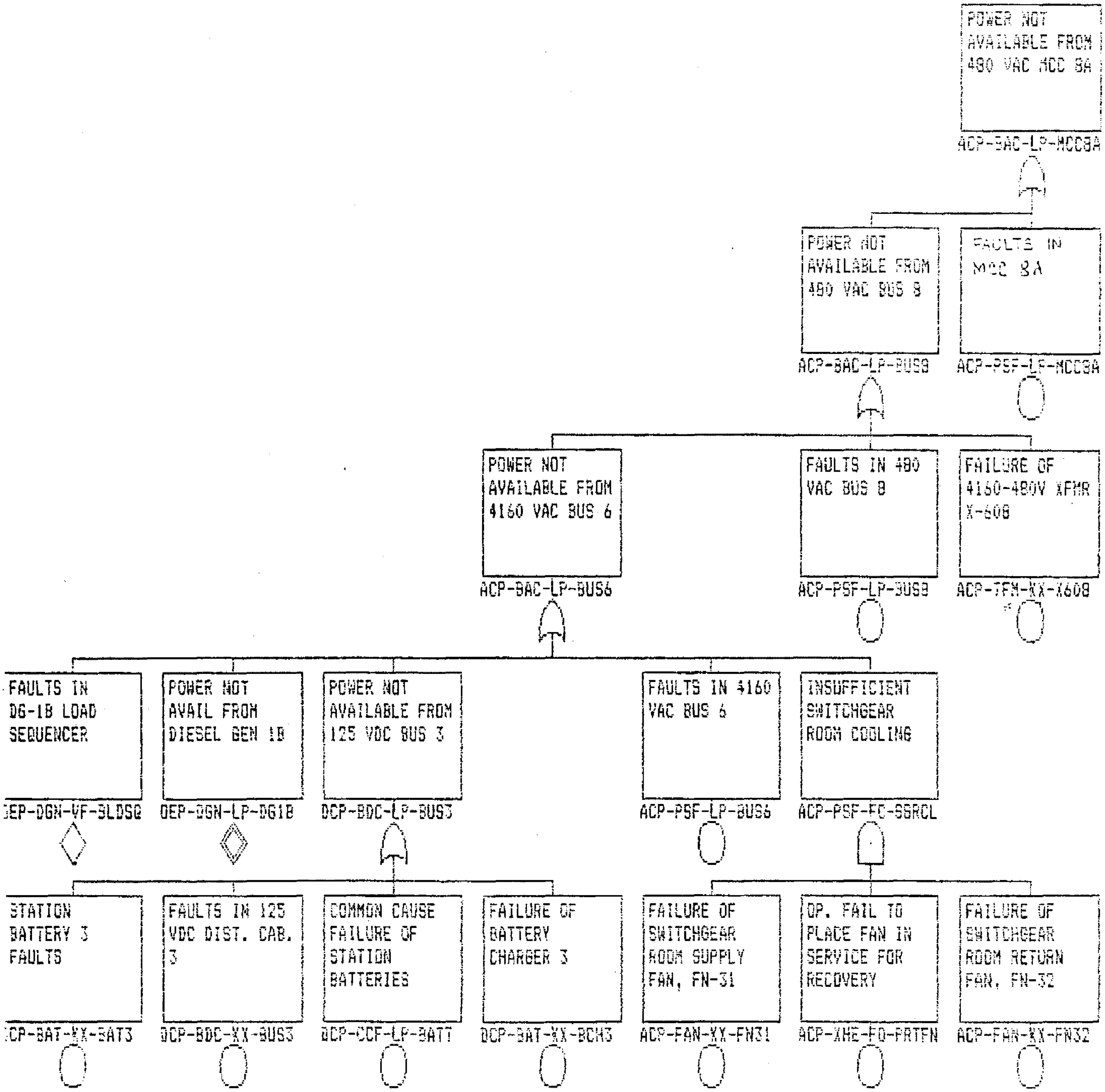


Figure H-2 AC Power (MCC 8A) Fault Tree.

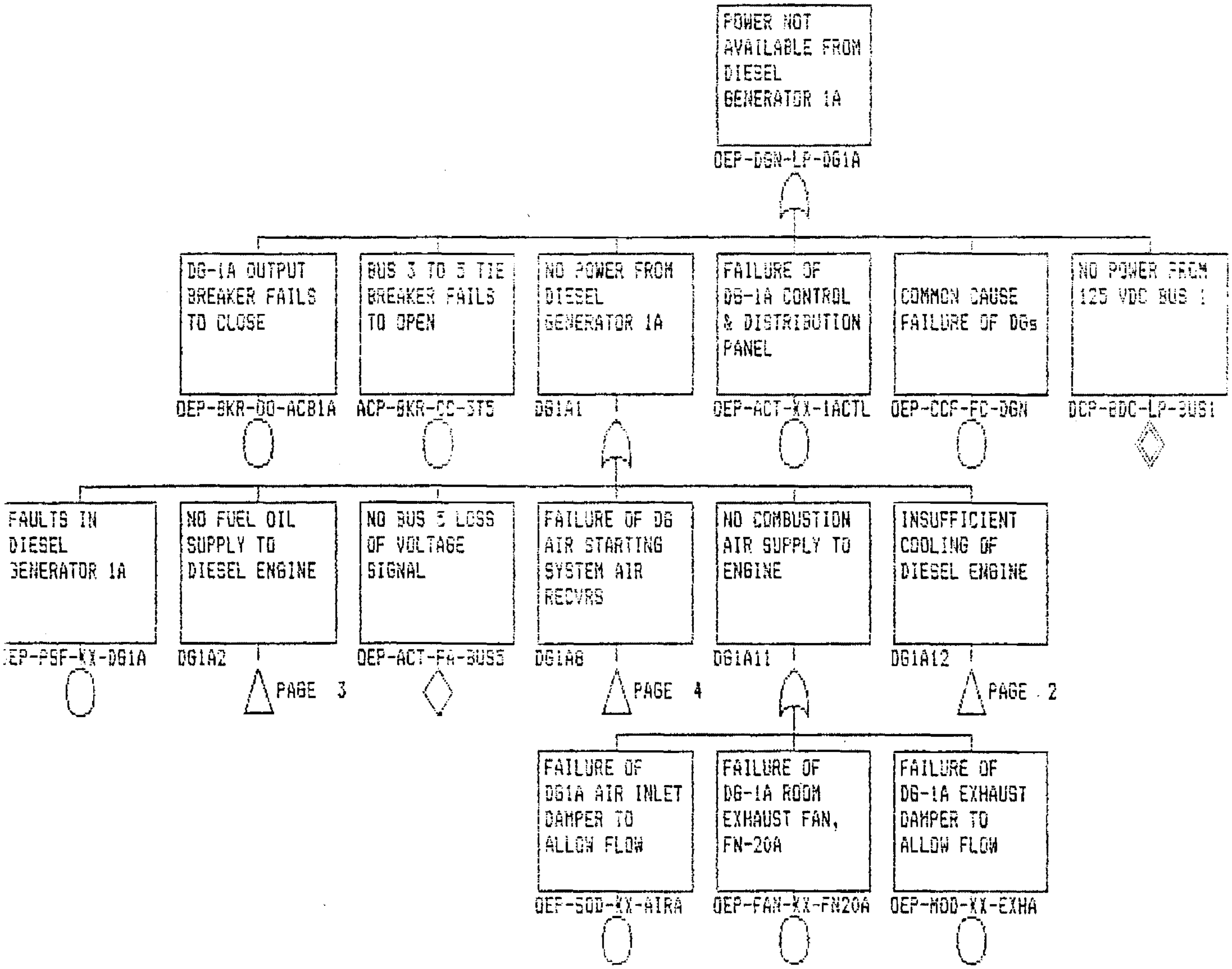


Figure H-3 Diesel Generator 1A Fault Tree.

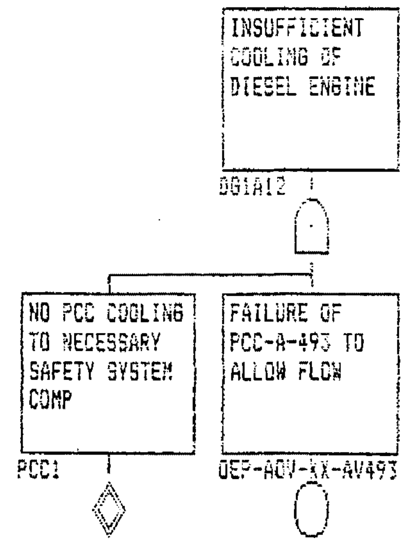


Figure H-3 (cont.)

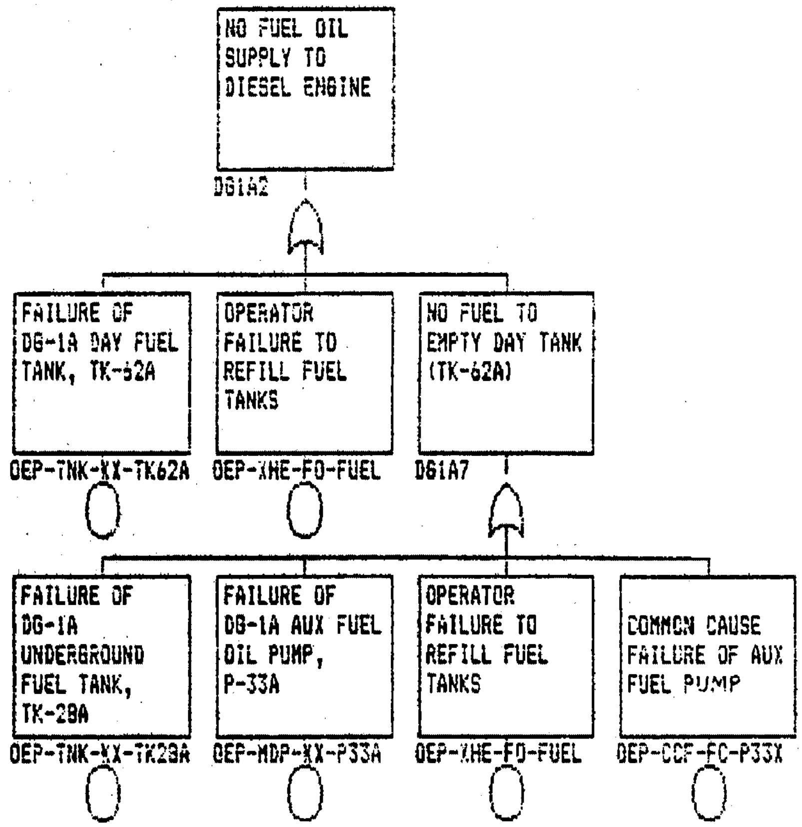


Figure H-3 (cont.)

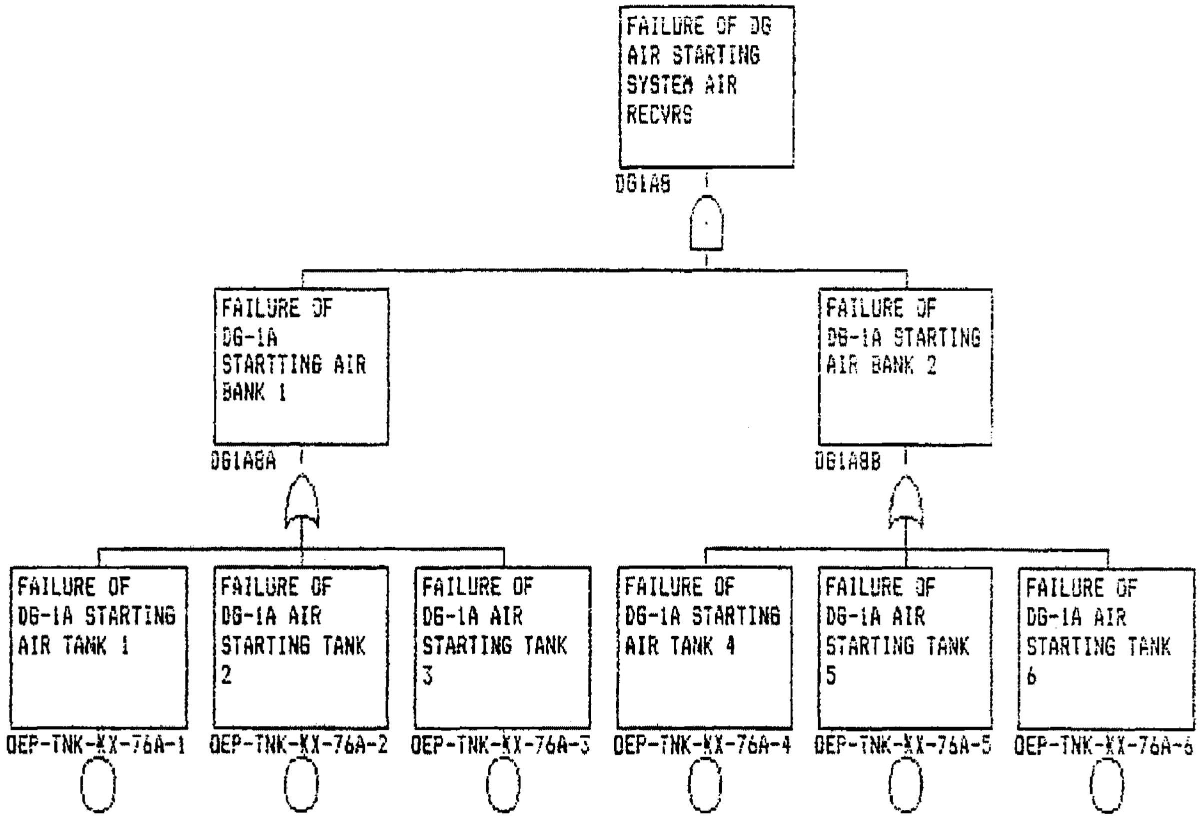


Figure H-3 (cont.)

APPENDIX I

ACTUATION SYSTEMS

Table I-1. Actuation systems.

Safety Function:	To initiate the engineered safeguard systems.	
System Components:		
	CIS	Containment Isolation System
	CSAS	Containment Spray Actuation System
	RAS	Recirculation Actuation System
	SIAS*	Safety Injection Actuation System
	RPS	Reactor Protection System
	- High Pressurizer Pressure (PSR)	
	- Low Steam Generator Level (SG LEV)	
Instruments:	LIC-303 AK,BK,CK	RWST Level (RAS-A)
	LIC-304 AK,BK,CK	RWST Level (RAS-B)
	LT-1213 A,B,C,D	Steam Generator 1 Level (SG LEV)
	LT-1223 A,B,C,D	Steam Generator 2 Level (SG LEV)
	LT-1223 A,B,C,D	Steam Generator 3 Level (SG LEV)
	PIA-102 A,B,C,D	Pressurizer Pressure (SIAS, PSR)
	PS-2003 A,B,C,D,E,F	Containment Pressure (CIS)
	PS-2009 A,B,C,D,E,F	Containment Pressure (CSAS)
	PS-2010 A,B,C,D	Containment Pressure (SIAS)
Support Systems:		
AC Power:	Channel A	120 VAC Vital Bus 1
	Channel B	120 VAC Vital Bus 2
	Channel C	120 VAC Vital Bus 3
	Channel D	120 VAC Vital Bus 4
DC Power:	CIS-A	125 VDC Batt 1
	CIS-B	125 VDC Batt 3
	CSAS-A	125 VDC Batt 1
	CSAS-B	125 VDC Batt 3
	RAS-A	125 VDC Batt 1
	RAS-B	125 VDC Batt 3
	SIAS-A	125 VDC Batt 1
	SIAS-B	125 VDC Batt 3

* SIAS is the only actuation signal included in the system FT.

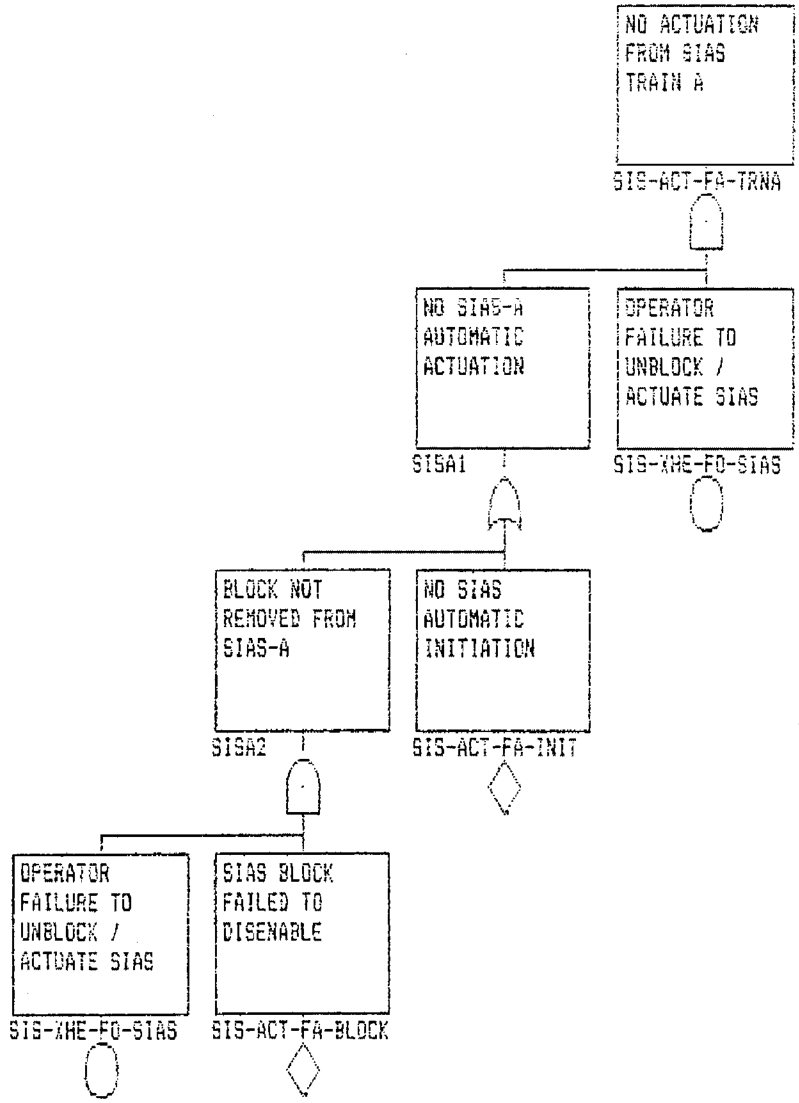


Figure I-1 Safety Injection Actuation System (Train A) Fault Tree.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION — **BIBLIOGRAPHIC DATA SHEET** — SEE INSTRUCTIONS ON THE REVERSE.	1. REPORT NUMBER *(Assigned by TIDC, add Vol. No., if any)* NUREG/CR-4826 UCID-20948 Vol. 2
2. TITLE AND SUBTITLE Seismic Margin Review of the Maine Yankee Atomic Power Station Volume 2. Systems Analysis	3. LEAVE BLANK
	4. DATE REPORT COMPLETED MONTH: February — YEAR: 1987
5. AUTHOR(S) D.L. Moore, D.M. Jones, M.D. Quilici and J. Young	6. DATE REPORT ISSUED MONTH: March — YEAR: 1987
7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* Energy Incorporated Kent, WA 98031 Under Contract to: Lawrence Livermore National Laboratory Livermore, CA 94550	8. PROJECT/TASK/WORK UNIT NUMBER 9. FIN OR GRANT NUMBER A0461
10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* Division of Engineering Safety Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555	11a. TYPE OF REPORT Technical b. PERIOD COVERED *(Inclusive dates)* June 1936 thru March, 1987
12. SUPPLEMENTARY NOTES	

13. ABSTRACT *(200 words or less)*

This Systems Analysis is the second of three volumes for the Seismic Margin Review of the Maine Yankee Atomic Power Station. Volume 1 is the Summary Report of the first trial seismic margin review. Volume 3, Fragility Analysis, documents the results of the fragility screening for the review. The three volumes demonstrate how the seismic margins review guidance (NUREG/CR-4482) of the NRC Seismic Design Margins Program can be applied.

The overall objectives of the trial review are to assess the seismic margins of a particular pressurized water reactor, and to test the adequacy of this review approach, quantification techniques, and guidelines for performing the review. Results from the trial review will be used to revise the seismic margin methodology and guidelines so that the NRC and industry can readily apply them to assess the inherent quantitative seimsic capacity of nuclear power plants.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS	15. AVAILABILITY STATEMENT
Seismic Margins Seismic Analysis	Unlimited
b. IDENTIFIERS/OPEN-ENDED TERMS	16. SECURITY CLASSIFICATION *(This page)* Unclassified *(This report)* Unclassified
	17. NUMBER OF PAGES
	18. PRICE

*U.S. GOVERNMENT PRINTING OFFICE:1987-181-682:60329

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE & FEES PAID
USNRC
WASH. D.C.
PERMIT No. G-67

NUREG/CR-4826
UCID-20948
Vol. 3

Seismic Margin Review of the Maine Yankee Atomic Power Station

Fragility Analysis

Prepared by M. K. Ravindra, G. S. Hardy, P. S. Hashimoto, M. J. Griffin

EQE, Inc.

Lawrence Livermore National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

NOTICE

Availability of Reference Materials Cited in NRC Publications

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

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

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

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

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

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

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

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

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

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

NUREG/CR-4826
UCID-20948
Vol. 3

Seismic Margin Review of the Maine Yankee Atomic Power Station

Fragility Analysis

Manuscript Completed: February 1987
Date Published: March 1987

Prepared by
M. K. Ravindra, G. S. Hardy, P. S. Hashimoto, M. J. Griffin

EQE, Inc.
Newport Beach, CA 92660

Under Contract to:
Lawrence Livermore National Laboratory
7000 East Avenue
Livermore, CA 94550

Prepared for
Division of Engineering Safety
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN A0461

ABSTRACT

This Fragility Analysis is the third of three volumes for the Seismic Margin Review of the Maine Yankee Atomic Power Station. Volume 1 is the Summary Report of the first trial seismic margin review. Volume 2, Systems Analysis, documents the results of the systems screening for the review. The three volumes are part of the Seismic Margins Program initiated in 1984 by the Nuclear Regulatory Commission (NRC) to quantify seismic margins at nuclear power plants.

The overall objectives of the trial review are to assess the seismic margins of a particular pressurized water reactor, and to test the adequacy of this review approach, quantification techniques, and guidelines for performing the review. Results from the trial review will be used to revise the seismic margin methodology and guidelines so that the NRC and industry can readily apply them to assess the inherent quantitative seismic capacity of nuclear power plants.

CONTENTS

ABSTRACT iii

LIST OF TABLES vii

LIST OF FIGURES viii

1. INTRODUCTION 1-1
   1.1 Background 1-1
   1.2 Objective of the Study 1-2
   1.3 Scope of the Study 1-2
   1.4 Organization of the Report 1-3

2. REVIEW EARTHQUAKE LEVEL 2-1
   2.1 Screening of Components 2-1
   2.2 Estimation of Seismic Capacities 2-1

3. DESCRIPTION OF PLANT STRUCTURES, SYSTEMS, AND COMPONENTS 3-1
   3.1 Maine Yankee Plant/Structures and Systems 3-1
      3.1.1 Structures 3-1
      3.1.2 Systems 3-1
   3.2 Seismic Design Criteria 3-2
   3.3 Availability of Plant Design Data 3-2

4. REVIEW OF PLANT INFORMATION AND WALKDOWN 4-1
   4.1 Initial Screening of Components 4-1
   4.2 Review of Design-Analysis and Seismic Reevaluation Reports 4-4
   4.3 Plant Walkdown 4-6
      4.3.1 Identification of Target Areas for First Walkdown 4-6
      4.3.2 Walkdown Procedures 4-22
         4.3.2.1 Walkdown Team 4-22

CONTENTS (Continued)

4.3.2.2 Procedures for Structures and Equipment 4-23

4.3.3 Walkdown Documentation 4-29

4.3.4 Walkdown Results 4-30

4.3.4.1 Walkdown Findings 4-30

4.3.4.2 Plant Unique Features 4-56

4.4 Maine Yankee Component Modifications 4-59

5. EVALUATION OF SEISMIC CAPACITIES OF COMPONENTS AND PLANT 5-1

5.1 Review of Structural Models 5-1

5.2 Simplified Analysis and Use of Screening Tools 5-2

5.3 Second Walkdown 5-4

5.4 HCLPF Capacity of Components 5-4

5.4.1 Fragility Analysis Method 5-4

5.4.1.1 Methodology 5-4

5.4.1.2 Steel Structures 5-23

5.4.1.3 Block Walls 5-30

5.4.1.4 Flat Bottom Storage Tanks 5-32

5.4.1.5 Inverter 5-40

5.4.1.6 Diesel Fuel Oil Day Tank 5-46

5.4.1.7 Containment Spray Fans 5-52

5.4.2 CDFM Method 5-53

5.4.2.1 Circulating Water Pumphouse 5-54

5.4.2.2 Block Wall SB 35-3 5-54

5.4.2.3 Refueling Water Storage Tank 5-55

5.5 HCLPF Capacity of Plant 5-56

5.5.1 Accident Sequences 5-56

5.5.2 Probabilistic Method 5-59

5.5.3 Deterministic Method 5-67

5.5.4 Sensitivity Studies 5-68

CONTENTS (Continued)

6. COMMENTS AND RECOMMENDATIONS ON SEISMIC MARGIN REVIEW METHODOLOGY 6-1
- 6.1 Selection of Review Earthquake Level 6-1
- 6.2 Use of Screening Guidelines 6-2
  - 6.2.1 Extent of Review Needed 6-2
  - 6.2.2 Additional Screening Guidelines 6-2
  - 6.2.3 Difficulties of Reviewing Certain Components 6-3
    - 6.2.3.1 Reactor Internals and CEDM 6-3
    - 6.2.3.2 Equipment Within Containment 6-3
- 6.3 Availability of Qualification Data 6-4
- 6.4 Walkdown Procedures 6-5
- 6.5 Guidance on CDFM Capacity Calculation Procedures 6-6
- 6.6 Staffing Requirements and Schedule 6-7
  - 6.6.1 Staffing Requirements 6-7
  - 6.6.2 Schedule 6-8
- 6.7 Applicability of Methodology for Other Plants 6-8

7. RESULTS AND CONCLUSIONS 7-1
- 7.1 Screened Out Systems and Components 7-1
- 7.2 HCLPF Capacities of Screened-in Components 7-2
- 7.3 HCLPF Capacity of Plant 7-32
- 7.4 Identification of Low Capacity Components 7-32
- 7.5 Conclusions 7-32

REFERENCES R-1

APPENDIX

A. Maine Yankee Atomic Power Station Arrangement Drawings A-1

LIST OF TABLES

3.2-1 Original Design Damping Values 3-5

4.1-1 Initial Screening of Maine Yankee Components by Categories Based on Seismic Capacities 4-2

4.3-1 Structure Separations 4-8

4.3-2 List of Block Walls Targeted for Walkdown 4-9

4.3-3 Maine Yankee Atomic Power Plant First Walkdown List of Equipment Components 4-13

4.3-4 Example Pump Walkdown Data Sheet 4-31

4.3-5 Example HVAC Components Walkdown Data Sheet 4-33

4.3-6 Example Block Wall Data Sheet 4-35

5.3-1 Second Walkdown Component Review List 5-5

5.4-1 Diesel Fuel Oil Tank Critically Stressed Areas 5-51

5.5-1 Component Seismic Fragility Parameters 5-57

5.5-2 Probabilities for Nonseismic Failures 5-58

5.5-3 Summary of Plant Level HCLPF Capacities 5-64

7.1-1 Summary of Group A Structures 7-2

7.1-2 Summary of Group A Block Walls 7-3

7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities 7-6

LIST OF FIGURES

2.1	0.3g pga Review Earthquake Ground Response Spectra, 5% and 10% Damping	2-2
3.2-1	0.05g Design Earthquake Ground Response Spectra	3-3
3.2-2	0.10g Hypothetical Earthquake Ground Response Spectra	3-4
4.3-1	Block Wall Evaluation Procedure	4-38
4.3-2	Example of a PORV at the El Centro Steam Plant	4-44
4.3-3	Maine Yankee PORV PR-S-14 and PR-S-15	4-45
4.3-4	Maine Yankee CS Pump Suction MOV's CS-M-91 and CS-M-92. Valves with Long Operator Arms	4-47
4.3-5	Example of Long Operator Valves Documented in the Data Base from the Coalinga Area	4-48
4.3-6	Steam Lines Spanning Between the Steam and Feed Water Valve Area and the Turbine Building	4-50
4.3-7	Photographs of Typical Maine Yankee Cable Trays	4-52
4.3-8	Photographs of Typical Maine Yankee Instrument Racks and Component Attachments	4-54
4.3-9	Photographs of the Maine Yankee Control Room Ceiling	4-55
4.4-1	Block Wall VE 21-1	4-57
4.4-2	Conceptual Seismic Retrofit to Block Wall VE 21-1	4-58

LIST OF FIGURES (CONTINUED)

4.4-3 Photograph of the Alarm Message Display Mounted on the Main Control Board 4-60

4.4-4 Photographs of the Maine Yankee Station Service Transformer Internal Core/Coil Assembly Before the Anchorage Modification 4-62

4.4-5 Proposed Anchorage Modification for the Maine Yankee Station Service Transformers Internal Core/Coil Assemblies 4-63

4.4-6 Maine Yankee Air Conditioners in the Unmodified Condition 4-64

4.4-7 Proposed Anchorage Modifications for the Maine Yankee Air Conditioners 4-65

4.4-8 Maine Yankee Containment Spray Fans Before the Installation of the Anchorage Modifications 4-67

4.4-9 Proposed Anchorage Modifications for the Maine Yankee Containment Spray Fans 4-68

4.4-10 Photographs of the Unstrapped Emergency Lighting Units in the Control Room and Welding Cart Near the Containment Spray Fans 4-69

5.4-1 Fragility Curves 5-9

5.4-2 Median, 5% Nonexceedance, and 95% Nonexceedance Fragility Curves for a Component 5-11

5.4-3 Family of Fragility Curves for a Component 5-13

LIST OF FIGURES (CONTINUED)

5.4-4 Circulating Water Pumphouse 5-24

5.4-5 Circulating Water Pumphouse, Typical Braced Frame 5-25

5.4-6 Turbine/Service Building, El. 35'-0"/El. 39'-0" 5-28

5.4-7 Turbine/Service Building, El. 61'-0" 5-29

5.4-8 Block Wall SB35-3 5-31

5.4-9 Refueling Water Storage Tank 5-34

5.4-10 Model for Determination of Tank Resistance Against Seismic Base Moment 5-36

5.4-11 Beam Model of Tank Bottom Plate 5-37

5.4-12 Yield Line Model of Anchor Bolt Chair Top Plate 5-39

5.4-13 Maine Yankee 12 kVA Inverter and Battery Charger, and a Close-up of the Anchorage Addition 5-41

5.4-14 Diesel Fuel Oil Day Tank (TK-62B) 5-47

5.4-15 Diesel Day Tank Critical Areas 5-48

5.4-16 Diesel Day Tank Critical Areas 5-49

5.4-17 Retrofit Anchorage Design for the Day Tank 5-50

5.5-1 Seismic Fragility Curves of a Component 5-60

5.5-2 Fragility Curves for Small LOCA Core Damage 5-62

5.5-3 Seismic Fragility Curves for No LOCA Core Damage 5-63

CHAPTER 1
INTRODUCTION

1.1 Background

A seismic margin review methodology has been developed by the NRC Expert Panel as documented in [Budnitz et al., 1985] and [Prassinos, et al., 1985]. The objective of the methodology is to estimate a high confidence value of the seismic capacity of a nuclear power plant. The methodology has been derived using the results and insights gained in conducting over 20 published and unpublished seismic probabilistic risk assessments, the data on actual performance of structures and equipment of industrial facilities and power plants in major earthquakes, and the data on qualification of equipment in nuclear power plants. The methodology consists of selecting a review earthquake level, screening out categories of plant structures and equipment which have seismic capacities generically higher than the review earthquake level, screening out components that are not essential for certain plant safety functions, performing plant walkdown to confirm that the screening of components (i.e., structures and equipment) is acceptable, and estimating the seismic capacities of the components and of the overall plant. The seismic capacity of a component or plant in this context is the so-called High Confidence Low Probability of Failure (HCLPF) capacity; it is a conservative representation of capacity and in simple terms corresponds to the earthquake level at which it is extremely unlikely that failure will occur. It can be viewed as approximately equal to the earthquake level for which we have 95% confidence that the probability of failure is less than 5%. At this time, seismic margin methodology has been developed for pressurized water reactors based on the review of system analysis results and insights of a number of seismic PRAs on these reactors; for boiling water reactors additional reviews of seismic PRAs are required to develop seismic margin review procedures.

The seismic margin review methodology has been developed and some preliminary guidelines for performing the margin review have been indicated in the foregoing references. It was felt necessary to apply the methodology to a selected plant on a trial basis in order to check whether the methodology is implementable and whether the guidelines need further revision and amplification. For this purpose, the Maine Yankee Atomic Power Plant was selected. It is expected that the methodology and margin review guidelines will be revised subsequent to this trial plant review.

The NRC selected the review level earthquake to be 0.3g peak ground acceleration with a NUREG/CR-0098 50th percentile spectral shape. Maine Yankee concured with this selection.

The trial plant review is conducted by the Lawerence Livermore National Laboratory (LLNL) and is being funded by the U.S. Nuclear Regulatory Commission. It consists of two major aspects: system analysis and fragility evaluation. The contracts to perform these two aspects of the trial plant review have been awarded to Energy International Incorporated and EQE Incorporated, respectively. Volume 3 of the report describes the fragility evaluation aspect of this seismic margin review. Volume I summarizes the entire seismic margin review study on Maine Yankee; Volume II discusses the system analysis aspects of the study.

1.2 Objective of the Study

This fragility aspect of the trial plant seismic margin review is aimed at achieving the following objectives:

- o Apply the seismic margin review methodology developed by the NRC Expert Panel and identify areas where the margin review guidelines need modification and clarification,
- o Estimate the HCLPF seismic capacity of the Maine Yankee Atomic Power Plant and identify any seismic vulnerabilities in the plant.

1.3 Scope of the Study

The primary purpose of this study has been to evaluate the NRC Expert Panel seismic margin review methodology using Maine Yankee as the trial plant; the secondary purpose is to assess the seismic margin of Maine Yankee. The scope of the study is defined with these two purposes in view.

- o The methodology outlines two approaches for estimating the component and plant-level seismic capacities: probabilistic and deterministic. The guidelines for the deterministic evaluation of the capacities are not sufficiently developed. Therefore, only the probabilistic approach is used in estimating both the component and plant-level seismic capacities in this study. Although this does not check out all the features of the Expert Panel methodology, it does provide an estimate of the seismic margin of Maine Yankee. Much further work is needed to develop more definitive guidelines for deterministic evaluation of seismic margins in the Expert Panel methodology.
- o The seismic capacities of structures and equipment are estimated using the structural models and qualification analysis results provided by the Maine Yankee utility. The adequacy of the structural models and the reasonableness of the seismic responses of structures and equipment were confirmed by cursory review and based on judgment. The floor response spectra generated by Maine Yankee are judged to be adequate for this seismic margin review. However, this cannot be construed to be a detailed review duplicating the review done as part of the plant QA/QC program in its licensing.
- o The limitations of the Seismic Margin Review Methodology as outlined in the cited references by the Expert Panel also apply to

the study described in this report (i.e., relay issues, design and construction errors, operator errors under seismic stress, etc).

- o The screening criteria for components developed by the Expert Panel based on their generic high seismic capacities (i.e., components denoted by letter "C" in Table 5-1 of NUREG/CR-4334) are assumed to be applicable using minimal review and engineering judgment. In general EQE is in agreement with the Panel's recommendations.

1.4 Organization of the Report

Chapter 2 discusses the review level earthquake specified for this trial plant seismic margin review. A general description of the Maine Yankee plant structures, systems, and components is given in Chapter 3. The processes of plant design review, initial screening of components, and the plant walkdown are detailed in Chapter 4. Review of structural models, simplified analysis and second walkdown for additional data, and HCLPF capacity calculation for components and plant are described in Chapter 5. Feedback on the methodology in the areas of selection of review earthquake level, screening guidelines, walkdown procedures, HCLPF capacity calculation, staffing requirements, and applicability to other plants are discussed in Chapter 6. The results of the study in terms of HCLPF capacities of components and plant, and any seismic vulnerabilities in the plant, are highlighted in Chapter 7. Appendix A consists of a set of general arrangement drawings showing the structures and equipment in the Maine Yankee plant.

CHAPTER 2

REVIEW EARTHQUAKE LEVEL

For this trial application of the NRC Expert Panel seismic margin review methodology, the NRC staff has specified a review earthquake level of 0.30g pga anchored to the median NUREG/CR-0098 ground response spectrum for rock sites (Figure 2.1). In this chapter, the implications of this review earthquake level will be discussed from the viewpoint of screening of components and seismic capacity calculations.

2.1 Screening of Components

The guidelines developed by the Expert Panel for screening of components based on their generic seismic capacities are considered applicable for review earthquakes of magnitudes less than 6.5, with 3 to 5 strong motion cycles and a total duration of 10 to 15 seconds. The spectral content of this earthquake is characterized by a broad-band spectra in the structural frequency range of 1 to 7 Hz.

The selected review earthquake for Maine Yankee meets all of the above requirements. It is, therefore, concluded that the screening guidelines given in the Expert Panel reports NUREG/CR-4334 and 4482 are applicable to the present seismic margin review.

2.2 Estimation of Seismic Capacities

The concept of HCLPF capacity requires that it is associated with a defined response spectrum and a specified nonexceedance probability.

The HCLPF capacity used for screening as well as the calculated HCLPF capacities for particular components not initially screened out and the final plant level HCLPF capacity are considered to be valid provided ground motion from any earthquake does not exceed the review earthquake level spectrum for more than 16% of the spectral frequencies within the range of interest. The review earthquake level spectrum is a spectral shape defined by the 50% exceedance spectrum specified in NUREG/CR-0098 and anchored at 0.3g pga for the initial screening. The seismic margin for the components and plant is referenced to this spectrum but anchored to the pga corresponding to the HCLPF capacity.

This definition of spectra used to determine a HCLPF capacity does not in any way refer to the probability of occurrence of an earthquake. It is no more than an arbitrary spectrum used to define the HCLPF capacity that recognizes the dependency of a component capacity on the frequency content of the spectrum and not just the pga.

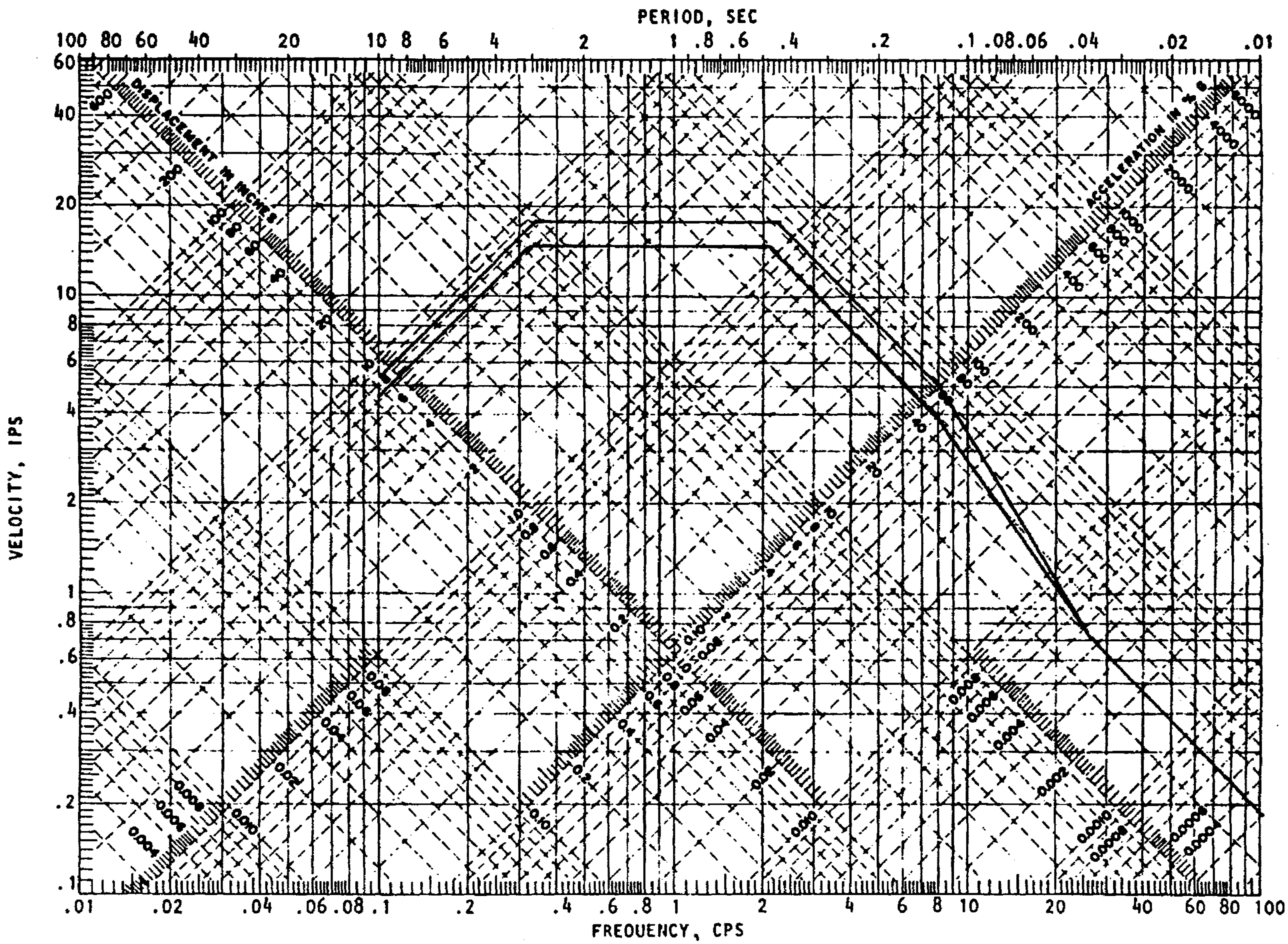


Figure 2.1 0.3g pga Review Earthquake Ground Response Spectra, 5% and 10% Damping.

The results of the seismic margin study are interpreted as follows. The HCLPF capacity of the structures, equipment and plant are conditional on the actual site-specific spectrum not exceeding the target spectrum; exceedance is defined as the event when 16 percent of the spectral ordinates exceed the target spectrum over the frequency range of interest. It is assumed that the spectrum peak-to-peak and earthquake direction variabilities (Sec. 5.4.1.1) are removed from the hazard analysis leading to the selection of the review earthquake. The review earthquake is specified by the same spectrum (Figure 2.1) in two horizontal directions and 2/3 of the horizontal spectrum in the vertical direction. It is also assumed that the review earthquake level is specified as the higher of the response spectra from the two orthogonal horizontal directions.

CHAPTER 3

DESCRIPTION OF PLANT STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Maine Yankee Plant/Structures and Systems

Maine Yankee plant is located on the west shore of the Back River approximately 3.9 miles south of the center of Wiscasset, Maine. The plot plan is shown in Appendix A.

Maine Yankee is a one-unit, 3-loop PWR supplied by Combustion Engineering with a rated capacity of 825 MWe. It began commercial operation in December 1972.

3.1.1 Structures

The major structures on the site are the reactor containment, primary auxiliary building, fuel building, turbine building, service building, and circulating water pumphouse. They are founded on hard rock.

The reactor containment is a steel-lined reinforced concrete cylinder with a hemispherical dome and an essentially flat reinforced concrete foundation mat.

The turbine building houses the turbine generator and the two diesel generators. The service building consists of the main control room, switchgear room, shops, and employee facilities. Portions of the turbine/service building, control/switchgear building, and diesel generator enclosure have reinforced concrete walls and slabs. The service building above El. 39 ft 0 in. and the remaining portions of the turbine/service building are of structural steel framing with diagonal bracing and reinforced concrete slab with metal roof deck.

The primary auxiliary building houses pumps, and tanks used for purification and processing of water from the reactor coolant system. It is made of reinforced concrete walls and slabs.

The circulating water pumphouse contains the circulating water pumps and service water pumps. Below El. 21 ft 0 in., the structure has reinforced concrete walls and slabs. Above this elevation, it is fabricated structural steel framing with diagonal bracing and reinforced concrete slab.

Other structures of interest to this study are the ventilation equipment room, containment spray pumphouse, and the main steam valve house. All of these structures are constructed of reinforced concrete walls and slabs except that the interior structure of the main steam valve house is of structural steel framing with diagonal bracing and metal grating.

3.1.2 Systems

In this study, the main focus is on the systems (i.e., front line and supporting) that support the Group A System functions. These systems are described in Volume 2 of this report.

3.2 Seismic Design Criteria

The original design of the Class I structures and components was based on a "Housner Spectrum" anchored to 0.05g for the design earthquake and 0.10g for the hypothetical earthquake (SSE). (See Figures 3.2-1 and 3.2-2.)

The damping values used for the design of different structures, piping and equipment are summarized in Table 3.2-1.

The following structures and components are characterized as Class I:

- Reactor containment, including penetrations
- Reactor vessel and its internals
- Reactor coolant system
- Reactor containment crane
- Chemical and volume control system
- Residual heat removal system
- Safety injection system
- All components affecting the ability of the control rods to scram
- Containment spray system
- Spent fuel pool and racks
- Component cooling system
- Circulating water system intake structure
- Service water system
- Emergency generators
- Refueling water storage tank
- Control room
- Emergency steam generator feed pumps and piping

The structures and components (including piping, cable trays, and HVAC systems) were designed and qualified to the requirements of the applicable AISC, ACI, and ASME codes of 1960 to 1970 versions.

The plant structures and equipment have been reevaluated at various times for earthquakes larger than the original SSE and for more recent regulatory standards [Stevenson, 1983]; [Hashimoto et al., 1984]; [Whittier, 1986]. Certain upgrades to improve the seismic capacity of the plant have also been made, e.g., additional anchoring of electrical equipment, and strengthening of block walls. The present study has focused on evaluating the seismic capacity of the plant in its current state, including certain modifications installed during the course of the margin review, or to be installed during the March 1987 outage.

3.3 Availability of Plant Design Data

Maine Yankee is an older plant designed and constructed before the development of quality assurance programs and seismic qualification methods presently existing in the nuclear industry. Hence, the qualification reports on certain equipment items were not available. Because of the reevaluation efforts, additional information and new structural models have become available. However, the extent of information available is not comparable to that normally available for a modern nuclear power plant (e.g., near term operating license plant).

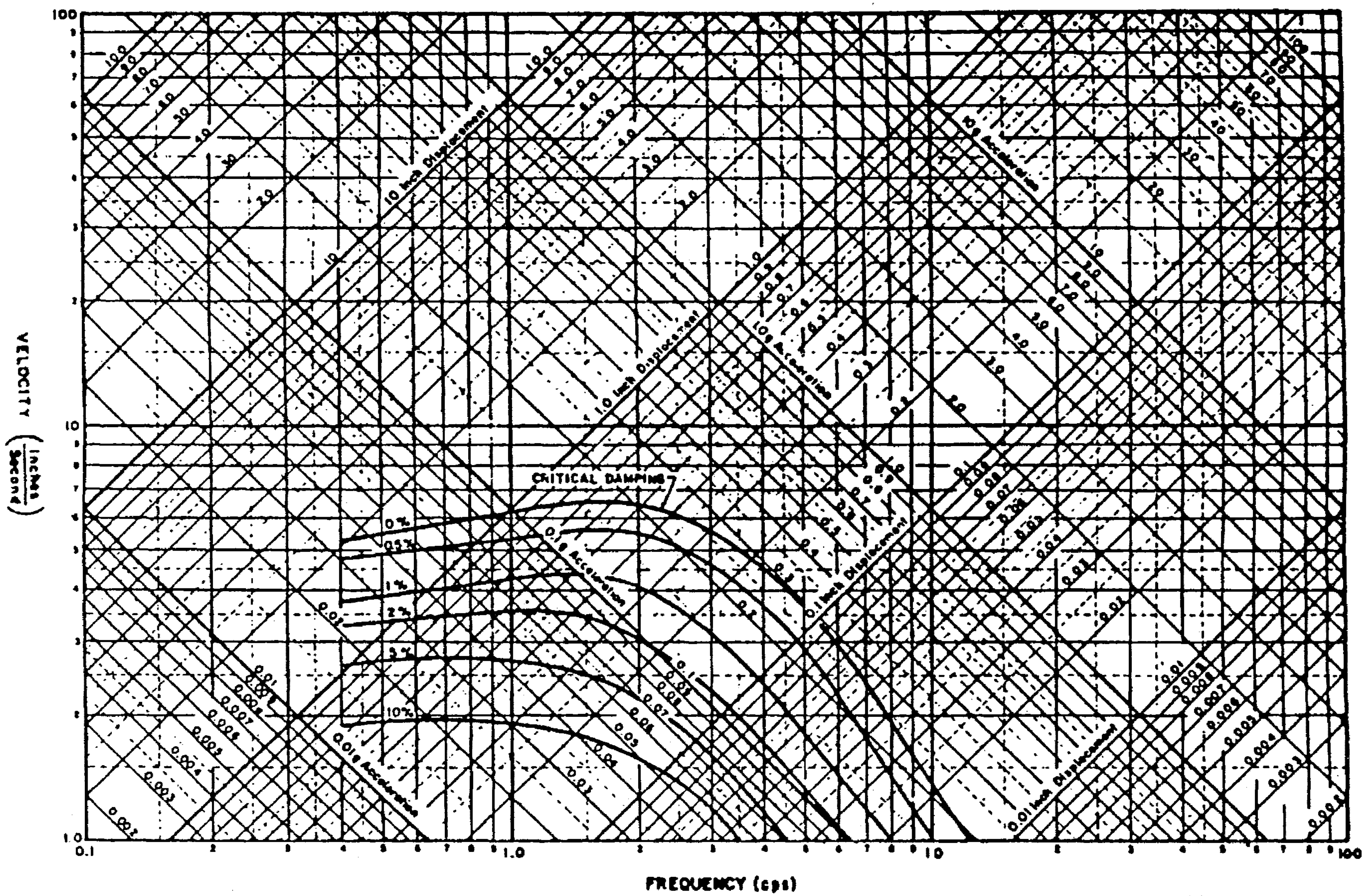


Figure 3.2-1 0.05g Design Earthquake Ground Response Spectra.

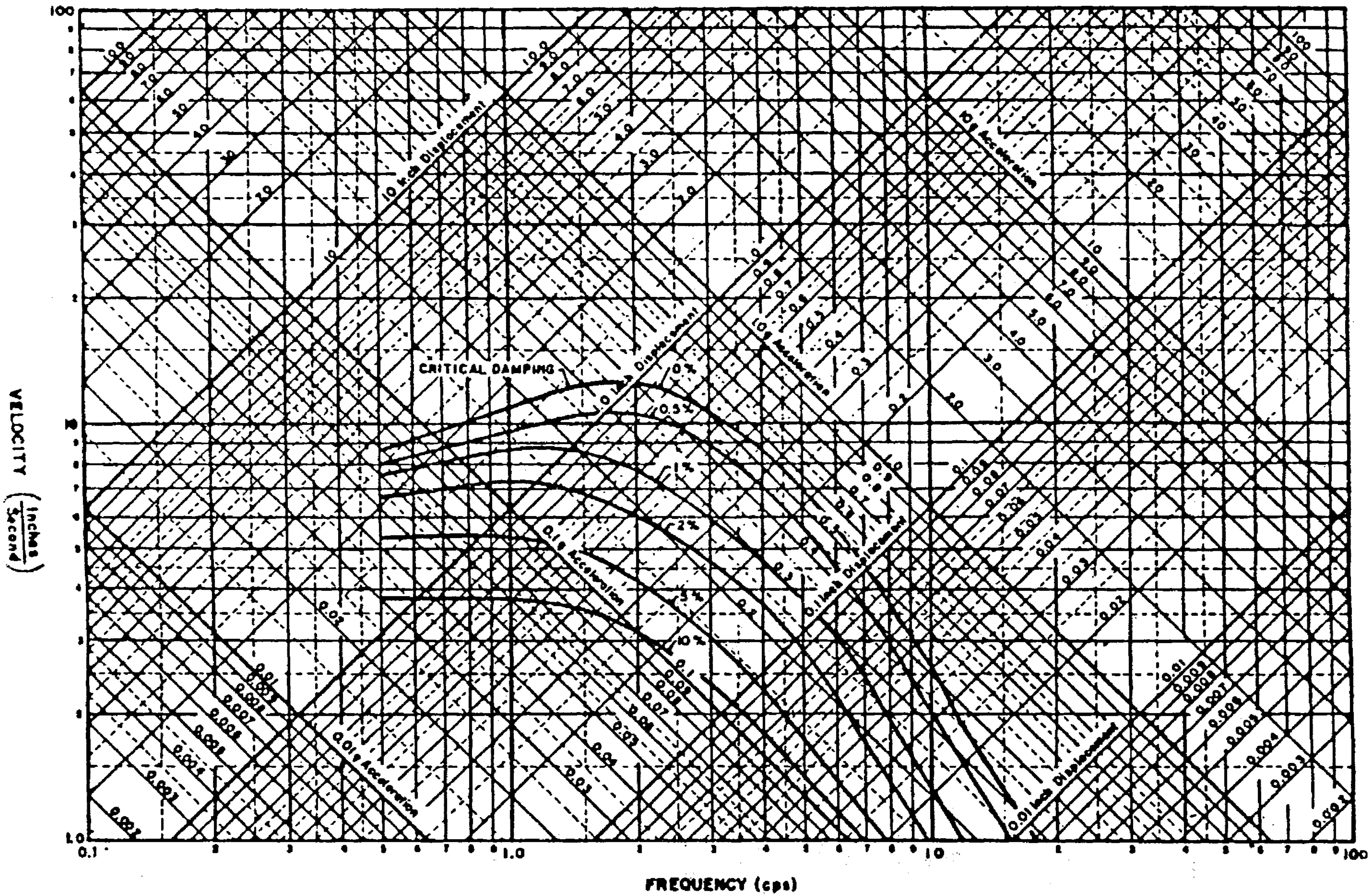


Figure 3.2-2 0.10g Hypothetical Earthquake Ground Response Spectra.

Table 3.2-1 Original Design Damping Values

	Percent of Critical Damping: Design Earthquake	Percent of Critical Damping: Hypothetical Earthquake
1. Reactor containment structure	2.0	5.0
2. Reinforced concrete structure, other than containment structure, founded on rock or soil	2.0	5.0
3. Reinforced concrete supporting structure (not founded on soil or rock)	2.0	5.0
4. Steel-framed structures, including supporting structures and foundations		
Bolted or riveted	3.0	5.0
Welded	1.0	2.0
5. Reactor vessel, internals and control rod drives		
Welded assemblies	1.0	1.0
Bolted assemblies	3.0	3.0
6. Mechanical equipment,including pumps, fans and similar items	2.0	2.0
7. Piping systems	1.0	2.0

The structural models were generally reviewed by EQE; the floor response spectra generated for the review earthquake spectrum were provided by Maine Yankee. The spectra were judged to be realistic and representative of the seismic response for the review earthquake. Seismic capacities of structures and equipment were estimated by comparing the original design-analysis (or reevaluation) spectral values with the new spectral values and taking into account the margins and variabilities due to differences in damping, methods of analysis and testing, and procedures for mode combination and directional components. Where the original design analysis or reevaluation information was not available or applicable, the structure or equipment was analyzed by EQE to establish its seismic capacity.

CHAPTER 4

REVIEW OF PLANT INFORMATION AND WALKDOWN

In this chapter, a discussion of the plant information gathered during a review of drawings, and FSAR and seismic analysis/qualification reports will be given. The plant design information for systems and components supporting Group A functions is used to perform the initial screening of components based on the Expert Panel recommendations. The review of plant information is also aimed at identifying target areas and developing a strategy for the first plant walkdown. This chapter describes the procedures used in the plant walkdown, documentation, and salient findings.

4.1 Initial Screening of Components

Energy Incorporated provided a list of systems and equipment at Maine Yankee that support the Group A functions. From this list the structures containing the equipment were identified. NUREG/CR-4334 gives guidance on screening out certain structures and equipment from further consideration based on their generically high seismic capacities.

For the chosen review earthquake level of 0.3g pga, the Expert Panel report recommends that the margin review may generically screen out the component categories after satisfying some caveats as shown in Table 4.1-1. Since the scope of this study includes both an evaluation of the seismic margin review methodology and a trial application to the Maine Yankee Atomic Power Station, the initial screening could not eliminate a large number of generic categories of components from further consideration in the review, walkdown, and analysis. This is so because the Expert Panel's recommendations on screening had to be generally confirmed as appropriate to Maine Yankee.

The following items were screened out at this stage:

- o Containment structure
- o NSSS supports
- o Soil liquefaction potential

For the screened-in components, further evaluation consisted of two levels:

1. For those components identified by the Expert Panel as having HCLPF capacities larger than 0.3g pga, a minimal evaluation was done during plant review and walkdown to confirm the applicability of the Panel's recommendations for the Maine Yankee component categories.

   This was done in the following manner:

   - o Control rod drive mechanism: Review of CE design
   - o Structures: All concrete structures housing Group A systems and equipment were reviewed
   - o Valves: Sampling review

Table 4.1-1 Initial Screening of Maine Yankee Components by Categories Based on Seismic Capacities

Component	Expert Recommendation	Remarks
1. Containment	C	Maine Yankee has a reinforced concrete containment structure; previous study by Hashimoto, et al (1984) confirms that the HCLPF capacity is in excess of 1g.
2. NSSS Supports	C	Studies done by LLNL as part of DEGB evaluation included the Combustion Engineering (CE) NSSS Supports. The NSSS Supports have HCLPF capacities in excess of 0.3g.
3. Reactor Internals	X	To be evaluated.
4. Control Rod Drive Mechanism	C	To be confirmed by a review of the CE design.
5. Concrete Structure Failures (Shearwalls, diaphragms, impact)	C	To be confirmed by a review and walkdown.
6. Steel Structures	X	To be evaluated.
7. Block Walls	X	To be evaluated.
8. Piping	X	Panel's caveats to be addressed.
9. Valves	C	To be confirmed during walkdown.
10. Heat Exchangers	X	Support and anchorage to be evaluated.
11. Tanks	X	To be evaluated.

Table 4.1-1 Initial Screening of Maine Yankee
Components by Categories Based on Seismic Capacities (Continued)

Component	Expert Recommendation	Remarks
12. Batteries and Racks	X	To be evaluated during review and walkdown.
13. Active Electrical Equipment	X	Anchorage to be reviewed.
14. Diesel Generators	C	To be confirmed during walkdown.
15. Pumps	C	To be confirmed during walkdown.
16. Soil Liquefaction	C	Rock site.
17. HVAC Systems		
Fans and Cooler Units	X	Units on vibration isolators to be reviewed
Ducting	C	To be confirmed during walkdown.
18. Cable Trays and Cabling	C	To be confirmed during walkdown.
19. Control Room Ceilings	X	To be evaluated.
20. Dams, Levees and Dikes	X	To be evaluated.

C = HCLPF capacity is larger than 0.3g pga.

X = HCLPF capacity needs to be established through review, walkdown, and/or calculations.

- o Diesel generators and peripherals: Complete review
- o Pumps: All pumps were evaluated
- o HVAC ducting: Sampling review
- o Cable trays and cabling: Sampling review

2. For those components identified by the Expert Panel as requiring a review and walkdown, a detailed evaluation was performed as described in this report; it includes steel structures housing Group A systems.

4.2 Review of Design-Analysis and Seismic Reevaluation Reports

Initial and subsequent data collection efforts concentrated on those structures and components identified by Energy Incorporated required for the reactor subcriticality and early emergency core cooling. The plant specific seismic qualification information was primarily made available to EQE by Maine Yankee. In a few instances, outside vendors were contacted for additional information that was either lacking or proprietary for certain components (e.g., reactor vessel internals and the control element drive mechanism). The following provides a list of the types of information collected for the Maine Yankee Atomic Power Station as part of the margin evaluation.

The data collected for use in the margin evaluation can be organized into four main categories:

- o Drawings
- o Maine Yankee reports and calculations
- o Independent review and reports
- o External information

Drawings

Types of drawings collected from Maine Yankee include the following:

- o Maine Yankee structural, architectural, and excavation design drawings

- o Design sketches of block wall seismic retrofits

- o Maine Yankee plant general arrangement drawings including floor elevations showing equipment locations

- o Support and or anchorage drawing details for equipment

- o Equipment vendor drawings indicating component construction (e.g., configuration, size, and materials used)

Maine Yankee Reports and Calculations

Typical types of reports and calculations collected include the following:

- Sections from the Final Safety Analysis Report (FSAR)
- Structural steel, roof deck, and block wall construction specifications
- Tables summarizing Maine Yankee block wall information
- Maine Yankee generated calculations for seismic evaluation of block wall and retrofit design
- Maine Yankee generated calculations for equipment seismic qualification.
- A review of the Maine Yankee FSAR Amendment No. 35, Volume II, which documents the seismic qualification of vital instrument and electrical equipment. The amendment consists primarily of certified letters from vendors regarding conformance of their components to the seismic requirements of the Maine Yankee component procurement specification; however, a few calculations were provided and used in the component evaluations.

Independent Reviews and Reports

Several independent reviews and reports conducted for selected structures and equipment components at Maine Yankee collected include:

- Cygna report describing structure dynamic analysis models (Cygna Report BM-Y-MY-80006-5, April 1982)
- Cygna reports describing dynamic analyses performed for several Maine Yankee critical equipment components
- Cygna computations for building floor spectra generation using the 0.18g pga NUREG/CR-0098 50th Percentile Ground Response Spectra
- Report by J. D. Stevenson, "Seismic Review of the Maine Yankee Nuclear Power Plant," which analyzed several critical components at Maine Yankee
- Report by Structural Mechanics Associates, Inc., "Conservative Seismic Capacities of the Maine Yankee Reactor Containment Including and Excluding Design Incident Pressure," [Hashimoto et al., 1984].

External information not available through Maine Yankee

Several examples of information collected include:

- Outline drawings and seismic qualification data for the Maine Yankee reactor vessel internals and the control element drive mechanism obtained from Combustion Engineering.
- Information regarding dimensional data and support configuration for the Maine Yankee station service transformer internal core/coil assembly collected from contacts with the manufacturer, General Electric Medium Voltage Transformer Division.
- Lateral load capacity of vibration isolators supporting the Maine Yankee computer room air conditioners and the laboratory air conditioner, collected from contacts with the manufacturer, Vibration Mountings and Controls Inc..

During the course of the margin evaluation, several requests were made for additional plant or component qualification data. The additional requests were required as a result of the following:

- The equipment component list was being refined, adding and deleting systems and components
- Low capacity components from the first plant walkdown were identified requiring additional component specific data to complete the evaluation.

4.3 Plant Walkdown

4.3.1 Identification of Target Areas for First Walkdown

Target areas for the first walkdown were developed from the initial equipment list provided by the system analysts. This list identified preliminary equipment components as critical to reactor subcriticality and early emergency core cooling. From the preliminary equipment component list the critical structures housing this equipment were identified. The following provides a list and brief discussion of the structures and equipment identified as target areas for the first walkdown.

Structures

Structures identified for the first walkdown were those identified as housing Group A components.

- Containment structure
- Primary auxiliary building
- Circulating water pumphouse
- Turbine/service building
- Containment spray pumphouse
- Main steam valve house
- M.C.C. room
- Aux feed pumphouse and purge air exhaust area

- o Fire water pumphouse
- o Fuel oil pumphouse
- o Appendix R diesel room

The containment internal structure was not targeted for walkdown due to inaccessibility. The location of these structures are identified on the plant layout drawings in Appendix A.

Structure Separations. Based upon a review of the structural drawings, a number of separations involving Group A structures were identified. These separations are listed in Table 4.3-1 along with their gap widths as indicated on the design drawings.

Block Walls. Summary tables providing information on the Maine Yankee block walls were available prior to the first walkdown. These tables describe the locations of all block walls in the plant, identify any safety-related equipment on or near the block walls, and categorize the block walls in terms of their seismic safety status.

These summary tables were used to develop a list of block walls to be inspected during the first walkdown. This list is shown in Table 4.3-2. Nearly all block walls in the plant are included in this list. Certain block walls were excluded because they are located in areas that obviously did not house Group A components, based upon a comparison of block wall and equipment locations. These walls are located in the administration building, front office, fuel building, gas house, LSA storage building, RCA building, office areas of the service building, and parts of the yard. Walls inside containment were not targeted since they were known to be inaccessible for the walkdown. Block walls affecting safety related equipment that were not walked down are assumed to have seismic capacities comparable to Group A walls.

Dams, Levees, and Dikes. Review of the drawings indicated that water for the fire pond is enclosed by a dike adjacent to the fire water pumphouse. Because of the proximity of the fire water pond to the plant, this dike was targeted for walkdown to determine if it could fail during a seismic event and cause flooding.

Equipment Components

The Maine Yankee structural drawings and general plant layout drawings were reviewed to locate the preliminary equipment items identified by the system analyst. A general walkdown sequence organized by structure was developed for maximum use of time during the first walkdown. Walkdown data sheets were developed for specific classes of equipment components to be used in recording manufacturer and dimensional information necessary for a fragility evaluation (reference Section 4.3.3 for a discussion and example of a typical walkdown sheet). In most cases the equipment walkdown data sheets were lengthy as sufficient vendor information had not been received from Maine Yankee prior to the first walkdown. The first walkdown list of equipment components reviewed at the Maine Yankee Atomic Power Station are listed in Table 4.3-3.

Table 4.3-1 Structure Separations

Structures	Floor Elevations	Separation Gap
Containment, containment spray pumphouse	El. 14'-6, El. 30'-0, El. 40'-0	3"
Containment, ventilation equipment room	El. 21'-0, El. 40'-0	3"
Containment, main steam valve house	El. 21'-0, El. 68'-0	3"
Containment, M.C.C. room	El. 21'-0, El. 46'-0, El. 68'-0	3"
Containment, aux. feed pumphouse/ purge air exhaust room	El. 9'-0, El. 22'-0, El. 33'-0	3"
Containment, fuel building	El. 44'-6	12"
Containment, equipment hatch shield	El. 20'-0, El. 54'-0	3"
Containment spray pumphouse, ventilation equipment room	El. 21'-0, El. 40'-0	3"
Ventilation equipment room, main steam valve house	El. 21'-0, El. 40'-0	3"
Primary auxiliary building, service building	El. 35'-0	3"

Table 4.3-2 List of Block Walls Targeted for Walkdown

Maine Yankee Wall ID No.	Building	Room Location	Wall Location
ARD 20 1-2	Appendix R Diesel	El. 21'	Dividing Wall
CS -4 1	Containment Spray	Sump El. 4'-0"	RHR HX 3A Sump
FPH 20 1-2	Fire Pump House	El. 20'-0"	Two walls near P-4
PAB 11 1	Primary Auxiliary	Door Leading to Filter Cubicles El. 11'-0"	East Door Jam
PAB 11 2	Primary Auxiliary	El. 11'-0"	Shielding Blocks Around Letdown Lines
PAB 11 3	Primary Auxiliary	El. 11'-0"	Loose Blocks Near East Wall of the Primary Drain Tk Cubicles
PAB 11 4	Primary Auxiliary	El. 11'-0"	Loose Blocks on East Side of Seal Water Cooler Cubicle
PAB 21 1-6	Primary Auxiliary	Boric Acid Storage Area El. 21'-0"	Between Lines 7 and 9 and Columns F and H
PAB 21 7	Primary Auxiliary	Deaerator Vent Condenser Cubicle El. 21'-0"	East Wall
PAB 21 8	Primary Auxiliary	Waste Evaporator Cubicle El. 21'-0"	East Wall
PAB 21 9	Primary Auxiliary	Sampling Cubicle Behind Charging Pumps, El. 21'-0"	North Wall

Table 4.3-2 List of Block Walls Targeted for Walkdown (Continued)

Maine Yankee Wall ID No.	Building	Room Location	Wall Location
PAB 21 10	Primary Auxiliary	Curbs Charging Pump Cubicles	South End
PAB 36 1-2	Primary Auxiliary	Degasifier Vent Condenser Area (Evaporator Cubicle) El. 36'-0"	South and East Wall Around PAB Non-Nuclear Safety Class Charcoal Filter
PAB 36 3	Primary Auxiliary	Waste Gas Surge Drum Area El. 36'-0"	Removable Shield Wall East Wall
PAB 36 4	Primary Auxiliary	El. 36'-0"	Primary Vent Hi-Range Monitor Shield Blocks
SB 21 1-3	Service	Control Room El. 21'-0"	Control Room Entrance
SB 21 4-7	Service	Corridor Along C-Line El. 21'-0"	Between Column 4 and Elevator
SB 21 8-10	Service	Corridor Along C-Line El. 21'-0"	Between Columns 1 & 4
SB 21 11-13	Service	Aux. Boiler Room, El 21'-0"	East, West, North Walls
SB 21 14-15	Service	Elevator Enclosure El. 21'-0"	Elevator Enclosure
SB 21 17-19	Service	Control Room El. 21'-0"	Toilet, South, West and East Wall
SB 35 1-4	Service	Cable Tray Room, El. 35'-0"	Battery Room 3 and 4 Area; South, North and West Wall

Table 4.3-2 List of Block Walls Targeted for Walkdown (Continued)

Maine Yankee Wall ID No.	Building	Room Location	Wall Location
SB 35 5-6	Service	Elevator Enclosure El. 35'-0"	Enclosure Except North Wall
SB 35 7	Service	Elevator Enclosure El. 35'-0"	Elevator Enclosure East Wall - North Side
SB 35 8	Service	Cable Tray Area El. 35'-0"	East Wall
SB 39 1	Service	Vent and Air Condition Equipment El. 39'-0"	Wall Along 7-Line Between Columns E and F
SB 39 2	Service	Cable Tray Area El. 35'-0"	South Wall Along Line 7 Between Columns D and E
SB 39 3	Service	Cable Tray Area El. 35'-0"	South Wall Along Line 7 Between Columns C and D
SB 45 1-3	Service	Switchgear Room, El. 45'-6"	Battery No. 2 and 1 Area; South and West Wall and Safety-Related SWGR Room
SB 45 4-5	Service	Elevator Enclosure El. 45'-6"	Enclosure Except East Wall
SB 45 6	Service	Switchgear Area, El. 45'-6"	East Wall
SB 45 7	Service	Switchgear Area, El. 45'-6"	North Wall

Table 4.3-2 List of Block Walls Targeted for Walkdown (Continued)

Maine Yankee Wall ID No.	Building	Room Location	Wall Location
SB 61 1 SB 77 1	Service	Elevator Enclosure El. 61'-0" and El. 77'-4"	Enclosure Except South Wall
SB 61 2 SB 77 2	Service	Elevator Enclosure El. 61'-0" and El. 77'-4"	Elevator Shaft and Equipment Room - South Wall
TB 21 1	Turbine	Lube Oil Room El. 21'-0"	North Wall
TB 21 2	Turbine	Corridor Along C-Line Columns 7 to 9, El. 21'-0"	Adjacent to PCC Hx E-4A, E-4B
TB 21 3-8	Turbine	Corridor Along C-Line Columns 1 to 7, El. 21'-0"	Corridor at Service/ Turbine Building Line
TB 21 9-11	Turbine	El. 21'-0"	Walls Separating Feed-water Pumps P-2A and P-2B
TB 21 12	Turbine	El. 21'-0"	Wall Between Turbine Pedestals North Side
TB 21 13	Turbine	El. 21'-0"	East Wall (Inside) Column 9 Doorway
TB 21 14	Turbine	El. 21'-0"	East Wall (Inside) Column 10 Above Doorway
VE 21 1-4	Vent Equipment Area	El. 21'-0"	Walls Near Entrance to Containment Spray Building
Y 28	Yard	RWST	RWST Shielding Blocks

Table 4.3-3 Maine Yankee Atomic Power Plant
First Walkdown List of Equipment Components

EQUIPMENT ITEM		BUILDING AND SYSTEM	ELEVATION
TANKS			
1.	Boric Acid Storage Tank TK-2	BAT	PAB + 36'
2.	Boric Acid Mix Tank TK-3	BAT	PAB + 11'
3.	Refueling Cavity Water Storage Tank TK-4	HPSI	Yd. + 20'
4.	Primary Component Cooling Surge Tank TK-5	PCC	SB + 61'
5.	Volume Control Tank TK-6	CH	PAB + 11'
6.	Demineralized Water Storage Tank TK-21	AFW	Yd. + 20'
7.	Auxiliary Fuel Oil Supply Tank (buried) TK-28A	FO	APR + 21'
8.	Auxiliary Fuel Oil Supply Tank (buried) TK-28B	FO	APR + 21'
9.	Chemical Spray Addition Tank TK-54	CS	Yd. + 21'
10.	Secondary Component Cooling Surge Tank TK-59	SCC	SB + 70'
11.	Emergency Diesel Day Tank TK-62A	FO	AB + 21'
12.	Emergency Diesel Day Tank TK-62B	FO	AB + 21'
13.	DG-1A Compressed Air Tank TK-76A1	DG	AB + 21'
14.	DG-1A Compressed Air Tank TK-76A2	DG	AB + 21'
15.	DG-1A Compressed Air Tank TK-76A3	DG	AB + 21'
16.	Diesel Starting Air Receiver 1A TK-76A-4	DG	AB + 21'
17.	Diesel Starting Air Receiver 1A TK-76A-5	DG	AB + 21'
18.	Diesel Starting Air Receiver 1A TK-76A-6	DG	AB + 21'
19.	DG-1B Compressed Air Tank TK-76B1	DG	AB + 21'
20.	DG-1B Compressed Air Tank TK-76B2	DG	AB + 21'
21.	DG-1B Compressed Air Tank TK-76B3	DG	AB + 21'
22.	Diesel Starting Air Receiver 1B TK-76B-4	DG	AB + 21'
23.	Diesel Starting Air Receiver 1B TK-76B-5	DG	AB + 21'
24.	Diesel Starting Air Receiver 1B TK-76B-6	DG	AB + 21'
25.	Chemical Feed Tank TK-89	AFW	AF + 20'
26.	Sample Tank Tk-94	PCC	TB + 21'
27.	Chemical Additive Tank (Pipe capped off)	SCC	PAB + 21'
28.	Fuel Tank DG-2	DG	Yd + 21'
PUMPS			
1.	Fire Pump (Diesel) P-5	ASDA	FP + 20'
2.	Boric Acid Transfer Pumps P-6A	BAT	PAB + 21'
3.	Boric Acid Transfer Pumps P-6B	BAT	PAB + 21'
4.	Boric Acid Transfer Pumps P-6C	BAT	PAB + 21'
5.	Auxiliary Charging Pump P-7	CH	PAB + 11'

Table 4.3-3 Maine Yankee Atomic Power Plant

First Walkdown List of Equipment Components (Continued)

EQUIPMENT ITEM		BUILDING AND SYSTEM	ELEVATION
PUMPS (Continued)			
6.	Primary Component Cooling Pump P-9A	PCC	TB + 21'
7.	Primary Component Cooling Pump P-9B	PCC	TB + 21'
8.	Secondary Component Cooling Pump P-10A	SCC	TB + 21'
9.	Secondary Component Cooling Pump P-10B	SCC	TB + 21'
10.	Charging Pump P-14A	HPSI	PAB + 21'
11.	Charging Pump P-14B	HPSI	PAB + 21'
12.	Charging Pump P-14S	HPSI	PAB + 21'
13.	Emergency Feed Pump P-25A	AFW	AF + 21'
14.	Auxiliary Feed Pump P-25B	AFW	VA + 21'
15.	Emergency Feed Pump P-25C	AFW	AF + 21'
16.	Service Water Pump P-29A	SW	CW + 7'
17.	Service Water Pump P-29B	SW	CW + 7'
18.	Service Water Pump P-29C	SW	CW + 7'
19.	Service Water Pump P-29D	SW	CW + 7'
20.	Auxiliary Fuel Oil Transfer Pump P-33A	FO	Yd + 21'
21.	Auxiliary Fuel Oil Transfer Pump P-33B	FO	Yd + 21'
22.	Service Water Sampling Pump P-38	SW	TB + 21'
23.	Containment Spray Pump P-61 A	CS	CS + 14'
24.	Containment Spray Pump P-61 B	CS	CS + 14'
25.	Containment Spray Pump P-61 S	CS	CS + 14'
26.	Boric Acid Mix Tank Pump P-81	BAT	PAB + 11'
27.	SWS Mussel Control Pump P-86	SW	CW + 7'
28.	Chemical Feed Pump P-115	AFW	AF + 20'
29.	Service Water Seal Pit Sample Pump P-116	SW	
30.	Fuel Pumps (4 pumps total on DG skid)	DG	on Diesel
31.	Lubrication Oil Pumps (8 pumps on DG skid)	DG	on Diesel
32.	Water Pumps (4 pumps total on DG skid)	DG	on Diesel
33.	DG-2 Fuel Pump	ASDA	Yd + 21'
HEAT EXCHANGERS			
1.	Pressurizer E-2	PCC	RC + 20'
2.	Residual Heat Removal Heat Exchanger E-3A	PCC	CS + 14'
3.	Residual Heat Removal Heat Exchanger E-3B	SCC	CS + 14'
4.	Primary Component Cooling Heat Exchanger E-4A	PCC	TB + 21'

Table 4.3-3 Maine Yankee Atomic Power Plant
First Walkdown List of Equipment Components (Continued)

EQUIPMENT ITEM		BUILDING AND SYSTEM	ELEVATION
HEAT EXCHANGERS (Continued)			
5.	Primary Component Cooling Heat Exchanger E-4B	PCC	TB + 21'
6.	Secondary Component Cooling Heat Exchanger E-5A	SCC	TB + 21'
7.	Secondary Component Cooling Heat Exchanger E-5B	SCC	TB + 21'
8.	AC Electric Compressor Fan Cooler E-20	Elec	TB + 39'
9.	Seal Water Heat Exchanger E-34	PCC	PAB + 11'
10.	Diesel Generator Heat Exchanger E-82A	PCC	AB + 21'
11.	Diesel Generator Heat Exchanger E-82B	SCC	AB + 21'
12.	Oil Coolers E-86A	AFW	AF + 20'
13.	Oil Coolers E-86B	AFW	VA + 20'
14.	Oil Coolers E-86C	AFW	AF + 20'
15.	Charging Pump Seal Leakage Cooler E-92A	SCC	PAB + 11'
16.	Charging Pump Seal Leakage Cooler E-92B	PCC	PAB + 11'
MISCELLANEOUS COMPONENTS			
1.	Control Air Compressor C-10A	ASDA	VA + 21'
2.	Control Air Compressor C-10B	ASDA	VA + 21'
3.	DG-1A Starting Air Compressor C-51A	DG	AB + 21'
4.	DG-1B Starting Air Compressor C-51B	DG	AB + 21'
5.	Diesel Generator DG-1A	DG	AB + 22'
6.	Diesel Generator DG-1B	DG	AB + 22'
7.	Diesel Generator DG-2	ASDA	AB + 21'
8.	Primary Ejector EJ-2A	SPC	TB + 39'
9.	Primary Ejector EJ-2B	SPC	TB + 39'
10.	Atmospheric Steam Dump Valve Silencer S-1	SPC	VA + 40'
11.	Traveling Screen SR-1A	SW	CW + 21'
12.	Traveling Screen SR-1B	SW	CW + 21'
13.	Traveling Screen SR-1C	SW	CW + 21'
14.	Traveling Screen SR-1D	SW	CW + 21'
15.	Turbine for P-25B T-1	AFW	VA + 21'
16.	Charging Pump Seal Leakage Cooler E-92B	PCC	PAB + 11'

Table 4.3-3 Maine Yankee Atomic Power Plant
First Walkdown List of Equipment Components (Continued)

EQUIPMENT ITEM	BUILDING AND SYSTEM	ELEVATION
ELECTRICAL DISTRIBUTION SYSTEMS		
1. 4160V Emergency Bus 5	Elec	SB + 46'
2. 4160V Emergency Bus 6	Elec	SB + 46'
3. 480V Emergency Bus 7	Elec	SB + 46'
4. 480V Emergency Bus 8	Elec	SB + 46'
5. 480V Emergency Motor Control Center MCC-7A	Elec	SB + 46'
6. 480V Emergency Motor Control Center MCC-7B	Elec	RMC + 21'
7. 480V Emergency Motor Control Center MCC-7B1	Elec	CS + 20'
8. 480V Emergency Motor Control Center MCC-8A	Elec	SB + 46'
9. 480V Emergency Motor Control Center MCC-8B	Elec	RMC + 21'
10. 480V Emergency Motor Control Center MCC-8B1	Elec	CS + 20'
11. 120V AC Vital Bus 1	Elec	SB + 21'
12. 120V AC Vital Bus 1A	Elec	SB + 21'
13. 120V AC Vital Bus 2	Elec	SB + 21'
14. 120V AC Vital Bus 2A	Elec	SB + 21'
15. 120V AC Vital Bus 3	Elec	SB + 21'
16. 120V AC Vital Bus 3A	Elec	SB + 21'
17. 120V AC Vital Bus 4	Elec	SB + 21'
18. 120V AC Vital Bus 4A	Elec	SB + 21'
19. 125V DC Bus 1	Elec	SB + 46'
20. 125V DC Bus 2	Elec	SB + 46'
21. 125V DC Bus 3	Elec	SB + 46'
22. 125V DC Bus 4	Elec	SB + 46'
23. Station Battery No. 1 (Lead Antimony)	Elec	SB + 46'
24. Station Battery No. 2 (Lead Antimony)	Elec	SB + 46'
25. Station Battery No. 3 (Lead Antimony)	Elec	SB + 35'
26. Station Battery No. 4 (Lead Antimony)	Elec	SB + 35'
27. Battery Charger No. BC-1	Elec	SB + 46'
28. Battery Charger No. BC-2	Elec	SB + 46'
29. Battery Charger No. BC-3	Elec	SB + 46'
30. Battery Charger No. BC-4	Elec	SB + 46'
31. Inverter No. INVR-1	Elec	SB + 46'
32. Inverter No. INVR-2	Elec	SB + 46'
33. Inverter No. INVR-3	Elec	SB + 46'
34. Inverter No. INVR-4	Elec	SB + 46'
35. Station Service Transformer X-507	Elec	SB + 46'
36. Station Service Transformer X-608	Elec	SB + 46'

Table 4.3-3 Maine Yankee Atomic Power Plant

First Walkdown List of Equipment Components (Continued)

EQUIPMENT ITEM		BUILDING AND SYSTEM	ELEVATION
ELECTRICAL DISTRIBUTION SYSTEMS (Continued)			
37.	480V MCC-9B1 (normally off MCC-9B)	Elec	APR + 20'
38.	120V Vital Bus 7	Elec	AF + 20'
39.	125V Bus 6	Elec	APR + 20'
40.	Station Battery 6	Elec	APR + 20'
41.	Battery Charger No. 6	Elec	APR + 20'
42.	Inverter No. 7	Elec	APR + 20'
43.	Alternate Shutdown Panel	Elec	AF + 21'
44.	480V MCC 11B (mix-tank agitator)	Elec	FB + 21'
45.	480V MCC-11D (for pump P-86)	Elec	CW + 21'
46.	480V MCC 9B (mix-tank pump)	Elec	PAB + 21'
47.	Diesel Generator Control Board DG-1A	Elec	AB + 27'
48.	Diesel Generator 480V Distribution Cab 1A	Elec	AB + 22'
49.	Diesel Generator Control Board DG-1B	Elec	AB + 22'
50.	Diesel Generator 480V Distribution Cab 1B	Elec	AB + 22'
52.	120V Distribution Cabinets (Diesel Backed)	Elec	SB + 46'
53.	Main Control Board	Elec	SB + 21'
HVAC			
1.	DG-1A Room Exhaust Fan FN-20A	HV	AB + 31'
2.	DG-1B Room Exhaust Fan FN-20B	HV	AB + 31'
3.	Protected SWGR Room Supply Fan FN-31	HV	SB + 39'
4.	Protected SWGR Room Exhaust Fan FN-32	HV	SB + 55'
5.	AC Electric Compressor Fan Fan FN-33	?	TB + 39'
6.	Fan FN-61	CH	PAB + 21'
VALVES			
1.	Aux Feedwater Regulating Valve AFW-A-101 AOV	AFW	AF + 23'
2.	Aux Feedwater Regulating Valve AFW-A-201 AOV	AFW	AF + 23'
3.	Aux Feedwater Regulating Valve AFW-A-301 AOV	AFW	AF + 23'
4.	Block Valve for AFW-A-101 AFW-A-338 AOV	AFW	AF + 23'
5.	Block Valve for AFW-A-201 AFW-A-339 AOV	AFW	AF + 23'
6.	Block Valve for AFW-A-301 AFW-A-340 AOV	AFW	AF + 23'
7.	Boric Acid VCT Isolation valve BA-A-32	BAT	
8.	Boric Acid VCT Isolation valve BA-A-80	BAT	
9.	Boric Acid Flow Control valve BA-F-30	BAT	
10.	Emergency Boration Isolation valve BA-M-36	BAT	

Table 4.3-3 Maine Yankee Atomic Power Plant

First Walkdown List of Equipment Components (Continued)

EQUIPMENT ITEM	BUILDING AND SYSTEM	ELEVATION
VALVES (Continued)		
11. Emergency Boration Isolation valve BA-M-37	BAT	
12. HPSI Pump B Discharge to charging header CH-A-32	HPSI	PT + 13'
13. HPSI Pump A Discharge to charging header CH-A-33	HPSI	PT + 13'
14. VCT Discharge to HPSI Pumps CH-M-1 MOV	CH	PAB + 21'
15. VCT Discharge to HPSI Pumps CH-M-87 MOV	CH	PAB + 24'
16. Containment Spray Header Isolation Valve CS-M-1	CS	CS + 19'
17. Containment Spray Header Isolation Valve CS-M-2	CS	CS + 19'
18. Spray Chem Tank Isolation Valve CS-M-66 MOV	CS	Yd + 29'
19. Spray Chem Tank Isolation Valve CS-M-71 MOV	CS	Yd + 29'
20. CS Pump Containment Suction CS-M-91 MOV	CS	CS - 08'
21. CS Pump Containment Suction CS-M-92 MOV	CS	CS - 08'
22. HPSI Discharge to Loop 1 HSI-M-11 MOV	HPSI	PAB + 23'
23. HPSI Discharge to Loop 1 HSI-M-12 MOV	HPSI	PAB + 23'
24. HPSI Discharge to Loop 2 HSI-M-21 MOV	HPSI	PAB + 23'
25. HPSI Discharge to Loop 2 HSI-M-22 MOV	HPSI	PAB + 23'
26. HPSI Discharge to Loop 3 HSI-M-31 MOV	HPSI	PAB + 23'
27. HPSI Discharge to Loop 3 HSI-M-32 MOV	HPSI	PAB + 23
28. HPSI Discharge to SI Header HSI-M-40 MOV	HPSI	PAB + 23'
29. HPSI Pump Discharge HSI-M-41 MOV	HPSI	PAB + 23'
30. HPSI Pump Discharge HSI-M-42 MOV	HPSI	PAB + 23'
31. HPSI Discharge to SI Header HSI-M-43 MOV	HPSI	PAB + 23'
32. HPSI Suction from RWST HSI-M-50 MOV	HPSI	Yd + 21'
33. HPSI Suction from RWST HSI-M-51 MOV	HPSI	Yd + 21'
34. CS Discharge to HPSI Pump HSI-M-54 MOV	CS	CS + 19'
35. CS Discharge to HPSI Pump HSI-M-55 MOV	CS	CS + 19'
36. RWST Discharge to LPSI LSI-M-40 MOV	CS	Yd + 28'
37. RWST Discharge to LPSI LSI-M-41 MOV	CS	Yd + 28'
38. Decay Heat Release Valve MS-A-162 AOV	ASDA	VA + 43'
39. AFW Pump B turbine throttle valve MS-A-173	AFW	VA + 21'
40. Main Steam Stop Check Valve MS-M-10 MOV	MS	VA + 49'
41. Main Steam Stop Check Valve MS-M-20 MOV	MS	VA + 49'
42. Main Steam Stop Check Valve MS-M-30 MOV	MS	VA + 49'
43. Decay Heat Release MS-M-161 MOV	SPC	VA + 43'
44. Auxiliary Steam Supply Valve MS-M-255 MOV	MS	VA + 43'
45. Turbine Steam supply pressure control MS-P-168	AFW	VA + 21'
46. Steam Generator Safety Valve MS-S-12	SPC	VA + 39'
47. Steam Generator Safety Valve MS-S-13	SPC	VA + 39'

Table 4.3-3 Maine Yankee Atomic Power Plant
First Walkdown List of Equipment Components (Continued)

EQUIPMENT ITEM		BUILDING AND SYSTEM	ELEVATION
VALVES (Continued)			
48.	Steam Generator Safety Valve MS-S-14	SPC	VA + 39'
49.	Steam Generator Safety Valve MS-S-15	SPC	VA + 39'
50.	Steam Generator Safety Valve MS-S-16	SPC	VA + 39'
51.	Steam Generator Safety Valve MS-S-17	SPC	VA + 39'
52.	Steam Generator Safety Valve MS-S-22	SPC	VA + 39'
53.	Steam Generator Safety Valve MS-S-23	SPC	VA + 39'
54.	Steam Generator Safety Valve MS-S-24	SPC	VA + 39'
55.	Steam Generator Safety Valve MS-S-25	SPC	VA + 39'
56.	Steam Generator Safety Valve MS-S-26	SPC	VA + 39'
57.	Steam Generator Safety Valve MS-S-27	SPC	VA + 39'
58.	Steam Generator Safety Valve MS-S-32	SPC	VA + 39'
59.	Steam Generator Safety Valve MS-S-33	SPC	VA + 39'
60.	Steam Generator Safety Valve MS-S-34	SPC	VA + 39'
61.	Steam Generator Safety Valve MS-S-35	SPC	VA + 39'
62.	Steam Generator Safety Valve MS-S-36	SPC	VA + 39'
63.	Steam Generator Safety Valve MS-S-37	SPC	VA + 39'
64.	Return from Penetration Coolers PCC-A-216	PCC	PT + 12'
65.	Return from Penetration Coolers PCC-A-238	PCC	PT + 12'
66.	Return from RCP Coolers PCC-A-252	PCC	RC + 01'
67.	Return from RCP Coolers PCC-A-254	PCC	PT + 12'
68.	Return from CEA Air Coolers PCC-A-268	PCC	RC + 01'
69.	Return from CEA Air Coolers PCC-A-270	PCC	RC + 01'
70.	Return from Drain Cooler PCC-A-299	?	
71.	Return from Drain Cooler PCC-A-300	PCC	RC + 01'
72.	Return from Drain Cooler PCC-A-302	PCC	RC + 01'
73.	Diesel 1A Cooling Water Outlet PCC-A-493	PCC	AB + 24'
74.	PCCW Outlet from RHR Heat Exchanger PCC-M-43	PCC	CS + 02
75.	PCCW Isolation to BR & LW Coolers PCC-M-90	PCC	PAB + 11'
76.	PCCW Isolation to Letdown Heat Exchangers PCC-M-150	PCC	PAB + 21'
77.	PCCW Isolation to Containment PCC-M-219	PCC	PAB + 11'
78.	Pressurizer Safety Valve PR-S-11	SRV	RC + 65'
79.	Pressurizer Safety Valve PR-S-12	SRV	RC + 65'
80.	Pressurizer Safety Valve PR-S-13	SRV	RC + 65'
81.	Power-Operated Relief Valve PR-S-14	PORV	RC + 66'
82.	Power-Operated Relief Valve PR-S-15	PORV	RC + 66'
83.	Power-Operated Block Valve MOV PR-M-16	PORV	RC + 64'
84.	Power-Operated Block Valve MOV PR-M-17	PORV	RC + 64'
85.	Nonseismic Return Header Stop SCC-A-460	SCC	TB + 43'

Table 4.3-3 Maine Yankee Atomic Power Plant
First Walkdown List of Equipment Components (Continued)

EQUIPMENT ITEM	BUILDING AND SYSTEM	ELEVATION
VALVES (Continued)		
86. Nonseismic Return Header Stop SCC-A-461	SCC	TB + 42'
87. RCP 1 Seal Water Return MOV SL-M-29	SL	RC + 2'
88. RCP 2 Seal Water Return MOV SL-M-40	SL	RC + 2'
89. RCP 3 Seal Water Return MOV SL-M-51	SL	RC + 2'

PIPING

CABLE TRAY AND CONDUIT

INSTRUMENT RACKS

CONTROL ROOM CEILING

Legend

SYSTEM

AFW	Auxiliary Feedwater
ASDA	Alternate Shutdown Decay Heat Removal
BAT	Boric Acid Transfer
CH	Charging
CS	Containment Spray
DG	Diesel Generator Starting
FO	Fuel Oil
HPSI	High Pressure Safety Injection
HV	Area Heating and Ventilation
PCC	Primary Component Cooling
PORV	Power-Operated Relief Valve
PPC	Primary Pressure Control
SCC	Secondary Component Cooling
SL	Seal Water
SPC	Secondary Pressure Control
SRV	Safety Relief Valves
SW	Service Water

Table 4.3-3 Maine Yankee Atomic Power Plant
First Walkdown List of Equipment Components (Continued)

EQUIPMENT ITEM	BUILDING AND SYSTEM	ELEVATION

Legend (Continued

BUILDING

AB	Turbine Building Auxiliary Bay
AF	Auxiliary Feed Pumphouse
APR	Appendix R Diesel
CS	Containment Spray Pumphouse
CW	Circulation Water Pumphouse
FP	Fire Pumphouse
PAB	Primary Auxiliary Building
PT	Pipe Tunnel
PV	Purge Air Valve Room
RC	Reactor Coolant
RMC	Reactor Motor Control Center Room
SB	Service Building
TB	Turbine Building
VA	Steam and Feed Water Valve Area
YD	Yard

Piping and Valving. For margin review levels up to 0.3g pga the Panel's guidelines recommend a sample review of accessible piping systems to verify that no problems exist, such as inflexibility of piping runs between adjacent buildings. A sample piping system to be reviewed in detail was selected by mutual agreement between the system analysis and the fragility analysis teams prior to the first walkdown. Additionally, piping as encountered during the course of the equipment component walkdown was reviewed, verifying no anomalies exist.

A sample review of valves from the preliminary valve list was determined to be the most effective way to confirm the Panel's guidelines regarding valve capacities. The selection of valves to be reviewed during the walkdown was based on a study performed by Maine Yankee [Henries et al., March 1986]. The study qualified the MYC critical valves using EQE's earthquake experience data for valves. Valves not enveloped by experience data were targeted for walkdown review. A more descriptive discussion on the targeted valves occurs under Section 4.3.2.2, Walkdown Procedures for Valves.

Cable Trays. Cable tray systems throughout the plant were targeted for general survey to determine if they could be screened out generically in conformance with the review guidelines. This general survey was planned to be supplemented by a detailed inspection of a representative cable tray run, with the selection of this run to be made during the walkdown.

Instrument Racks. A sample of instrument racks, although not specifically required for review by the Expert Panel guidelines, were targeted for walkdown review. The instrument racks support critical system actuation instruments and components. Although many of the critical instrument racks are located inside containment, a sample review was conducted on similar instrument racks encountered during the course of the equipment walkdown.

Control Room Ceiling. The control room suspended ceiling system was targeted for walkdown since inspection is required by the panel's guidelines.

4.3.2 Walkdown Procedures

4.3.2.1 Walkdown Team

The fragility analysis team that conducted the plant walkdown consisted of engineers experienced in structural and equipment fragility analysis, seismic analysis and design of nuclear power plants, assessment of actual earthquake experience, and seismic margin studies, and probabilistic risk assessments. The analysis team was divided into two groups as follows:

Ravindra (EQE)	Hardy (EQE)	Swan (EQE)
Hashimoto (EQE)	Quilici (EI)	Moore (EI)
Prassinos (LLNL)	Murray (LLNL)	Griffin (EQE)

As can be observed, each group consisted of fragility analysts and system analysts. The close interaction between the two aspects of the review was considered important. During the course of the walkdown, some of the members of the two groups were switched to confirm the findings of the other group and to ensure that certain items are not missed by either of the groups. At the end of each day, the groups met to compare notes and to identify the areas and items to cover in the next day's walkdown.

4.3.2.2 Procedures for Structures and Equipment

Structures

Information necessary for seismic evaluation of civil structures is normally obtained from the design drawings rather than a walkdown. A complete set of drawings for the Maine Yankee structures was available prior to the first walkdown. These drawings were reviewed to obtain a general understanding of construction and configuration of the structures and to identify any specific data to be obtained during the walkdown.

Walkdown of the targeted Group A civil structures was performed to determine the following:

- o Verify that the structures are in general conformance with the design drawings.
- o Identify any gross deficiencies that would imply a reduction in seismic capacity.
- o Confirm that structure separations indicated on the drawings were provided.
- o Obtain structural details not available from the drawings.

The first two items above were obtained in a general manner, rather than performing a rigorous walkdown of all seismic load resisting structural members. The latter two items were specifically identified for inspection. Targeted structure separation gaps were determined before the walkdown and tabulated. Specific structural data not contained in the design drawings were obtained in the walkdown. For example, as-built sketches of weld and bolt details for certain structural steel connections were developed.

Block Walls. In preparation for the first walkdown, the following tasks were performed:

- o A target list of block walls in the plant was compiled using Maine Yankee summary tables.
- o Detailed walkdown data sheets were prepared to facilitate the compilation of block wall information.
- o Locations of block walls were highlighted on mechanical layout or architectural drawings to permit wall identification during the walkdown.

- o Information, if already available, was entered into the walkdown data sheets in advance of the walkdown. For example, wall identification numbers and locations were recorded based upon the Maine Yankee summary tables. Also included were any available sketches of seismic retrofits.

Block walls identified on the target list were inspected to the extent possible. The following information was typically obtained and recorded on the walkdown data sheets:

- o Location
- o Dimensions
- o Boundary conditions
- o Seismic retrofits, if any
- o Other physical conditions (any cracking, gaps at boundaries, openings or penetrations)
- o Identification of Group A components or lifelines directly attached or nearby

These data were supplemented by photographs and sketches.

During the walkdown, it was possible to identify several walls which obviously do not pose hazard to Group A components. Since the purpose of the walkdown was only to verify that their failure will not damage Group A components, the detailed data listed above were not recorded for them.

Equipment

In general, preparation for the first walkdown began with a review of the available equipment data. This prereview was used to accomplish two goals:

1. To obtain as much familiarity with the equipment component as possible prior to the first walkdown.

2. Identify areas where additional details were required to assess component capacity and any possible low capacity items requiring a detailed review during the walkdown.

Typical data reviewed prior to the first walkdown included a review of:

- o Plant general layout drawings determining equipment location.

- o Structural drawings determining equipment support and anchorage details.

- o If available, equipment vendor drawings or data to determine configuration, size, and material properties.

Equipment walkdown data sheets were developed for each class of equipment (reference Section 4.3.3 for examples of walkdown data sheets used). These data sheets were used as checklists to verify details identified during the data review and to record walkdown inspection notes and details. To expedite and make efficient use of the time available during the walkdown, the data sheets were filled out as completely as possible prior to the first walkdown.

The majority of the equipment components identified by the system analysts were reviewed during the first walkdown. Components located in highly radioactive areas (containment was not accessible at Maine Yankee) were not reviewed. Generically reviewed components included piping, valving, ducting, cable trays, and instrument racks. The following provides specific procedures used for the review of different classes of equipment inspected during the walkdown.

Tanks. Design drawings for the tanks and their foundations and or supports were available prior to the first walkdown. These drawings were reviewed to obtain a general understanding of the tank configurations and anchorage details. Walkdown procedures for the tanks included the following:

- o Verification that the overall tank configuration and anchorage details conform with the design drawings.

- o Review of piping and other attachments to identify any potential sources of damage due to seismic anchor point motion.

- o Inspect any unique features, which are not common to tanks, but were identified during review of the drawings.

An example of the latter item is the concrete enclosure surrounding the demineralized water storage tank (DWST). The enclosure was inspected to confirm the separation gap from the tank itself and to determine if the enclosure access room was sealed to prevent loss of tank contents to the environment in case of tank failure.

Pumps. Historical performance during past earthquakes of horizontal and vertical pumps have shown HCLPF capacities greater than 0.3g acceleration levels (reference NUREG/CR-4334). The Panel's recommendations for horizontal and vertical pumps are that for a margin review level of 0.3g, a high HCLPF capacity exists. The walkdown procedures concentrated on verifying pump and motor anchorage, type of anchorage, foundation configuration and integrity, as well as any interaction potential from attached or adjacent components. This review aimed at a confirmation of the Panel's recommendations based on our judgment as well as documentation of the pump configuration and anchorage via the walkdown data sheets and photographs.

Heat Exchangers. Walkdown procedures for heat exchangers concentrated on reviewing the supports, support saddles, anchorage details and interaction potential from attached or adjacent components. This is consistent with the Panel's recommendations for establishing heat exchanger capacity. Heat exchanger internals were not reviewed as past PRAs have established their capacity to be greater than that of their supports or anchorage. Data sheets were used to record configuration and dimensional data from the walkdown for heat exchanger support, anchorage, and attached or adjacent component interaction potential details that were not available from the plant data reviewed prior to the walkdown.

Diesel Generators. The Panel's guidelines for diesel generators recommends a review similar to that for pumps. Past performance of diesel generators demonstrates HCLPF capacity levels of at least 0.5g pga. The Maine Yankee diesel generators were reviewed for anchorage and support integrity, noting if any vibration isolators were present, and the review of the peripherals for positive anchorage. The two major peripherals reviewed during the walkdown were the engine control panel and the heat exchanger, both mounted on the diesel generator skid. Walkdown data sheets were developed and used during the walkdown to record any problem areas encountered. Photographs were also used to document items reviewed during the walkdown.

Electrical Distribution Equipment. The Panel's recommended walkdown procedures for a 0.3g pga review earthquake for electrical distribution equipment include reviewing that the cabinet or enclosures and internal instruments and components are positively anchored. Past performance of electrical distribution equipment during earthquakes suggests HCLPF capacities near 0.5g, providing the equipment and internals, instruments, breakers, contactors, etc., are positively anchored. This supports the Panel's recommendation regarding the walkdown evaluation of electrical distribution equipment. It should be noted that relays have been specifically excluded from the Maine Yankee seismic margin study.

The walkdown procedures at Maine Yankee concentrated on reviewing and collecting cabinet or enclosure anchorage details on the larger equipment items for a subsequent analytical review. The smaller (wall-mounted distribution cabinets) equipment items were reviewed for positive anchorage, but typically very few details were recorded. These smaller items were judged to have a HCLPF capacity of greater than 0.3g pga during the walkdown review. Internal component anchorage was inspected for positive attachment to the cabinet framing or cabinet walls for all electrical equipment components reviewed during the walkdown. Additionally, any interaction problems or seismic deficiencies observed were recorded. Walkdown data sheets and photographs were used to record and document the walkdown findings.

HVAC. The Panel's recommended walkdown procedures for a margin evaluation of HVAC equipment include reviewing the component for positive anchorage for a 0.3g pga review earthquake level acceleration margin level. Additionally, if a component is supported from vibration isolators, then an evaluation is required to establish lateral stability. The historical performance of HVAC equipment supports the Panel's recommendations. HVAC equipment positively anchored, as well as vibration isolator supported equipment having positive lateral restraints installed, have performed well during past earthquakes.

The procedures for the walkdown review of the Maine Yankee HVAC equipment followed the Panel's recommended guidelines. Two procedures for reviewing the HVAC equipment were used.

1. For HVAC equipment found mounted on vibration isolators, a detailed walkdown review was performed.

2. For HVAC equipment found positively anchored to a supporting structure, an engineering judgmental evaluation was performed during the walkdown review.

The review for HVAC equipment mounted on vibration isolators included recording the dimensional data and support configuration sufficient to perform an analytical evaluation after the walkdown. Comprehensive data sheets were developed to record and document details and sketches. Photographs were also used to record the walkdown findings.

The review of components in the second case included several air intake and exhaust dampers, and exhaust fans. The walkdown review assessed anchorage and any seismic deficiencies present in order to judge that the component had a HCLPF capacity greater than 0.3g pga. The predominant form of documentation for these components was the use of photographs to record the walkdown findings. Little details or notes were recorded on the walkdown data sheets.

HVAC Ducting. The Panel's guidelines state that for HVAC ducting, a HCLPF capacity exists for acceleration levels up to 0.3g pga. The walkdown procedures used to confirm the panel's guidelines consisted of two approaches:

1. Inspect the ducting in close proximity to the HVAC equipment components to be reviewed.

2. Inspect a sample ducting system selected during the walkdown.

Inspection items included reviewing the vertical- and lateral-load-resisting members of the ducting system and any possible anchor point displacements that could impart significant loads to connecting equipment. Documentation consisted of noting any anomalies and taking several photographs.

Valves. For review earthquake levels up to 0.3g pga the Panel's recommended walkdown procedures require a review of valves encountered during the sample piping system walkdown. Areas of concern to be reviewed during the walkdown include observing for interaction potential between the valve operator and adjacent structure or component and reviewing possible anchor point displacements between piping and valve. Historical performance of valves during earthquakes at this acceleration level supports the Panel's walkdown review recommendations.

In addition to reviewing valves during the piping walkdown, a sample list of valves targeted for review was developed. The method used to select a valve for review was determined after a prior review of a Maine Yankee (MYC) document [Henries et al., March 1986], which evaluated the seismically critical valves using seismic experience data. The document recorded items such as valve description and function, height above grade, cast iron body or yoke, pipe diameter, type of operator, actuator weight, distance between the pipe center line and the top of the operator, and the evaluation conclusions. The margin study valve list was reviewed against the MYC document. The selected valves included those valves not previously shown to be within the bounds of the experience data or were not included in the document list.

The review of the targeted valves was conducted during the course of two walkdowns. The review procedures consisted of inspecting each valve on the list accessible to the walkdown teams. Seven valves targeted for review were located in containment and could not be reviewed during the walkdown. For these valves, vendor literature and any previous photographs taken by Maine Yankee were requested for review to confirm the Panel's recommendations. Walkdown procedures consisted predominantly of inspecting the valve and area around it, noting any potential interaction or anchor point displacement problems. Documentation consisted of taking several photographs and in some instances recording specific valve data on walkdown data sheets. This was limited to valves not included in the Maine Yankee document, because those valves listed had this information tabulated.

Piping. Past seismic PRA studies and earthquake experience data have shown that welded steel piping systems have a very high resistance to seismic loads. NUREG/CR-4334 contains the Expert Panel recommendation that "piping systems in nuclear power plants have HCLPF capacities greater than 0.5g pga." The panel recommends that for a 0.3g pga review level earthquake, a walkdown of a sample piping system should be conducted and that piping between buildings should be inspected.

Specific areas associated with the piping which were reviewed during the plant walkdown for Maine Yankee were:

- o Walkdown of the auxiliary feedwater system (AFW)
- o Assessment of piping which spans between two buildings

Underground piping systems could obviously not be addressed on the plant walkdown, and were evaluated based on their design drawings. Two other piping failure modes that were addressed during the walkdown included the impacting failures of valve operators and the damage of piping caused by the failure of anchorage of attached equipment. The valve clearance issue and the equipment anchorage issue are addressed as a part of the margin evaluation for the specific equipment component and not as a part of the piping margin review.

The sample system to be walked down for the seismic margins review was the AFW system. The AFW system was chosen based on its importance to the safety of the plant and because of the variety of piping sizes, supports, and components (branches, elbows, reducers, tee connections, etc.) in the system. The procedure for walking down the piping system included following the piping layout drawings to verify support locations, assessing system interaction potential to the piping, and taking detailed configuration information for piping details that are judged to be of potential concern.

Cable Trays. In the first walkdown, inspection of the cable trays was performed at two levels:

- o General survey of cable tray systems in the plant
- o Detailed inspection of a representative cable tray run

The general survey was performed to obtain an overview of cable tray construction throughout the plant. This included a review of the variety of cable tray system layouts, support configurations, and construction details. The inspection also considered items identified in the review guidelines as being of potential concern, including failure of taut cables due to large relative displacement, severing of cables caused by sharp edges at the ends of cable trays, and weld failure.

As a part of the walkdown of cable tray systems, a representative run was inspected in detail to obtain specific information on the construction. This run, located in the cable spreading room, was selected since it exhibited many of the features common to cable trays throughout the plant. Information was documented on walkdown data sheets. Information collected included run location and layout supplemented with sketches, support configuration details and spacings, cable tray configuration and loading, tray and support connection details, interfaces with the structure and other components, and any potential problems.

Instrument Racks. Instrument racks were not specifically addressed by the Panel in establishing its margin review guidelines. Historical performance during earthquakes of a variety of instrument rack configurations documented in the experience data base suggests that a HCLPF capacity of greater than 0.3g pga exists. Walkdown procedures of the Maine Yankee instrument racks consisted of reviewing a sample of racks encountered during the course of the equipment walkdown. The reviewed racks were inspected for positive anchorage and similarity to documented data base racks. Attached instruments and components were also inspected for positive anchorage to the rack. Walkdown documentation consisted of taking several photographs.

Several of the most critical instrument racks at Maine Yankee are located inside the containment building and could not be reviewed during either the first or the second walkdown. Yankee Atomic engineers have photographic records of these instrument rack installations which were taken on previous plant outages. These photographs were used as a basis for evaluating the seismic margin inherent in the instrument racks located in containment and the components (transmitters, transducers, sensors, etc.) supported on these racks. The impulse lines and electrical leads which enter and exit these instrument racks were assessed by the systems analysts on the basis of multiple trains being separated from one another.

Control Room Ceiling. Sketches provided prior to the first walkdown illustrated design modifications incorporated in the suspended ceiling system over the control room. These modifications consisted of safety wiring the T-bar ceiling and light fixtures to the concrete slab overhead. The ceiling system was inspected during the walkdown to verify that the safety wiring was installed consistent with the sketches and appeared adequate. In accordance with the Panel's guidelines, other fixtures above the control room were inspected to identify any other potential hazards to personnel and equipment below.

4.3.3 Walkdown Documentation

Walkdown documentation for equipment and structures consisted of recording the findings using walkdown data sheets and photographs. The walkdown data sheets were developed for each particular class of component indicating specific information required to confirm and verify the Panel's recommendations as well as to record details sufficient to perform a seismic fragility evaluation if necessary. Typical examples of equipment data sheets used during the Maine Yankee walkdowns for equipment are presented in Tables 4.3-4, 4.3-5, and 4.3-6 for pumps, HVAC components and block walls, respectively. The data sheets reflect the varying levels of information required between different classes of equipment depending on their seismic ruggedness (e.g. pumps require little review other than to verify anchorage and interaction potential whereas HVAC components supported by vibration isolators require a detailed review recording greater degrees of information necessary for a fragility evaluation).

Photographs were also used to record details of the walkdown. Photographs provide a permanent record of what was reviewed and support any notes or details taken during the walkdown. System interaction concerns are typically documented with photographs. Additionally, photographs are used in the fragility evaluation to confirm details taken or to provide additional clarification. Photographs are a valuable part of the complete walkdown documentation.

4.3.4 Walkdown Results

4.3.4.1 Walkdown Findings

Structures

The original seismic design criteria for the Maine Yankee civil structures are described in Section 3.2. With the exception of the 10 CFR 50, Appendix R diesel room, and the turbine/service building, all of the Group A structures listed in Section 4.3.1 were categorized as Class 1 structures by the original design basis. However, the following areas within the turbine/service building were considered Class 1: control room, cable room, switchgear room, service building area housing the control room air conditioning, breathing air, and switchgear room ventilating equipment, diesel generator enclosure, and turbine building portion housing the component cooling heat exchangers, pumps and air compressor receivers. The Appendix R diesel system was subsequently deleted from the Group A systems by the systems analyst.

Seismic load-resisting systems for the Group A civil structures are composed of reinforced concrete and/or structural steel. HCLPF capacities were not established by the Expert Panel for steel frame structures. The following steel frame structures are therefore screened in and require a seismic capacity evaluation:

- Circulating water pumphouse, portion above El. 21'-0"
- Turbine/service building, steel framed portions
- Main steam valve house, interior steel structure

During the first walkdown, one of the diagonal braces for the mainsteam valve house steel structure was found to be missing. This was considered in the evaluation of the structure HCLPF capacity.

The Expert Panel concluded that concrete containments, concrete shear walls, diaphragms, and footings, special nonductile details, and impact between buildings

TABLE 4.3-4

EXAMPLE PUMP WALKDOWN DATA SHEET

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 1/

JOB NO. 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ______ DATE ______

FACILITY: Maine Yankee Atomic Power Station

COMPONENT: ______

LOCATION: Building = ______
Elevation = ______

1.0 COMPONENT DATA:

Plant ID Number = ______
Manufacturer = ______
Model = ______
Function = ______

Photograph (overall) Roll No. ______ Frame No.s ______

2.0 AREAS REQUIRING DETAILED REVIEW:

2.1 Anchorage: Number and size of anchor bolts = ______
Type of anchor bolts = ______
Description of foundation = ______

Photograph Roll No. ______ Frame No.s ______

Note: Provide a sketch of anchorage plan with dimensions and indicate any foundation deficiencies observed in space provided below.

TABLE 4.3-4 (CONTINUED)

EXAMPLE PUMP WALKDOWN DATA SHEET

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 2/

JOB NO 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ______ DATE ______

3.0 ADDITIONAL COMMENTS OR OBSERVATIONS:

Note any system interactions.

TABLE 4.3-5

EXAMPLE HVAC COMPONENT WALKDOWN DATA SHEET

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 1/____

JOB NO 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ______ DATE ______

FACILITY: Maine Yankee Atomic Power Station

COMPONENT: ____________________

LOCATION: Building = ____________________
Elevation = ____________________

1.0 COMPONENT DATA:

Plant ID Number = ____________________
Manufacturer = ____________________
Model = ____________________
Function = ____________________

Photograph (overall) Roll No. ________ Frame No.s ________

2.0 AREAS REQUIRING DETAILED REVIEW:

2.1 Housing: Overall dimensions = ____________________
Anchorage: Type of anchorage (vibration isolators?) = ____________________
Lateral restraints on isolators = ____________________
Number and size of anchor bolts = ____________________
Type of anchor bolts = ____________________
Description of foundation/supt. = ____________________

Photograph Roll No. ________ Frame No.s ________

Note: Provide a sketch of anchorage plan with dimensions and indicate any foundation/support deficiencies observed in space provided below. If vibration isolated sketch lateral restraint.

TABLE 4.3-5 (CONTINUED)

EXAMPLE HVAC COMPONENT WALKDOWN DATA SHEET

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. 2/

JOB NO. 8227-09 JOB LLNL Seismic Margin Review BY ______ DATE ______

CLIENT LLNL SUBJECT Plant Walkdown Data Sheet CHK'D ______ DATE ______

3.0 ADDITIONAL COMMENTS OR OBSERVATIONS:

Note any system interactions.

TABLE 4.3-6

EXAMPLE BLOCK WALL DATA SHEET

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. ______

JOB NO 2227-09 JOB Seismic Margins BY ______ DATE ______

CLIENT LLNL SUBJECT Walkdown Data CHK'D ______ DATE ______

BLOCK WALL DATA SHEET

Wall ID Number:

Building:

Floor:

Location:

Ref. Drawing Number:

Film Roll Number: Frame Numbers:

Wall Dimensions: W x H x T

Lateral Supports:

Any Visible Cracking?

Any Gaps At Boundaries?

Any Openings and/or Penetrations?

Group A Components Directly Attached:

Group A Components Nearby:

Supported or Adjacent Lifelines:

Group A Components or Lifelines Likely to Be Damaged By Wall Failure:

TABLE 4.3-6 (CONTINUED)

EXAMPLE BLOCK WALL DATA SHEET

EQE

ENGINEERING, PLANNING AND MANAGEMENT CONSULTANTS

SHEET NO. ______

JOB NO 8227-09 JOB Seismic Margins ______ BY ______ DATE ______

CLIENT LLNL SUBJECT Walkdown Data ______ CHK'D ______ DATE ______

BLOCK WALL DATA SHEET

Wall ID Number:

PLAN

ELEVATION

should have HCLPF capacities greater than 0.3g. The Group A concrete structures were reviewed to confirm that this conclusion is appropriate for Maine Yankee.

The containment structure consists of a base mat, a cylindrical shell, and a hemispherical dome. The wall is reinforced in a two-way pattern. The containment structure is comparable to other reinforced concrete containments evaluated in past fragility evaluations. In addition, HCLPF capacities have already been developed in [Hashimoto et al., 1984] using approaches essentially the same as the fragility analysis and CDFM methods. Even with the concurrent effects of the design incident internal pressure, the HCLPF capacity was found to be equal to or greater than 1.0g.

The other reinforced concrete structures are typically composed of integral walls and slabs. A review of the design drawings was performed to verify that they are adequate to resist the review level earthquake. This review considered the ability of the shear walls to transmit the overall seismic loads into the foundation, the availability of local load paths to deliver inertial loads to the walls, and the presence of any nonductile detailing. The concrete structures were found to be comparable to other structures analyzed in past fragility evaluations. No inherent weaknesses in the seismic load-resisting capabilities were noted. The Group A concrete structures were concluded to have HCLPF capacities greater than 0.3g and were therefore screened out.

A review of the structural drawings indicates that the buildings are typically separated by gaps of three inches or more. Readily accessible building separations were inspected during the first walkdown. The walkdown confirmed that these separations are present. Three-inch separation is greater than gaps provided for most nuclear plant structures. Since the Maine Yankee structures are typically of shear wall construction and founded on rock, lateral seismic displacements will be small. It is concluded that impact between buildings is very unlikely at the review earthquake level of 0.3g.

Block Walls. The Maine Yankee block walls are typically unreinforced. A seismic evaluation of the block walls was conducted in response to I & E Bulletin 80-11. As a result of this evaluation, seismic retrofits were installed to increase the resistance of certain block walls supporting or adjacent to safety-related equipment.

The review guidelines specify that a margin evaluation is required for these types of block walls. However, an evaluation should not be necessary for all walls in the plant. Walls that can be screened out include those that are not supporting or adjacent to Group A components, and those that can collapse but not cause damage to adjacent Group A components.

To identify the subset of block walls requiring a HCLPF calculation, the block wall evaluation procedure in Figure 4.3-1 was developed. Prior to the walkdown, block walls in the plant were identified and located. Based upon the walkdown, the walls were screened in or out depending on their potential hazard to Group A components. All walls supporting attached Group A components were screened in. For a wall adjacent to (i.e., within about one wall height) Group A components, an assessment was made of the likelihood of damage to the components should the

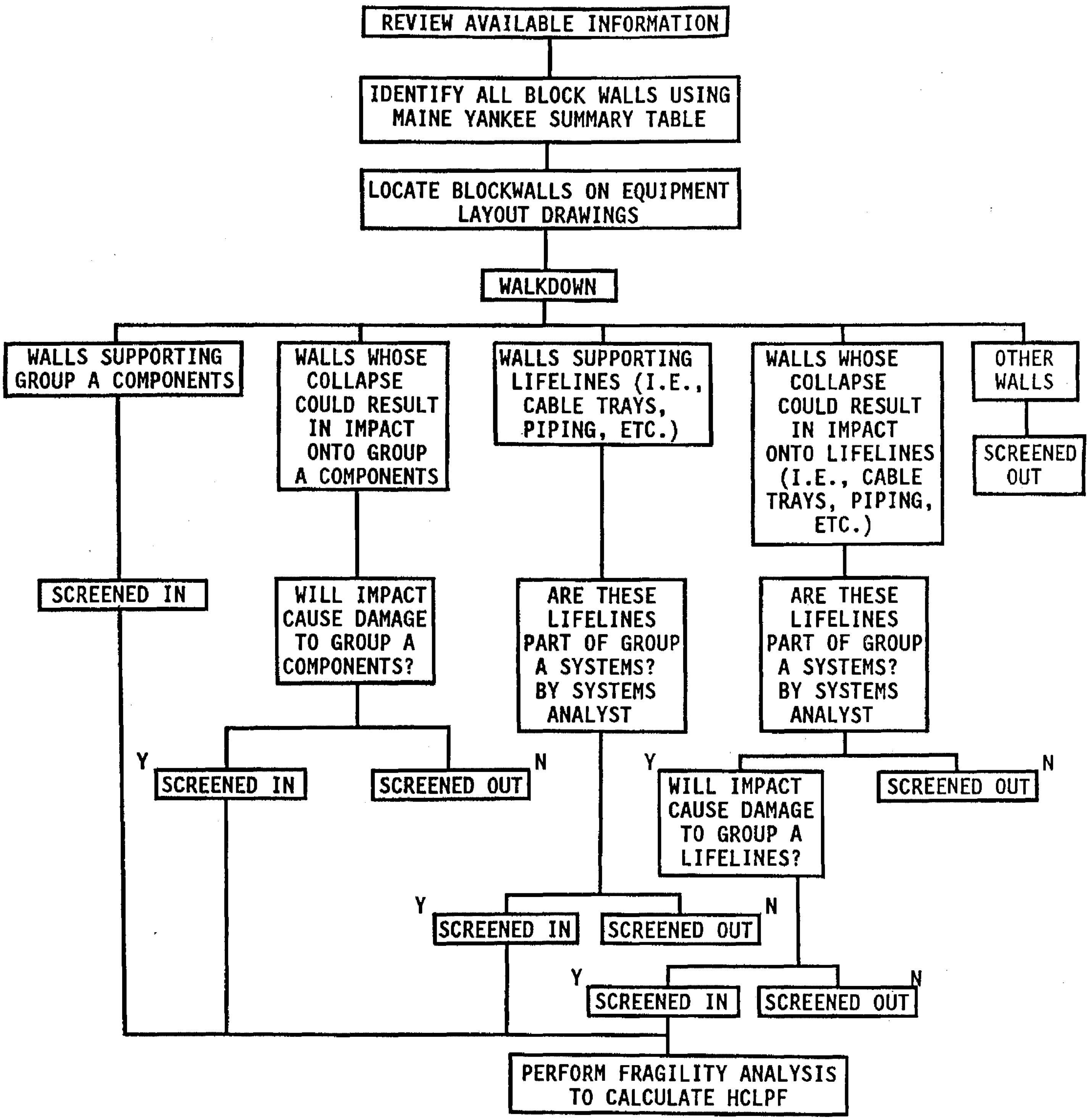


Figure 4.3-1 Block Wall Evaluation Procedure

wall collapse. Examples of walls screened out at this point include walls shielded from adjacent components by built-up steel framing and walls adjacent to components judged to be capable of withstanding the impact without damage. Block walls affecting lifelines such as piping and cable trays were screened in the same manner. If necessary, confirmation of the Group A status of these lifelines was provided by the systems analyst. All other block walls not supporting or adjacent to Group A components were screened out. Block walls affecting safety related equipment that were not walked down are assumed to have seismic capacities comparable to the Group A walls. Review of the block walls inside of containment was conducted on the basis of architectural and design modification drawings.

Dams, Dikes, and Levees. Walkdown of the fire water pond indicated that grading around the pond would cause the water to flow away from the plant even if the surrounding dike should fail during an earthquake. The dike was therefore screened out since its failure would not have any impact on plant safety systems.

Equipment

Tanks. Except for the buried diesel fuel oil storage tanks, TK-28A and B, all the tanks identified on the Maine Yankee margin review equipment list required a walkdown review and capacity evaluation. The tanks reviewed during the walkdown were all found to be anchored (i.e., no unanchored tanks were observed). The majority of tanks observed were consistent with the design drawings and details reviewed prior to the walkdown. Few anomalies affecting tank capacity were found during the course of the walkdown. The results of specific tank findings are discussed in greater detail below.

The auxiliary fuel oil supply tanks, TK-28A and B, were reviewed during the walkdown. The fuel oil supply tanks are buried with only the upper portion of the tank visible; consequently, only the visible connecting piping was reviewed for possible failure caused by large relative displacements of the ground surrounding the buried tanks. The tank fill lines were the only observable piping. The fill lines are attached to the top of the tank; consequently, if a piping failure occurred a loss of diesel oil would not result.

The emergency diesel day tanks, TK-62A and B, were reviewed in detail during the walkdown. The review found the tank elevated and welded to a braced steel frame. The support frame was bolted to the concrete floor with two expansion anchor bolts per column. Under each column was observed to be a shim plate and approximately one inch of grout. This installation detail varied between the four columns due to the sloping floor with the distance between the top of concrete and bottom of column base plate varying from 1.75 in to 2.5 in. This observation brought up the concern of how much of the anchor bolt was embedded in the concrete floor. To verify bolt embedment, Maine Yankee plant personnel ultrasonic tested (UT) each of the anchor bolts. The UT inspection confirmed that the bolt embedment did not extend into the concrete floor. Maine Yankee subsequently modified the day tank anchorage by installing new anchor bolts which are fully embedded into the concrete floor slab (Section 4.4 describes the modification).

Pumps. The critical pumps identified on the margin equipment list were reviewed during the walkdown. For a 0.3g pga review level earthquake the Panel requires a review of the pump anchorage and any potential interaction problems. Horizontal pumps at Maine Yankee were determined to be well anchored on the basis of the first walkdown. Piping anchor point displacement and interaction potential concerns were reviewed for each pump with no areas of concern observed. Maine Yankee horizontal pumps were subsequently screened out with a HCLPF capacity judged to be greater than 0.3g pga.

The Expert Panel guidelines recommend a capacity evaluation for vertical pumps if the shaft length is cantilevered greater than 20 ft. There were three sets of vertical pumps identified on the Maine Yankee margin review equipment list:

- o Service Water Pumps
- o Containment Spray Pumps
- o Auxiliary Fuel Oil Transfer Pumps

The Maine Yankee service water pumps were identified from the initial drawing and vendor data review as vertical pumps with long cantilevered shafts greater than 20 ft in length. The service water pumps shaft base support could not be reviewed during the plant walkdown as required by the Panel's recommendations; however, the vendor drawings identified the pumps to have 26 ft shaft lengths with pin supports at the base. Consequently, with respect to this issue, the service water pumps were screened out per the Panel's guidelines.

The two other sets of vertical pumps identified on the equipment list for review were observed to have shaft lengths less than 20 ft in length. These pumps were subsequently screened out per the Panel's guidelines.

Two vertical pumps (service water and containment spray pumps) were observed during the walkdown to have a possible weak connection between the motor and pump interface due to a small number of bolts and small bolt circle dimension. Dimensional data and motor name plate data was recorded in order perform a capacity calculation for these pumps.

Heat Exchangers. All heat exchangers identified on the margin review equipment list were reviewed during the walkdowns except for the reactor containment air recirculation coolers, the reactor coolant regenerative heat exchanger, the seal water heat exchanger and seal water heater due to inaccessibility (high radiation areas). Per the Panel's guidelines the exchangers were reviewed for support and anchorage integrity as well as any potential interaction or seismic anchor point displacement problems. The supports and anchorage require an evaluation to show capacity greater then 0.3g pga. The following briefly describes the types of heat exchangers reviewed at Maine Yankee and the walkdown findings.

The heat exchangers reviewed at Maine Yankee can be classified into four categories.

1. Vertically oriented heat exchangers.

2. Horizontally oriented heat exchangers supported from a concrete pier or steel support frame.

3. Small heat exchangers mounted directly to a larger supporting component.

4. Heat exchangers not accessible during the walkdown due to high local radiation levels.

The only vertical heat exchangers reviewed were the residual heat removal (RHR) heat exchangers which were observed to be supported from anchor lugs at approximately the midpoint of the exchanger shell. Additionally, bottom lateral supports in both horizontal directions were observed. An analytical evaluation of the supports and anchorage was performed.

The standard horizontal heat exchangers were all observed to have an adequate support system (concrete pier or braced steel frame). All were observed to be well anchored with no anomalies noticed. An analytical evaluation of the supports and anchorage was performed.

The small skid-mounted heat exchangers included as a peripheral component of the component to which it was attached. These heat exchangers were observed to typically provide a bearing cooling function, and varied from small diameter (approximately 5 to 6 inches) cylindrical units to just helical coils attached to the side of the pump casing. Capacity for these exchangers was judged to be greater than 0.3g pga during the walkdown.

The heat exchangers not accessible for the walkdown were evaluated via a drawing and vendor data review with the support from photographs previously taken by Maine Yankee engineers. The reactor containment air recirculation coolers were the only component that fell into this category requiring an evaluation. The reactor coolant regenerative heat exchanger, the seal water heat exchanger, and seal water heater did not require a review, because valves PCC-M-90 and PCC-M-219 have a HCLPF capacity greater than 0.3g pga, which will isolate the PCC system if these components fail.

Diesel Generators. The walkdown findings verified the Panel's recommendations that a high capacity exists for diesel generators at the 0.3g pga review level earthquake . The diesel generator skid assembly rests on a large grout pad which is used for leveling purposes. The anchors for the skid consist of large "J" bolts embedded well into the concrete foundation. These 1 1/4-inch-diameter "J" bolts are judged to have a very high seismic capacity and the effect of these anchors passing through the grout pad is felt to be minimal. The diesel generator peripheral components which were mounted on the skid were reviewed for structural integrity and observed to be well anchored. No potential interaction problems were observed from surrounding components. The diesel generators and peripheral components were subsequently screened out.

Electrical Distribution Equipment. In general the electrical equipment reviewed for the margins program were observed consistent with the Panel's guidelines stating that active electrical equipment can survive ground accelerations up to 0.5g pga, provided that the cabinets and instruments are anchored. The equipment reviewed at Maine Yankee was well anchored except for one anomaly that occurred with the station service transformers which is described below. Most of the critical electrical equipment at Maine Yankee had seismic anchorage upgrades installed during one of several previous seismic evaluations conducted by the Utility. Upgraded anchorage and supports were evident on the switchgear, motor control centers, inverters, distribution cabinets, main control board, and battery chargers. The instruments and internal components of most electrical cabinetry were all observed to be well anchored. However, one noncritical component in the main control board was observed unanchored. This component has been subsequently anchored by Maine Yankee (component reviewed during the second walkdown was observed positively anchored). The following discusses several walkdown findings with regard to particular types of electrical equipment reviewed at Maine Yankee.

The station battery cells at Maine Yankee were observed to be of lead-antimony flat-plate construction. Data on the performance of lead-antimony batteries during earthquakes is lacking at the present time. However, extreme corrosion degraded electrical performance has been recorded for aged cells such that a HCLPF capacity cannot be confidently established. The Expert Panel (NUREG/CR-4334) suggests that battery cells are not vulnerable at a 0.3g earthquake and can be screened out. EQE feels that this recommendation has not been proven to be reasonable for all emergency batteries in nuclear power plants. Maine Yankee batteries proved to be a case where further examination/review was necessary. Section 4.4, Maine Yankee Component Modifications, discusses this finding further.

The inverters were one of the components which had an upgraded anchorage from its originally installed condition. The anchorage modifications were somewhat unorthodox due to adjacent equipment constraining the placement of the anchorage addition. Details were recorded and an anchorage capacity evaluation was performed. Additionally, the back shear panel of the cabinet had ventilated cutouts which raised some concern about the lateral load resistance of the cabinet frame. After the front doors were opened it was observed that the transformer was bolted to the bottom base framing with few component attachments occurring above the base framing level.

The station service transformers enclosure framing was observed to be well anchored and also had an anchorage upgrade installed. However, when the internal core/coil assembly was reviewed it was found to be unanchored. The core/coil assembly was observed to be supported by rubber isolators in four places with no vertical uplift or lateral restraints installed. The transformer internal core/coil assembly has been subsequently identified as requiring additional anchorage to be installed during the next refueling outage (March 1987). Section 4.4, Maine Yankee Component Modifications, describes the modifications which are scheduled to be installed.

Numerous wall-mounted panels were reviewed during the course of the walkdown. All of these panels were observed to be light in weight and well anchored with a minimum of four bolts per panel, one at each corner. The HCLPF capacities for the wall mounted panels were determined to be greater than 0.3g pga review level earthquake using the methods described under Section 5.2, Simplified Analysis and Use of Screening Tools.

<u>HVAC Components</u>. The walkdown review consisted of inspecting several fan units, air conditioner units and dampers. Except for the diesel generator exhaust fans, all the fan units and air conditioner units were observed to be supported from vibration isolators; consequently, per the Panel's guidelines the fans and air conditioners required an evaluation to establish lateral stability. An initial evaluation determined the lateral load capacity to be deficient for both items and Maine Yankee subsequently scheduled both items to be upgraded during the March 1987 refueling outage. Section 4.4 Maine Yankee Component Modifications describes these modifications.

The diesel generator exhaust fans and the diesel generator air intake and exhaust dampers were observed to be well anchored into the building walls. A capacity evaluation based on the walkdown inspection judged the fans and dampers to have a HCLPF capacity greater than 0.3g pga, thus were subsequently screened out.

Overhead HVAC ducting in the Maine Yankee plant was observed to be well supported with lateral bracing in both horizontal directions. The vertical and lateral supports were found to be threaded rods anchored into threaded inserts in the concrete ceiling and adjacent walls. The ducting reviewed at each component was observed to have flexible joints at the interface connections. Consequently, lateral movement of the ducting will not impart significant loads to the connected component. Maine Yankee ducting was judged to have a HCLPF capacity greater than 0.3g pga, thus was subsequently screened out.

<u>Valves</u>. The sample list of valves reviewed during the walkdown were all observed to be seismically adequate such that a high confidence exits that the valves addressed in the Maine Yankee margin study will survive a 0.3g pga margin earthquake. These findings agree with the Panel's guidelines on valve performance in a 0.3g pga margin earthquake. The walkdown findings of the sample list of valves is discussed below for specific types of valves reviewed.

The power operated relief valves (PORV's) could not be reviewed during the walkdown due to inaccessibility (high local radiation). This included several other valves as well (Reference Table 7.1-3 in Chapter 7 for the capacity evaluation of the valves). However, vendor data and photographs were provided by Maine Yankee to aid in EQE's review of the valves. The vendor data illustrated the construction and operation of the PORV's. The function is essentially a solenoid operator cantilevered from the valve body, not unlike a standard MOV, actuating the valve upon the presence of an operation signal. The Maine Yankee PORV is quite similar to the PORV's located at the El Centro Steam Plant which was subjected to the 1979 Imperial Valley earthquake. Figures 4.3-2 and 4.3-3 show a comparison of the Maine Yankee PORV's and those from the El Centro Steam Plant.

There were several valves reviewed which had long "reach rods" between the operator and the valve body. The configuration of these valves is such that operators can operate the valves from a safe distance and not be exposed to dangerous levels of radiation. The valve operator was well anchored to a pedestal

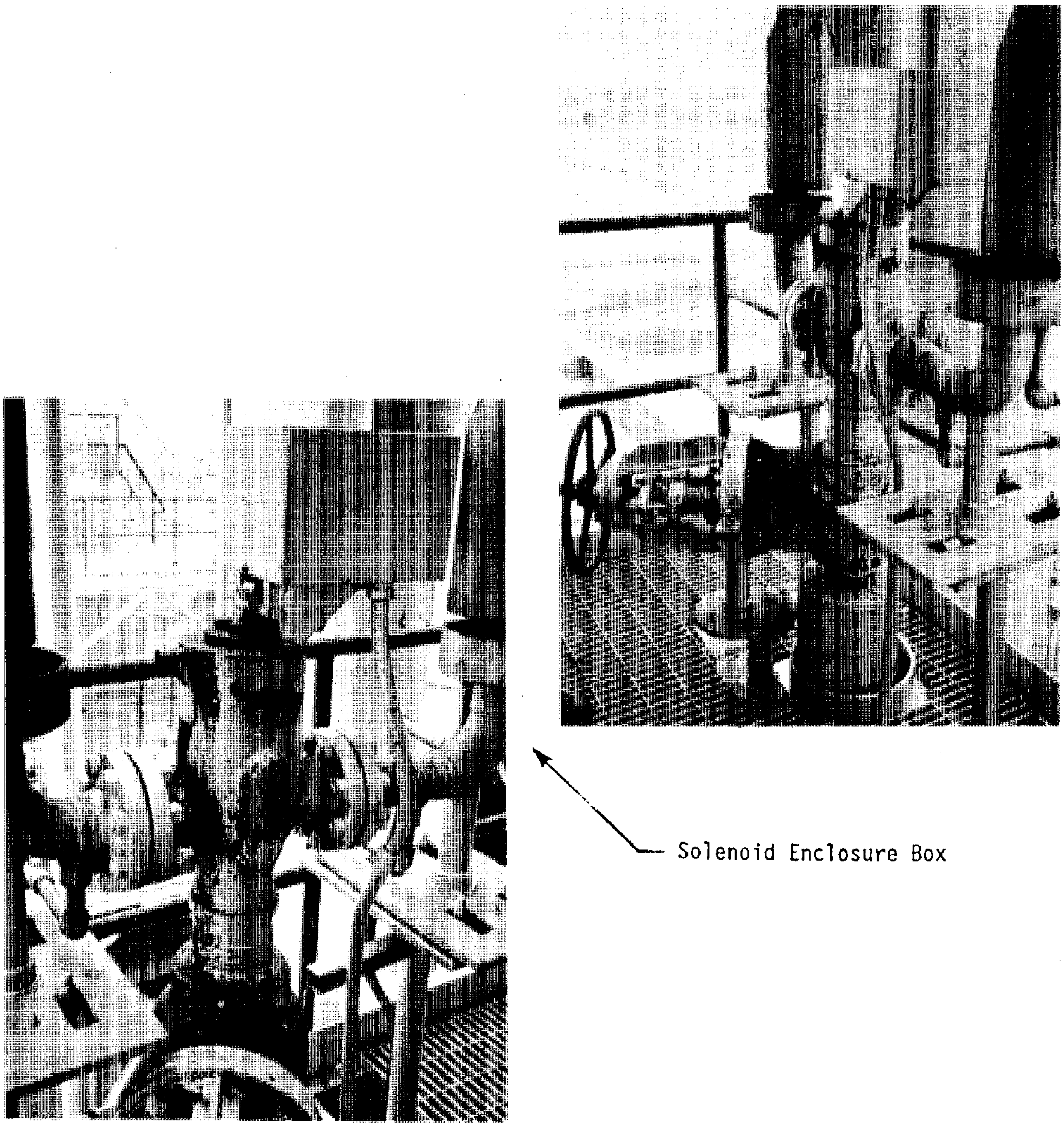


Figure 4.3-2 Example of a PORV at the El Centro Steam Plant.

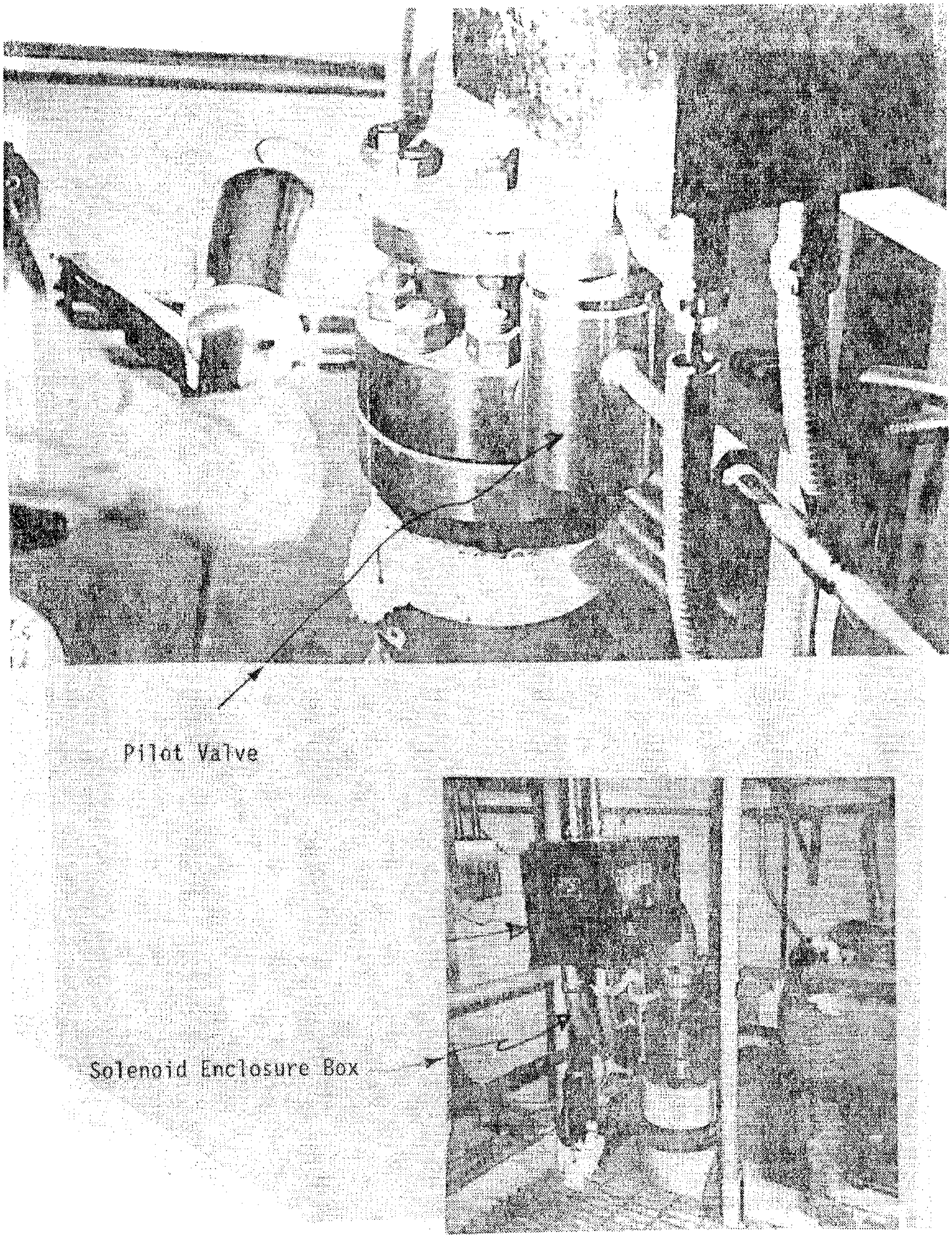


Figure 4.3-3 Maine Yankee PORV PR-S-14 and PR-S-15.

while the operator rod was connected to the operator with a universal joint. The operator rod was then routed through the floor supported at regular spacings before connecting to the valve body. At each support point and at the valve body was a universal joint. From the walkdown review it was judged that these valves have a HCLPF capacity greater than the 0.3g pga review level earthquake. Earthquake experience data supports our judgment on these "reach rod" valves. Valves similar to the long "reach rod" valves were found at several Coalinga area sites subjected to the 1983 earthquake. Figures 4.3-4 and 4.3-5 show a comparison of the Maine Yankee valves and a sample of those from the Coalinga sites.

Several Maine Yankee valves fell out of the bounds of the earthquake experience data based on conservative estimates of operator weight and operator height. These valves were reviewed and found to be reasonably close to the experience data bounds. Seismic anchor point displacements and interaction potential were assessed for all valves addressed on the walkdown. No problems were observed and the valves subsequently judged to have a HCLPF capacity of greater than 0.3g pga. Table 7-3 in Chapter 7 identifies the Maine Yankee valves and also indicates which valves were judged to have a HCLPF capacity greater than 0.3g pga compared to those where experience data supported the evaluation.

Piping. The auxiliary feed water system (AFW) was the sample piping system selected for a detailed walkdown review. The AFW extends from the Containment penetration to the demineralized water storage tank.

Three Maine Yankee drawings were used (verification of support types, support spacing, unsupported spans, etc.) during the walkdown to aid in the walkdown:

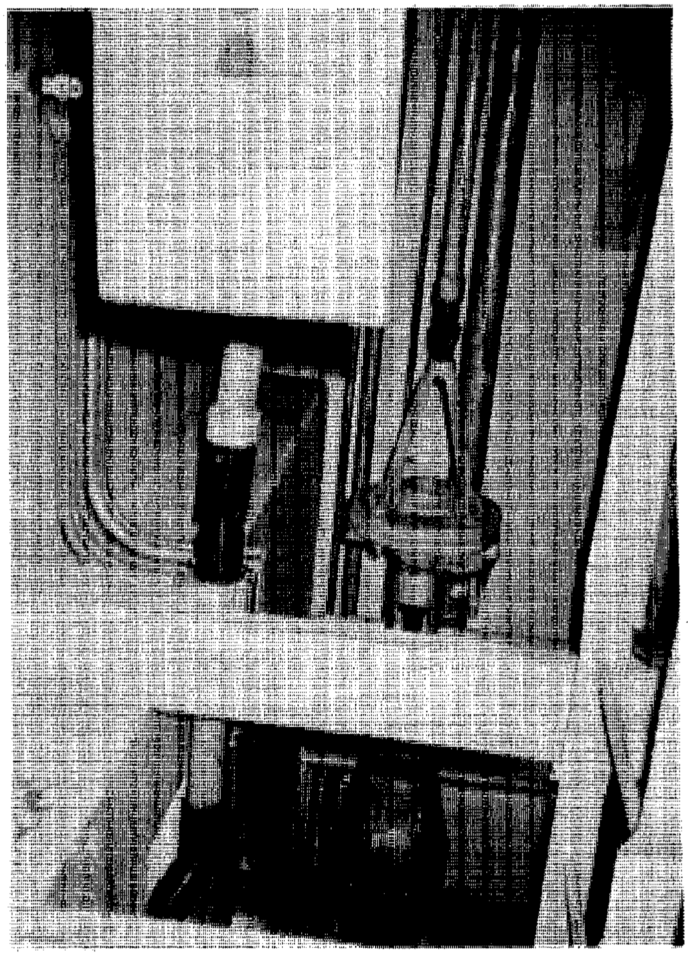
- o Drawing 12365.10-MKS-103RI-4 shows the piping from the containment penetration to anchor H-102

- o Drawing 12365.10-MKS-103N1-4 shows the piping from anchor H-102 to the discharge side of the emergency feedwater pump (P-25A)

- o Drawing 12365.10-MKS-103N1-4 shows the piping from the suction side of Pump 25A to the point where the piping goes underground to the DWST.

The walkdown of the AFW piping produced nothing which altered the Panel's assessment of a high confidence exits that the piping will survive a 0.3g pga review level earthquake. The piping was observed to be of all welded steel construction with standard fittings (tees, elbows, and branch connections). The supports were typically welded steel frames constructed of angles or boxbeams and attached to the walls/ceiling/ floors by welding to embedded or structural steel, and on occasion with expansion anchors. Interaction with surrounding components was assessed throughout the AFW piping system, but no possible damage scenarios were found.

Piping systems which span between two structures were also found to have high capacities. Maine Yankee is a rock site, thus very little relative motion between



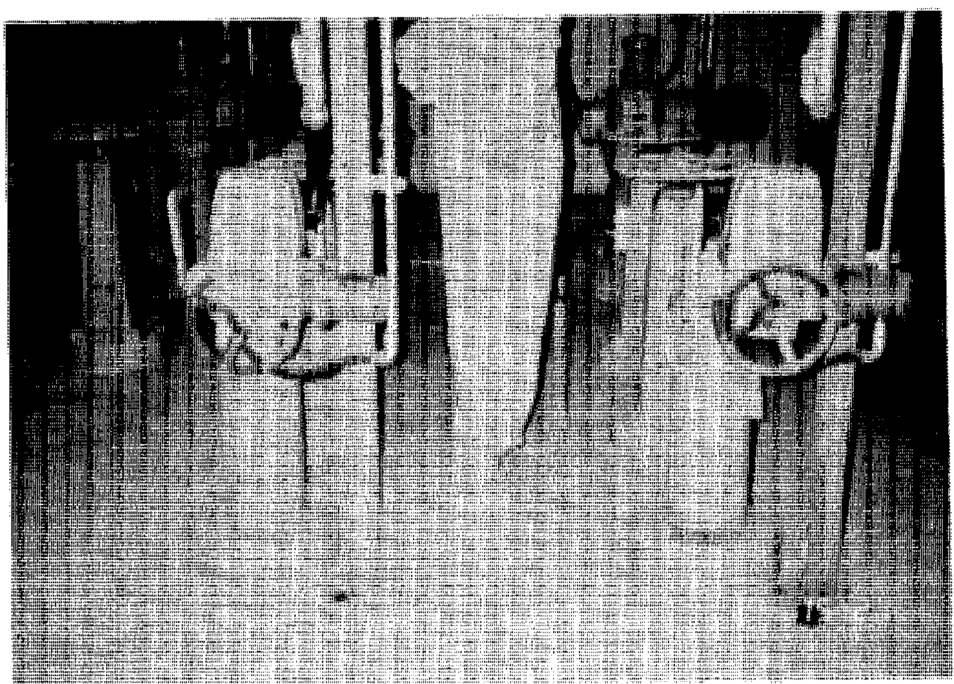


Figure 4.3-4 Maine Yankee CS Pump Suction MOV's CS-M-91 and CS-M-92. Valves with Long Operator Arms.

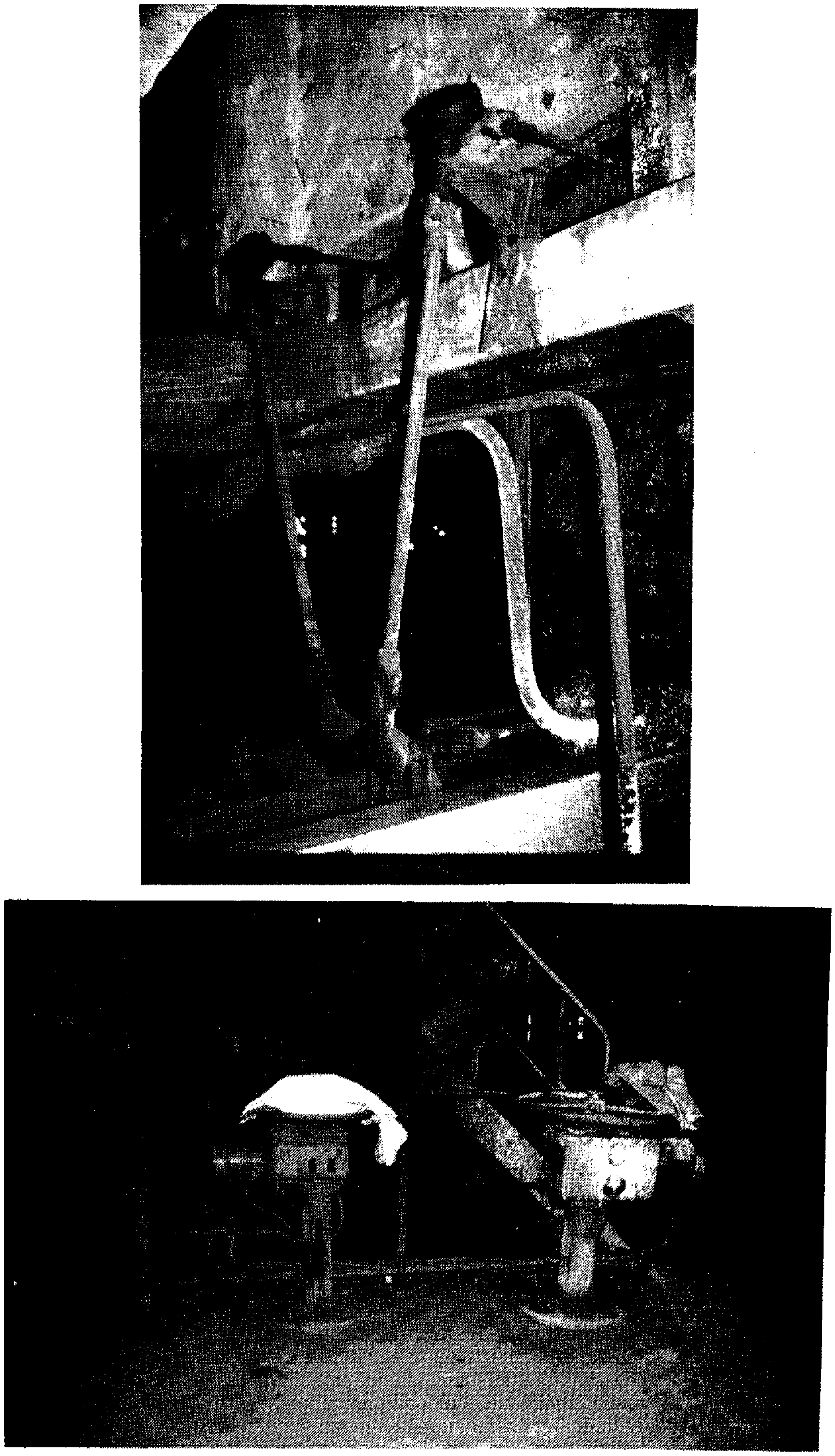


Figure 4.3-5 Example of Long Operator Valves Documented in the Data Base From the Coalinga Area.

the foundations of different structures would be expected to occur in a 0.3g pga earthquake. The areas which are of a possible concern for a plant on a rock site are places where short spans of piping traverse two buildings at a location which will experience large relative displacements. This could occur high up in a relatively flexible structure. Maine Yankee has very few piping systems which traverse two structures at elevations above grade. The most critical location at Maine Yankee exists for the steam lines running between the steam and feedwater valve area and the turbine building (Figure 4.3-6). These interbuilding pipes were judged to have a high capacity for two reasons:

- The length of piping is long relative to the diameter of the piping. The relative building displacements required to fail the piping would have to be very large given the distance between buildings and the enormous ductility which has been demonstrated for welded steel piping.

- The steam and feed valve area structure is a very stiff concrete structure and will have a small displacement at the mid-height location of these steam pipes. Displacements of the turbine building on the other end of the pipe will be limited by the stiffer control building.

The cast iron service water header from the service water pumps to the PCC and SCC Heat Exchangers was found to be predominately buried. The only exposed portion of the piping was from Pump Discharge to ground penetration in the Pumphouse. The exposed portion of the cast iron service water piping was retrofitted with large seismic supports which were anchored back to the pumphouse structure. Short span lengths and adequate bracing were observed during the walkdown, such that capacity was judged greater than the margin earthquake level of 0.3g. The buried portion of the service water header could not be inspected; however, Maine Yankee calculations were reviewed with the subsequent capacity of the buried service water header determined to be greater than 0.3g.

<u>Cable Trays and Cabling</u>. Under the Panel's guidelines, cable trays and cabling are screened out for review earthquake levels less than 0.3g pga. The Expert Panel recommended that example cable trays be inspected to verify that they are adequately anchored and braced and to confirm that taut cables will not be affected by anticipated relative displacement or any sharp edges at the ends of the trays.

Cable tray walkdown effort was focused on the following areas which contain much of the cable trays in the plant:

- Turbine/service building
  Cable vault, El. 21'-0"
  Cable tray room, El. 35'-0"
  Cable tray area, El. 35'-0"
  Switchgear room, El. 45'-6"
  Turbine hall, El. 21'-0"
- Circulating water pumphouse
- Primary auxiliary building, El. 21'-0"

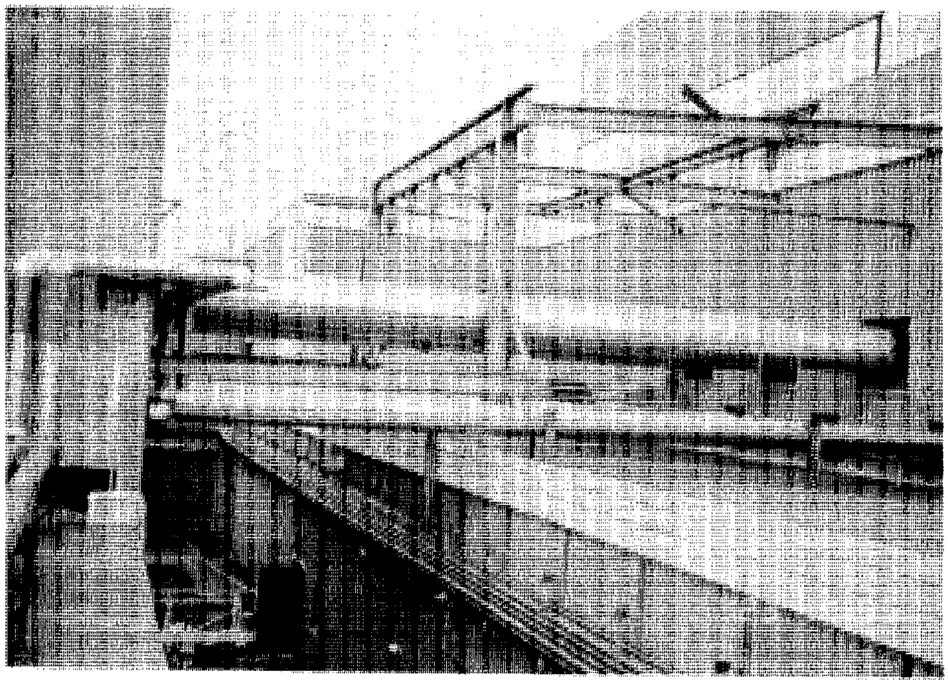


Figure 4.3-6 Steam Lines Spanning Between the Steam and Feed Water Valve Area (Left) and the Turbine Building.

- o M.C.C. room
  M.C.C. rooms, El. 21'-0" and El. 33'-4"
  Containment penetration area, El. 46'-0"
- o Ventilation equipment area

Cable trays in other areas are generally similar to cable trays in the areas listed above.

Typical Maine Yankee cable trays are shown in Figure 4.3-7. The following general description of the Maine Yankee cable trays is obtained from the walkdowns:

- o Cables are typically contained in 24-inch-wide aluminum, ladder type cable trays.

- o Maximum cable tray loading was found in the service building cable vault where inserts were installed within the trays to permit cable fill somewhat in excess of 100% relative to the tray itself.

- o Cable tray supports are typically unbraced rod hung trapezes, screwed into Phillips Redhead concrete inserts. Braced cantilever bracket floor to ceiling column supports are used in the cable vault. Wall-mounted cantilever brackets were found at a few locations in the plant.

- o Support spacings are six feet or less.

- o Supports carry a maximum of six tiers of cable trays.

Cable tray systems similar to those found at Maine Yankee have been subjected to shake table testing and actual earthquakes. The test programs have shown that cable tray systems are capable of withstanding significant seismic input levels without gross damage that would compromise cable integrity. [EQE Inc., 1986] presents a summary of the performance of cable tray and conduit systems in past earthquakes. Experience data shows that cable tray and conduit systems constructed according to normal industrial standards have a large capacity for the absorption of seismic inertial loads. There have only been a few instances of local damage to cable tray systems due to actual earthquakes. These occurrences did not compromise the structural integrity of the cable tray systems or the integrity or function of the supported cables. The single instance of cable tray collapse involved an anomalous support configuration not observed at Maine Yankee.

To confirm that the Maine Yankee cable trays can be screened out at the 0.3g pga review earthquake level, they were compared to cable trays in the earthquake experience data base [EQE Inc., 1986]. This comparison was performed for a number of parameters important to cable tray seismic adequacy, including the following associated with seismic input, seismic response, and system integrity:

- o Input Parameters
  - Peak ground acceleration
  - Duration of strong ground motion

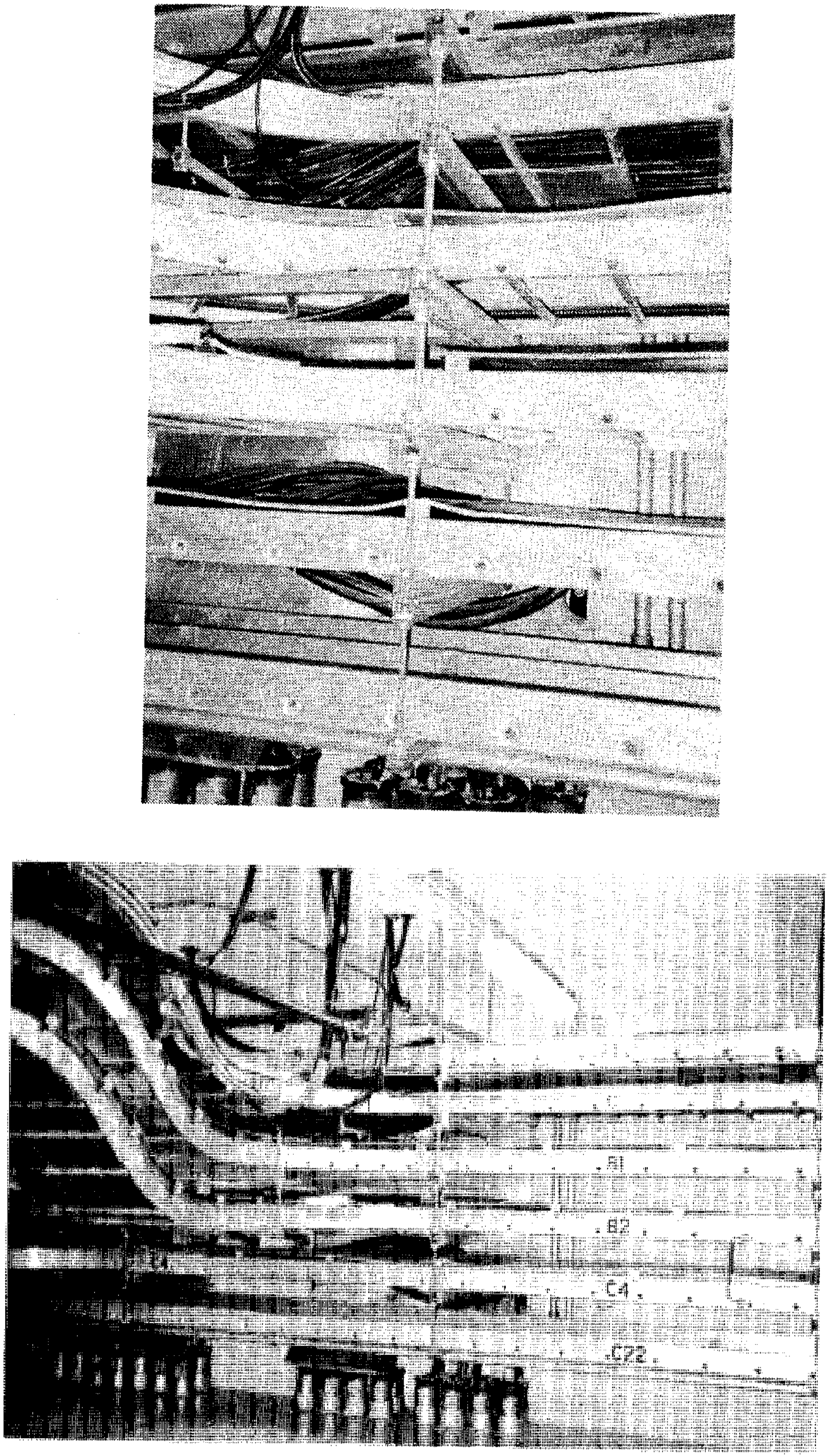


Figure 4.3-7 Photographs of Typical Maine Yankee Cable Trays.

- Frequency content of ground motion
- Soil type
- Building type and size
- Elevation in building

o Response Parameters
  - Extent and complexity of systesm
  - System interfaces
  - System interactions

o System Integrity Parameters
  - Support type and support members
  - Connection details
  - Cable tray type
  - Cable tray loading
  - Support span
  - Number of tiers per support

The Maine Yankee cable trays were found to be enveloped by the experience data base for these parameters, thereby demonstrating that they can be screened out for the 0.3g pga review earthquake.

Potential concerns identified by the Expert Panel include failure of taut cables due to large relative displacements, severing of cables caused by sharp edges at the ends of cable trays, and failure of welds. Review of the Maine Yankee cable trays did not identify any of these conditions.

Based upon a comparison of the Maine Yankee cable trays with cable trays in the earthquake experience data base [EQE Inc., 1986], and walkdown of the systems for particular problem areas, it is concluded that the cable trays and cabling can be screened out for the review earthquake level.

Instrument Racks. The sample of instrument racks reviewed during the course of the walkdown were all judged to have a HCLPF capacity of greater than the 0.3g pga review level earthquake. The racks reviewed were observed to be similar to rack configurations documented in the experience data base. The racks were constructed of structural angle with bracing near the base well anchored to the concrete floor or walls with expansion anchors. The rack member connections were of all welded construction. The instruments and attached components were all observed to be well anchored to the rack face plate. Typical Maine Yankee instrument racks are shown in Figure 4.3-8.

Control Room Ceiling. Control room ceilings are screened out for review earthquake levels less than 0.3g pga. This is subject to verification that the ceilings are adequately braced and that other overhead fixtures are properly anchored.

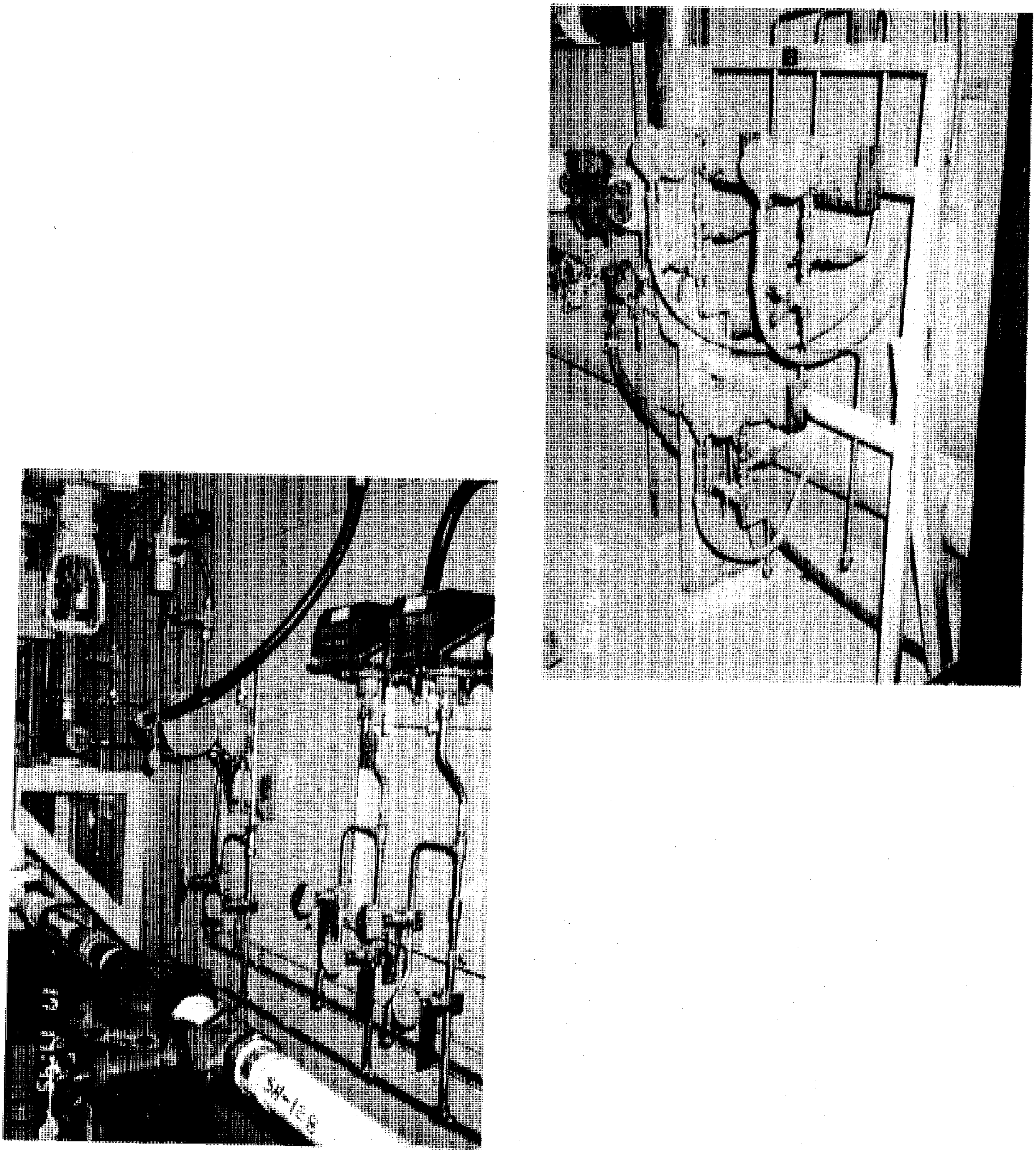
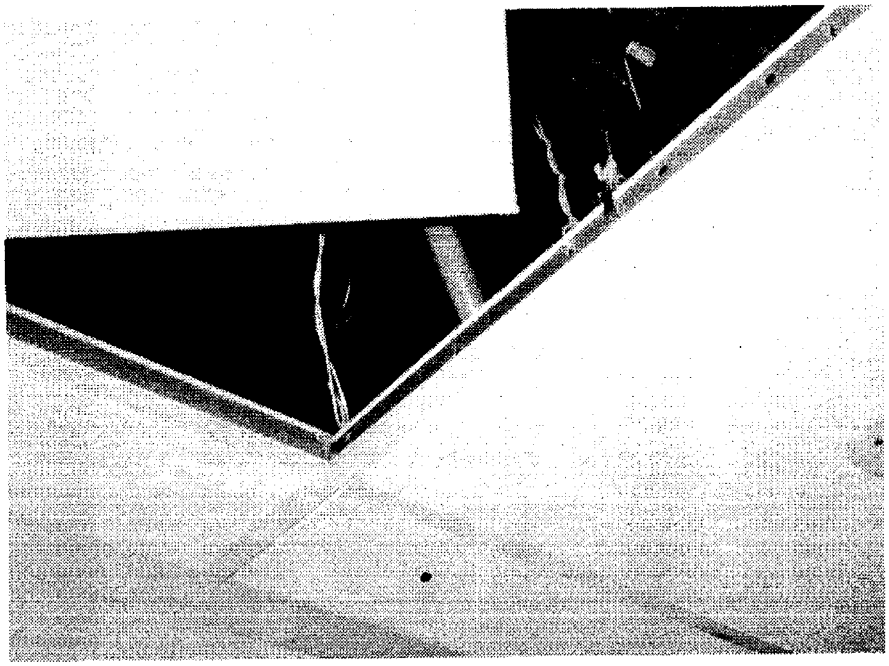


Figure 4.3-8 Photographs of Typical Maine Yankee Instrument Racks and Component Attachments.



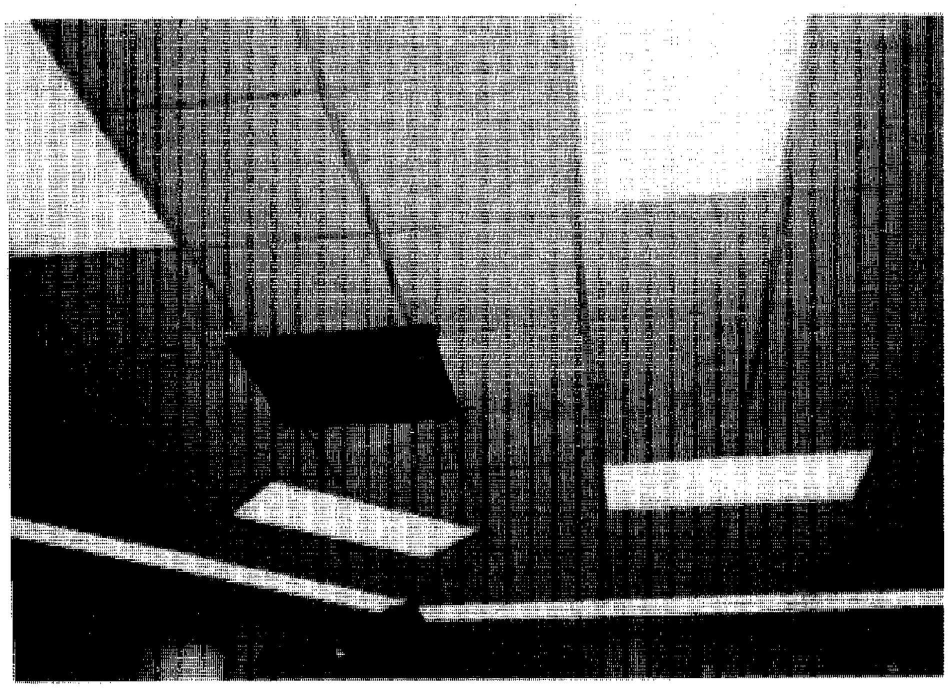


Figure 4.3-9 Photographs of the Maine Yankee Control Room Ceiling.

The suspended ceiling system over the control room is the conventional T-bar type with fibrous acoustic panels typically 2'-0" square. The T-bars are safety wired to the concrete slab above. The safety wiring is judged to be sufficient to prevent collapse of the T-bar ceiling to the floor below. The acoustic panels are not positively attached to the T-bars. Suspended ceiling panels have been dislodged in past earthquakes. However, this has not been a source of damage to control panels or injury to operators. Furthermore, attempts to push the Maine Yankee ceiling panels out met with significant resistance due to the presence of the safety wire loops at the T-bar intersections. Damage resulting from falling ceiling panels is considered unlikely.

The control room and battery room light fixtures are made of sheet metal and are suspended from the slab above. They are safety wired to the slab similar to the T-bar ceiling (Figure 4.3-9). This should be sufficient to prevent the light fixtures from dropping due to the 0.3g review level earthquake. It may be possible for translucent panels and light tubes to become dislodged during an earthquake. However, as with the ceiling panels, the likelihood of damage to equipment and personnel below is small.

Other ceiling fixtures above the control room include HVAC ducting and conduit. As with the rest of the plant, the ducting is typically rod hung and braced by rods attached to the ceiling by concrete inserts. Conduit above the control room is typically about two inches in diameter or less. It is supported by threaded rod hangers anchored to the ceiling by concrete inserts, similar to the cable trays. Ducting and conduit are judged to be adequate for the review level earthquake of 0.3g.

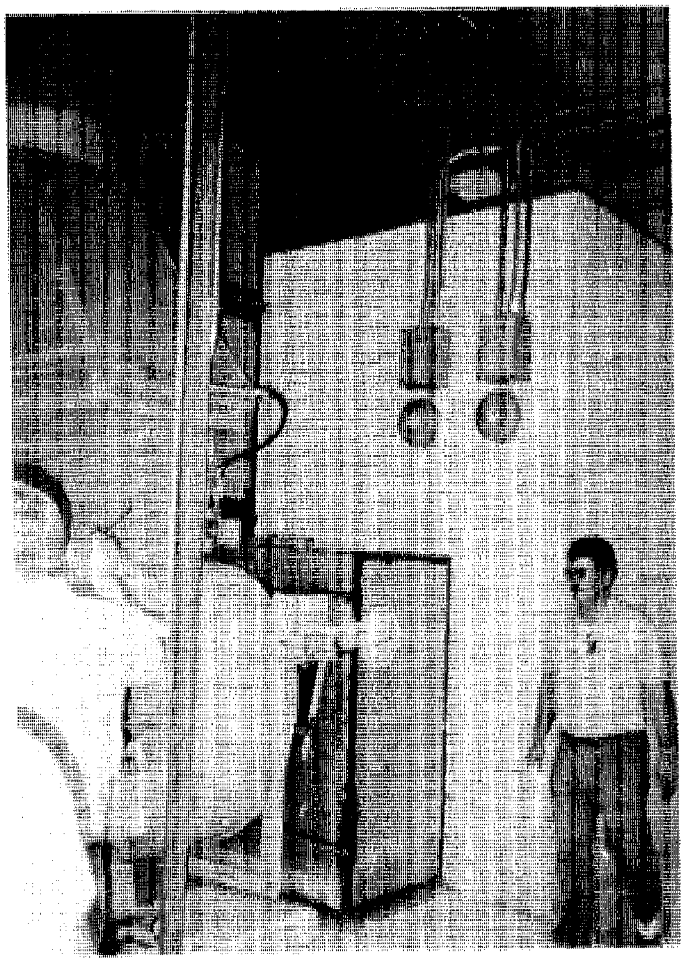
In conclusion, review of the control room ceiling and other overhead fixtures confirmed that they could be screened out.

4.3.4.2 Plant Unique Features

During the review and walkdown of the plant, particular attention was paid to identify any unique features. The unique features are defined as the ones that either have proved to be important contributors in the past seismic PRAs or have not been considered in the previous PRAs. The example of the first kind is the Jocasse dam at Oconee. At Maine Yankee, the earth dike enclosing the fire water pond was examined for potential safety significance to the plant. It was found that terrain around the plant is such that the water from the pond would flow away from the plant in case of a dike failure.

Other features that were not addressed in the previous seismic PRAs and that should be examined in future seismic margin studies as a result of this study are:

- Mussel pump
- Steel structures
- Lead-antimony batteries
- Anchorage of the transformer core/coil assembly



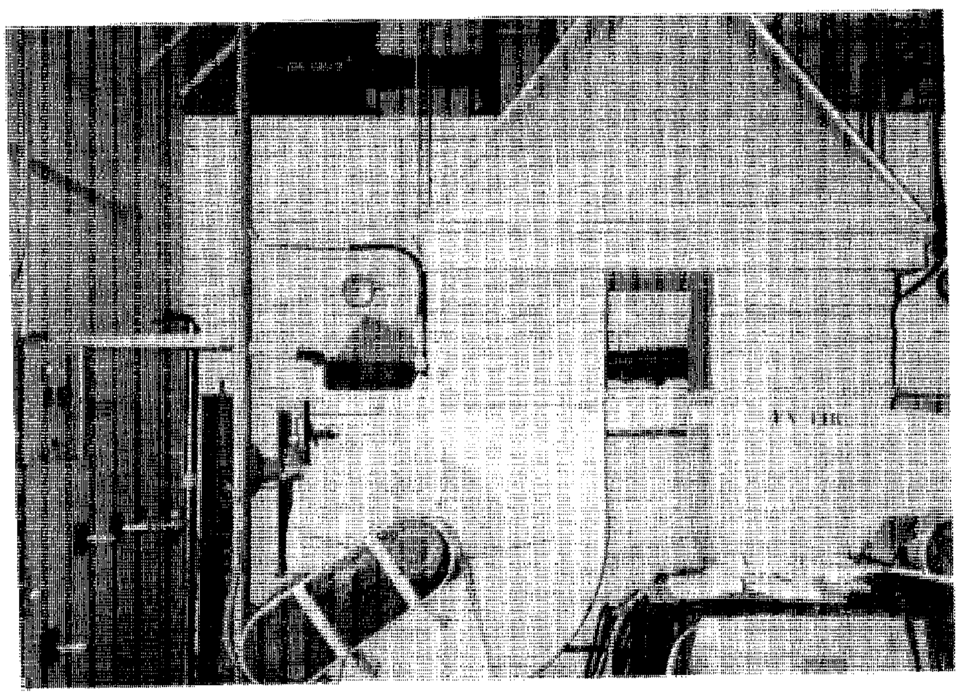


Figure 4.4-1 Block Wall VE 21-1.

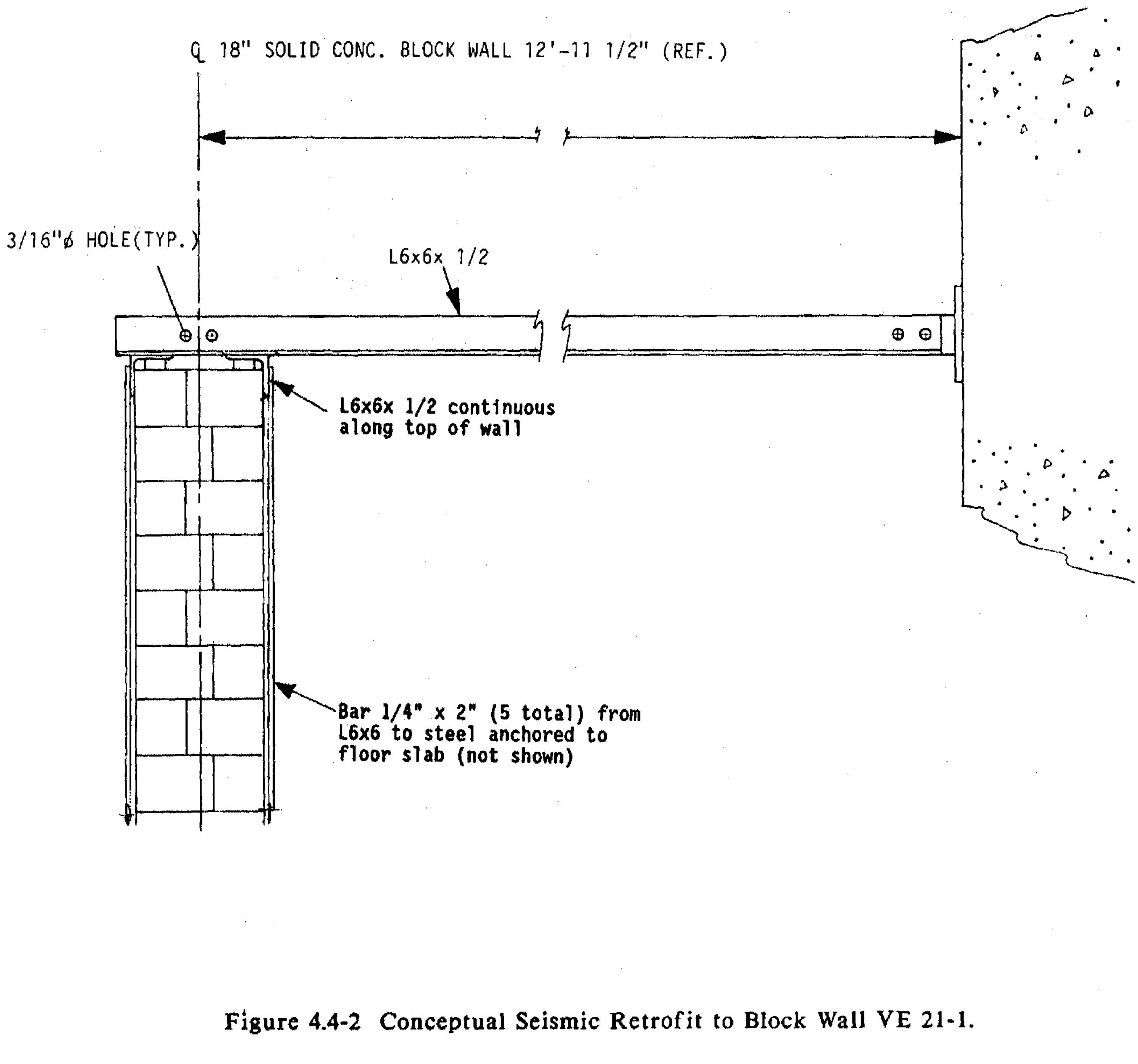


Figure 4.4-2 Conceptual Seismic Retrofit to Block Wall VE 21-1.

The report by Energy Incorporated (Vol.2) discusses the significance of mussel pump and where similar situations could occur in other plants. The Panel report should include some discussions and guidance on reviewing steel structures and about their seismic capacities as discussed in Chapter 6. The topics of lead-antimony batteries and transformers are covered in later sections.

4.4 Maine Yankee Component Modifications

Several Maine Yankee components were identified during the margin evaluation as potentially having a relatively low capacity. These components included critical equipment items identified as part of the margin review and noncritical items that posed a possible interaction hazard to critical equipment. During the course of the margin review Maine Yankee took upon itself to modify those components identified as having a potentially low capacity. The installation of the modifications occurred in two stages. Modifications which could be installed without affecting plant operation were undertaken. Modifications which would be disruptive and pose a risk to plant operation and maintenance personnel were scheduled for the March 1987 refueling outage. The modifications completed before the refueling outage were verified during the second walkdown; however, a third walkdown during or after the March 1987 outage is necessary to verify the modifications scheduled to be completed during that time period (reference Section 6.4). The specific component modifications described below are reflected in the structures, equipment and plant capacities presented in Section 5.

Structures

Block Wall VE 21-1. Block Wall VE 21-1 is located in the ventilation equipment room. This wall is adjacent to the containment spray fans, Fans 44A and 44B, and their ducting and filter (Figure 4.4-1). It is freestanding and built integral with an intersecting wall at one side. Structural drawings show this wall to be 18 in. thick by 10 ft 0 in. high, and constructed of solid concrete block.

Failure of Wall VE 21-1 due to out-of-plane seismic response could result in damage to the adjacent ducting and filter, thus causing the fans to draw air from the ventilation equipment room itself rather than the containment spray pumphouse. This wall may be particularly vulnerable to seismic effects since it is freestanding. Conceptual sketches for seismic retrofits call for structural steel restraints to be added at the top and bottom. Top supports will consist of steel angles spanning back to the concrete wall of the containment spray pumphouse (Figure 4.4-2). Calculations based upon the preliminary retrofit sketches provided indicate that the modifications should be sufficient to provide the wall with a HCLPF capacity greater than 0.3g.

Equipment

The Emergency Diesel Day Tanks TK-62A & B. The day tanks were observed during the walkdown to be elevated tanks supported from a braced steel frame. Each column of the frame was observed anchored to the concrete floor with two expansion anchor studs with provisions for the installation of two additional studs. Under each column was a leveling plate and grout pad which raised concern about

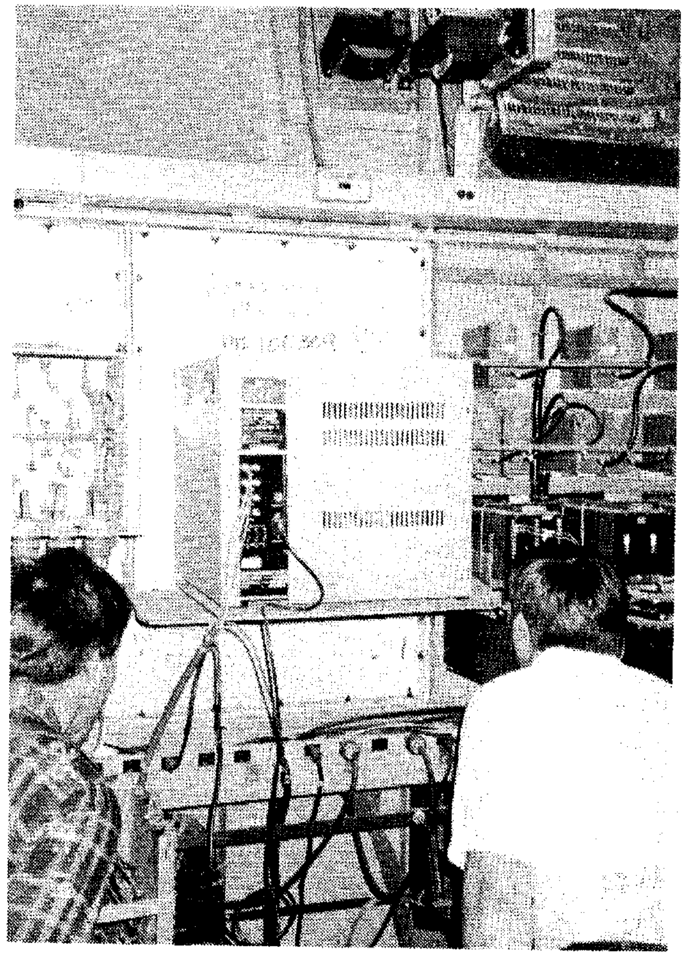


Figure 4.4-3 Photograph of the Alarm Message Display Mounted on the Main Control Board.

the expansion anchor embedment into the concrete floor. To confirm bolt embedment each bolt was ultrasonic tested by Maine Yankee technicians and found to be only partially embedded into the concrete floor. The modification for the day tank anchorage consisted of installing two additional bolts through existing unused holes in the column base plates. Seven-inch Hilti Qwik Bolts were installed with a four-inch minimum embedment into the concrete floor. The modification was performed by Maine Yankee without affecting plant operation or safety.

Main Control Board Alarm Message Display. The Alarm Message Display is located on the Main Control Board and is not critical to plant operation. However, during the first walkdown the component was observed to be unanchored (Figure 4.4-3) and posed a significant risk to adjacent components in the main control board (the component was mounted approximately six feet above the floor). This was a classic example of system interaction. There were anchor bolt holes in the support such that it was believed the component was removed for maintenance, replaced and not anchored. Maine Yankee subsequently reinstalled the bolts which were verified during the second walkdown.

Station Batteries No. 1,2,3 & 4. The station batteries were observed to have lead-antimony flat-plate battery cells during the first walkdown. Lead-antimony flat-plate batteries have been subject to controversy regarding their seismic performance, particularly in the aged state.

There is no positive data substantiating the lack of performance during earthquakes. However, there is sufficient data to suggest a reduced capacity over lead-calcium type batteries where a wealth of test data exists. This lack of data precludes the determination of a HCLPF capacity for lead-antimony battery cells.

Maine Yankee had previously scheduled Station Battery No. 1 for change out during the March 1987 refueling outage based strictly on the battery age. Subsequent to the first walkdown, Maine Yankee personnel decided to replace both batteries No. 1 and No. 3 during the March 1987 refueling outage. Station Batteries No. 2 and 4 were not scheduled to be changed out at this time, but Maine Yankee anticipates replacing them at the next refueling outage. The component and the plant HCLPF capacities presented in Section 5 are based on the Station Batteries No. 1 and 3 being replaced with new C & D lead-calcium batteries, LC-25, supported from battery racks, Model No. RD-938G Type 2. The Station Batteries No. 1 and 3 battery cells and racks should be verified for correct installation during the third walkdown.

Station Service Transformers X-507 & X-608. The station service transformer enclosures were observed to be well anchored during the walkdown. The review of the internal transformer core/coil assembly was observed to be not anchored, but supported by four rubber isolators with no vertical uplift or lateral restraints. Figure 4.4-4 shows several photographs of the transformer core/coil assembly in the unanchored state. The transformers in the unanchored state have a low seismic capacity. Maine Yankee engineers designed a modification to securely anchor the core/coil assembly by utilizing the existing shipping studs. The electrical implications of this modification were discussed with the manufacturer, General Electric. Figure 4.4-5 is a sketch of the proposed modification. Maine Yankee has

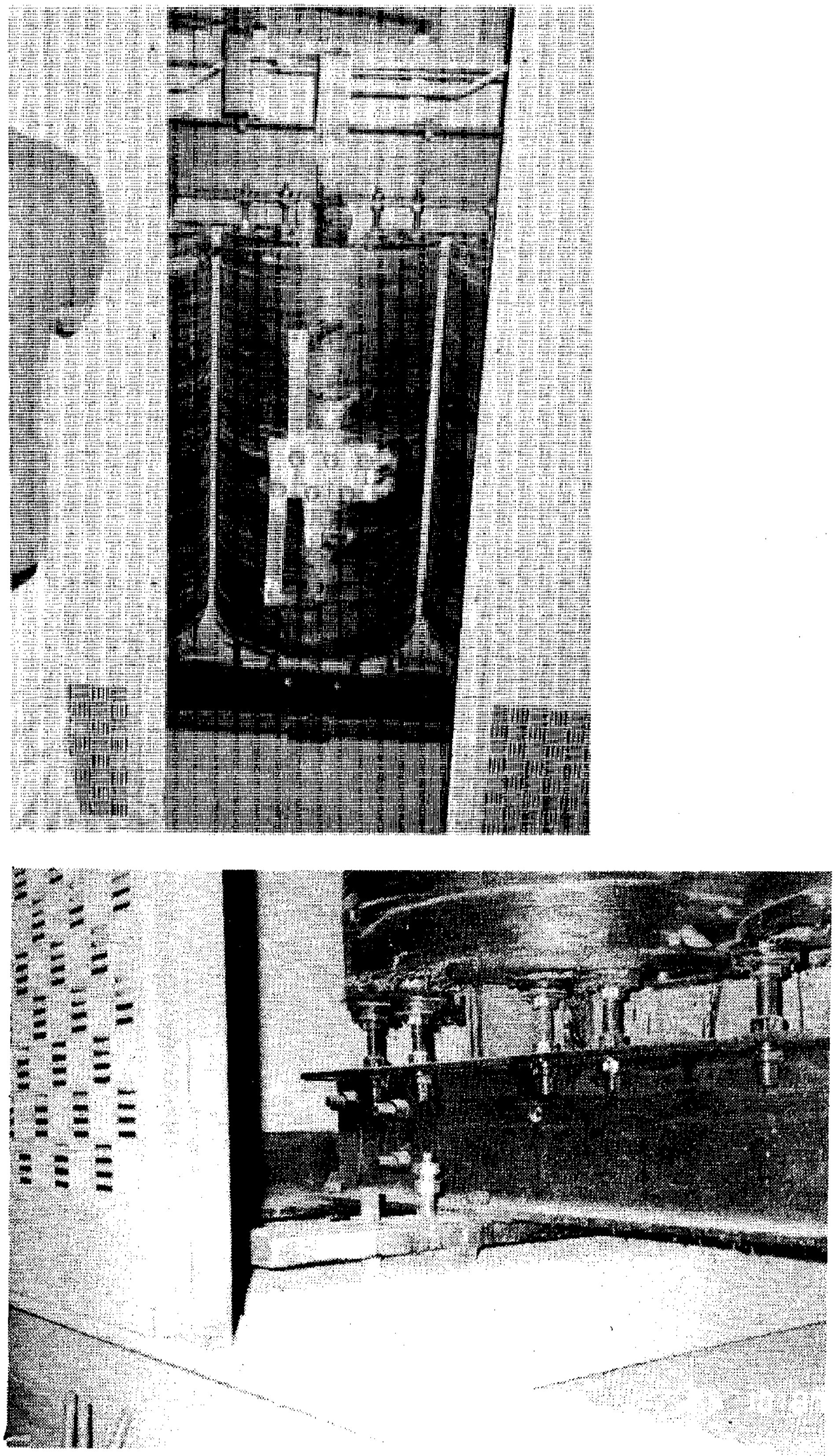
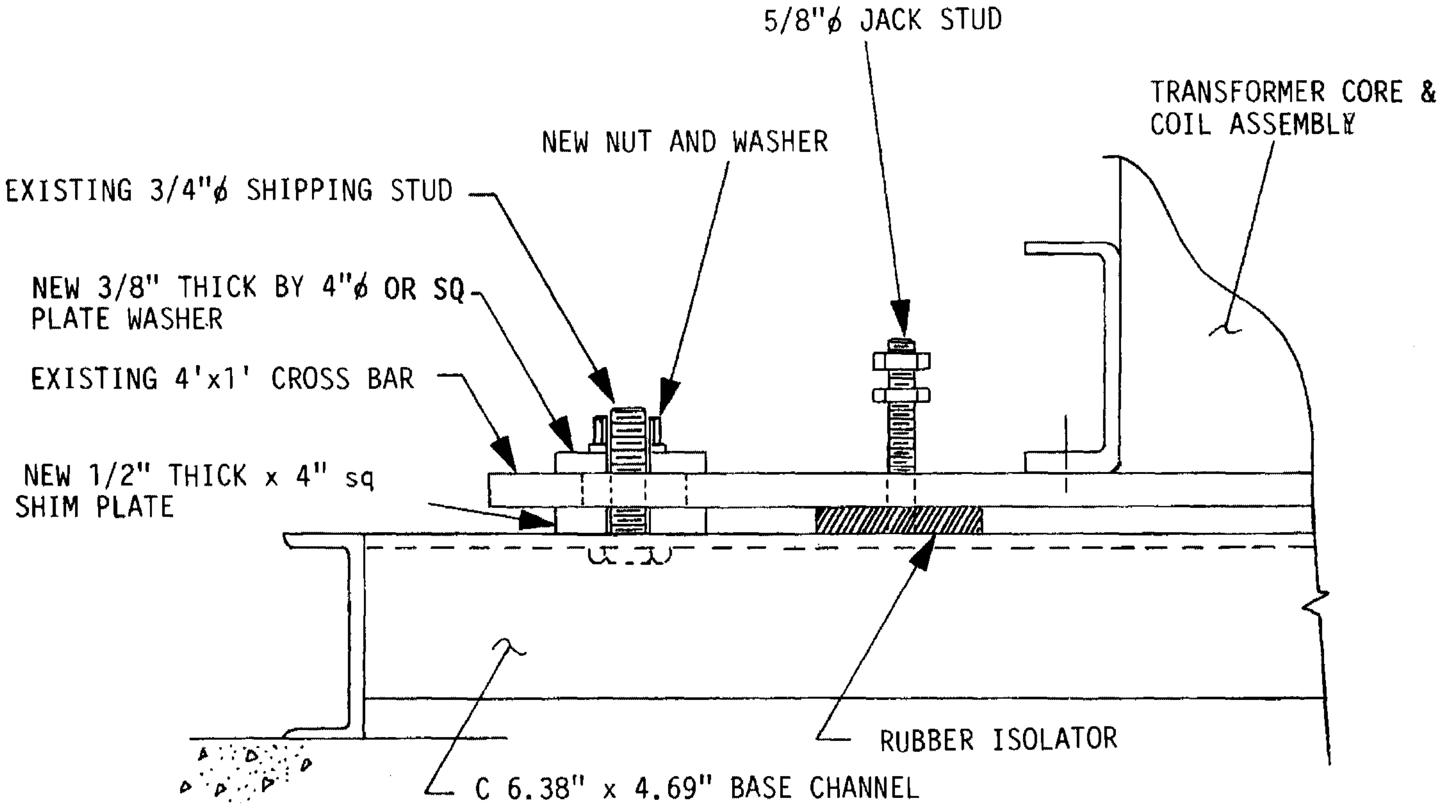
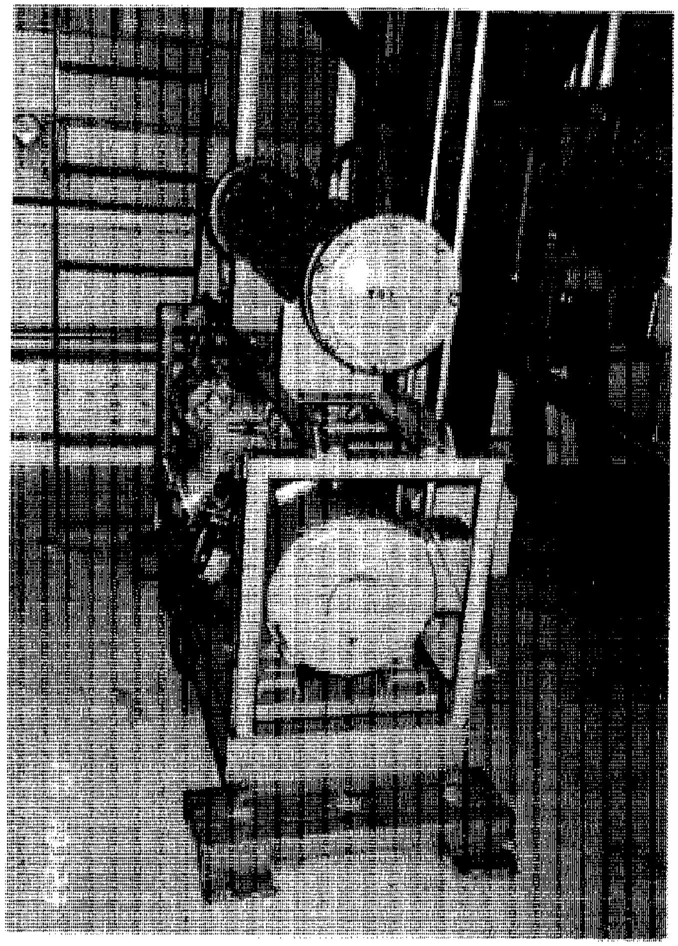


Figure 4.4-4 Photographs of the Maine Yankee Station Service Transformer Internal Core/Coil Assembly Before the Anchorage Modification.



ELEVATION VIEW OF TRANSFORMER CORE/COIL
ASSEMBLE ANCHORAGE (TYP 4 PLACES)

Figure 4.4-5 Proposed Anchorage Modification for the Maine Yankee Station Service Transformers Internal Core/Coil Assemblies.



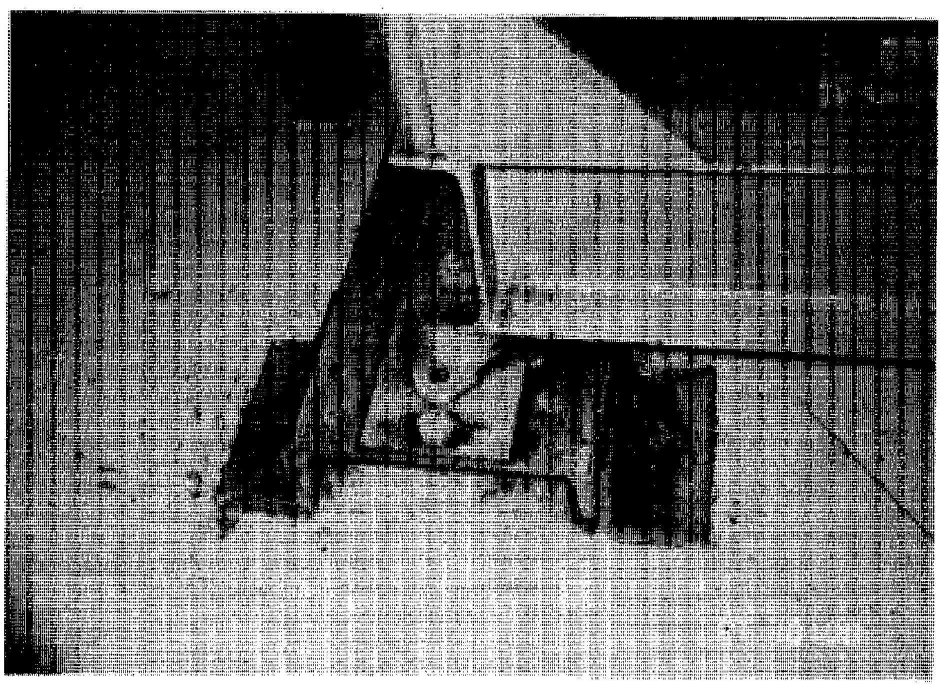


Figure 4.4-6 Maine Yankee Air Conditioners in the Unmodified Condition.

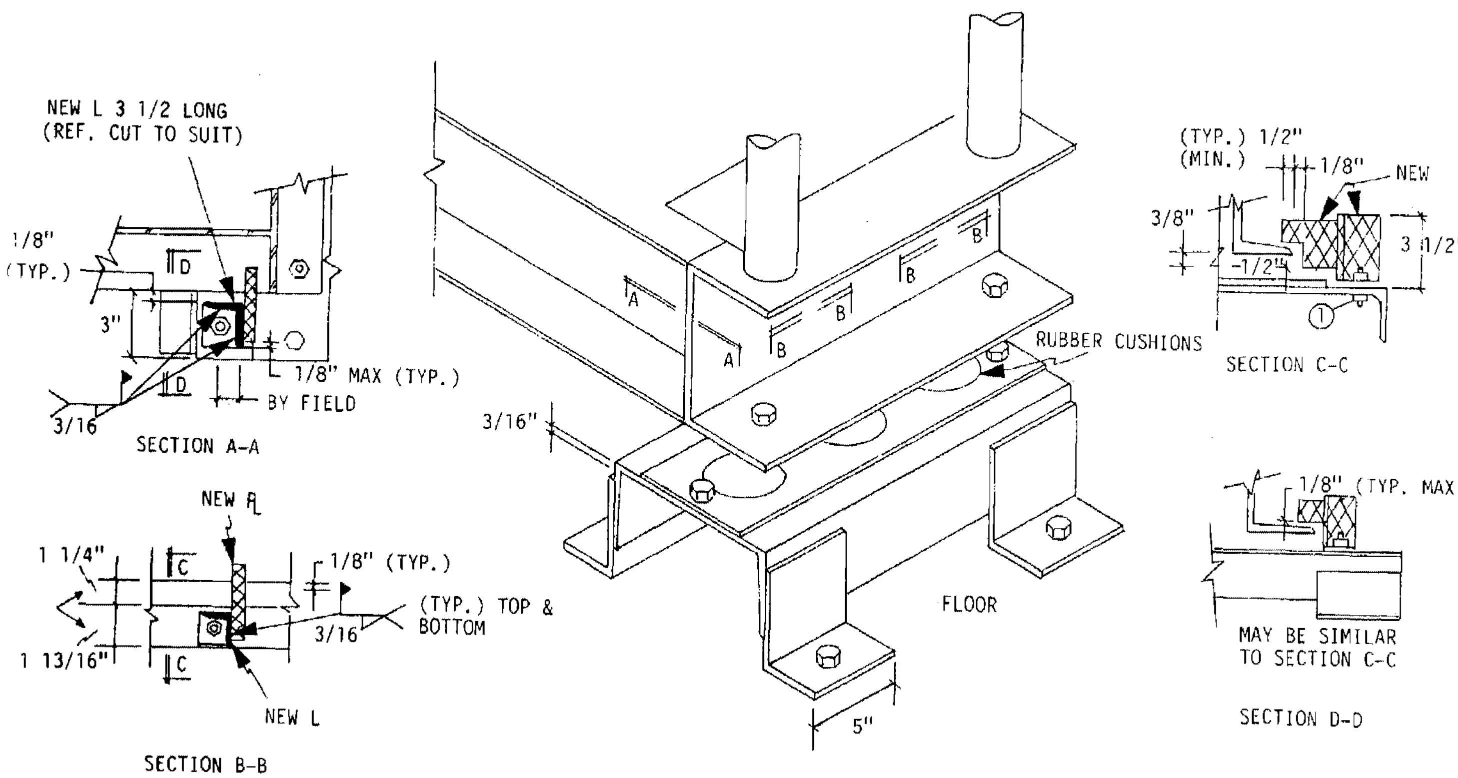


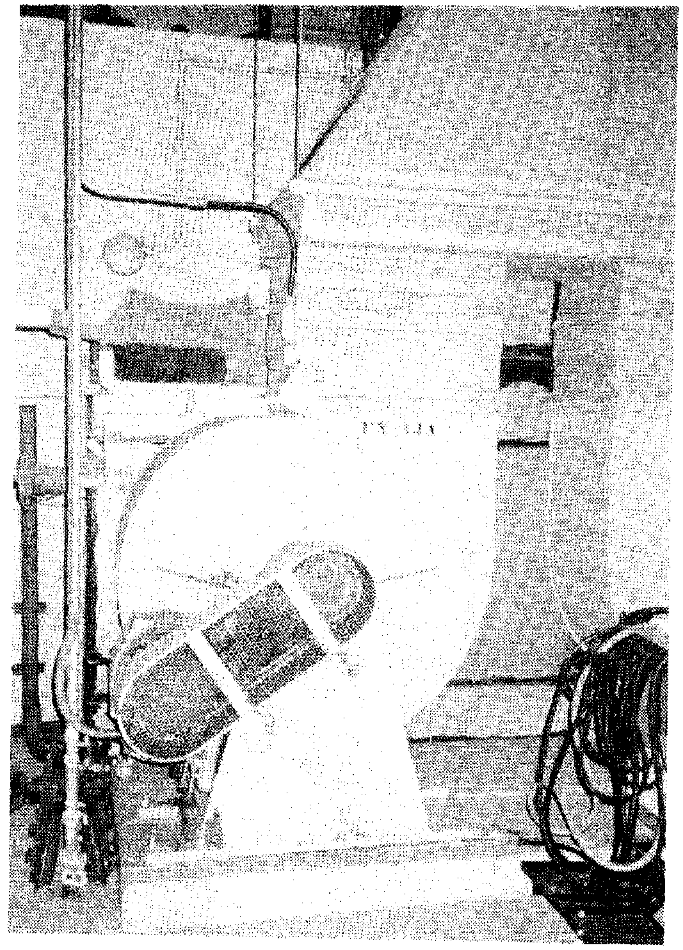
Figure 4.4-7 Proposed Anchorage Modification for the Maine Yankee Air Conditioners.

scheduled the anchorage modification to be installed during the March 1987 refueling outage. The transformer and plant HCLPF capacities represent the modified condition. The station service transformers should also be inspected during the third walkdown to verify the modification has been installed.

HVAC Computer Room and Lab Air Conditioners. The air conditioners reviewed during the walkdown were found to be supported from vibration isolators. Conversations with the isolator manufacturer determined the isolators to have no vertical uplift capacity and marginal lateral load resisting capacity. Maine Yankee developed a modification that would provide vertical and lateral load capacity to be installed during the March 1987 refueling outage. Sketches of the modification were provided to EQE for the capacity evaluation of these components. Figures 4.4-6 and 4.4-7 show photographs of one air conditioner and a sketch of the proposed modification provided to EQE, respectively. The HCLPF capacities of air conditioners represent this modified condition. The air conditioners should also be inspected during the third walkdown to verify the modification has been installed in accordance with the proposed design provided to EQE.

HVAC Containment Spray Fans. The fans were observed during the walkdown to be supported from vibration isolators. The configuration of the isolators was such that vertical uplift was provided. Lateral stability of the isolator configuration was identified as marginal in resisting the seismic loadings. The existing containment spray fan isolator assembly was scrutinized by both the fragility analysts and by the peer review group members. The earthquake experience data base was not useful as a resource for this particular isolator configuration due to a lack of similarity with the data base isolator systems. A rigorous fragility analysis of the cold formed steel attachment strip and the attachment bolt was deemed to be nearly impossible to complete due to low cycle fatigue, lack of ductility, and stress concentration issues associated with the attachment strip. Maine Yankee engineers agreed to design an anchorage upgrade based on the good engineering judgment that a modification was necessary in order to ensure that the fan could withstand (with high confidence) the 0.3g review level earthquake. Maine Yankee provided EQE with a sketch of the proposed modification for the margin evaluation. Figures 4.4-8 and 4.4-9 show photographs of the fans and a sketch of the proposed modification provided to EQE, respectively. The fans and plant HCLPF capacities represent this modified condition. The fans should also be inspected during the third walkdown to verify that the modification has been installed.

Miscellaneous Components. Miscellaneous components include noncritical components which were identified during the walkdown to posses a potential risk to plant safety by failing and impacting a critical equipment component. All of these components fall into the general category of system interaction which the Expert Panel identifies in its review guidelines. These components included the alarm message display discussed above, emergency lighting units throughout the plant, compressed gas boilers, and a welding cart on wheels near the containment spray fans. Figure 4.4-10 shows photographs of the emergency lighting units and the welding cart. All components that were identified during the first or second walkdown have been subsequently modified, secured, or removed by Maine Yankee. These modified components were evaluated during the second walkdown (or based on verbal communication by Maine Yankee engineers in the case of the welding cart) and judged to have a HCLPF capacity greater than the 0.3g pga.



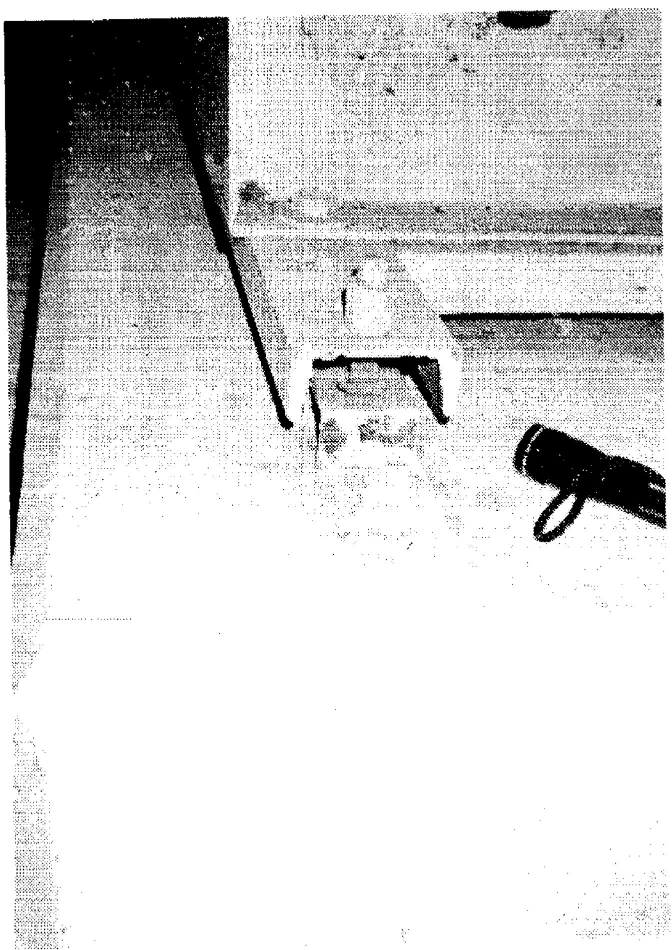


Figure 4.4-8 Maine Yankee Containment Spray Fans Before the Installation of the Anchorage Modification.

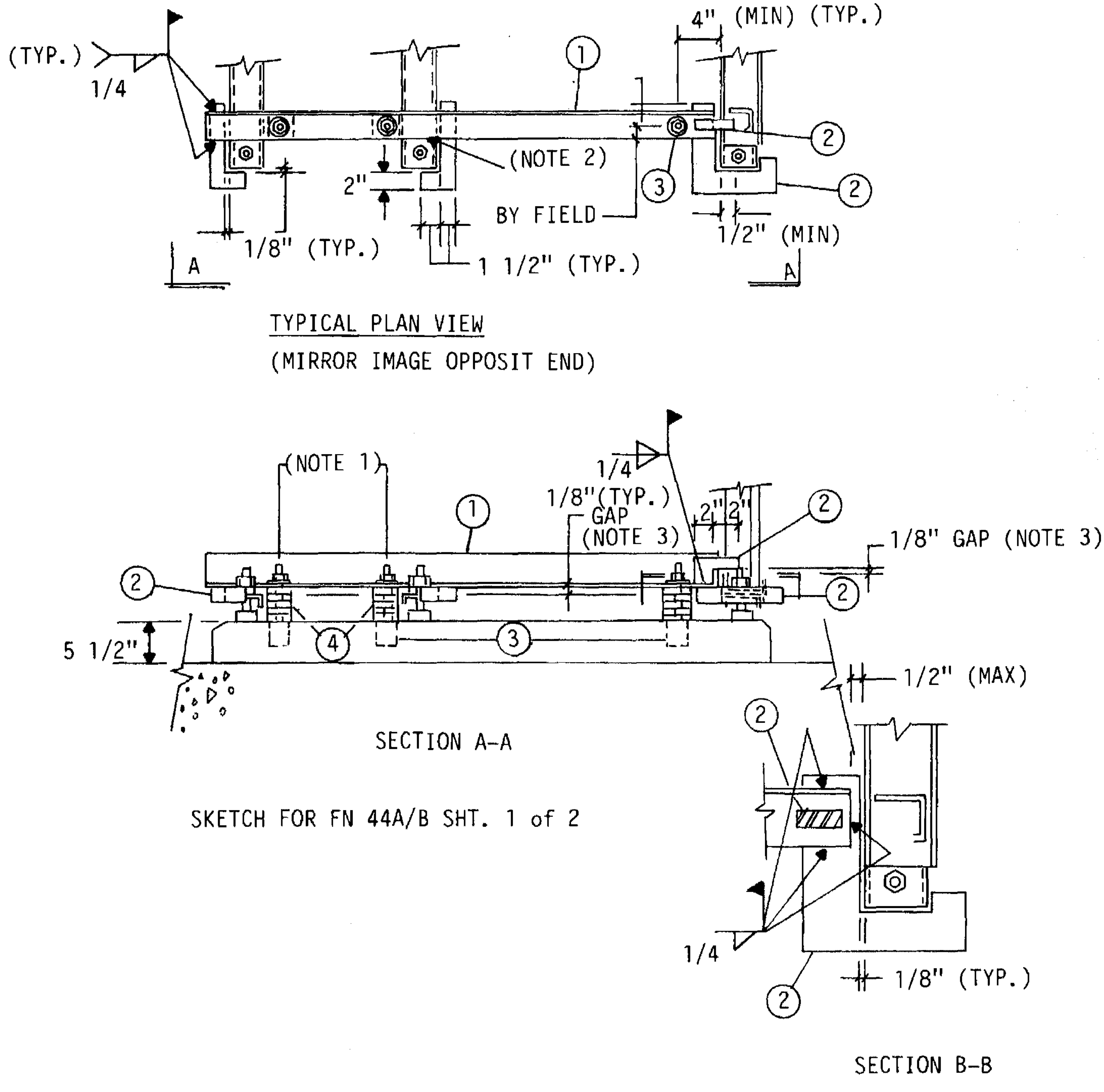
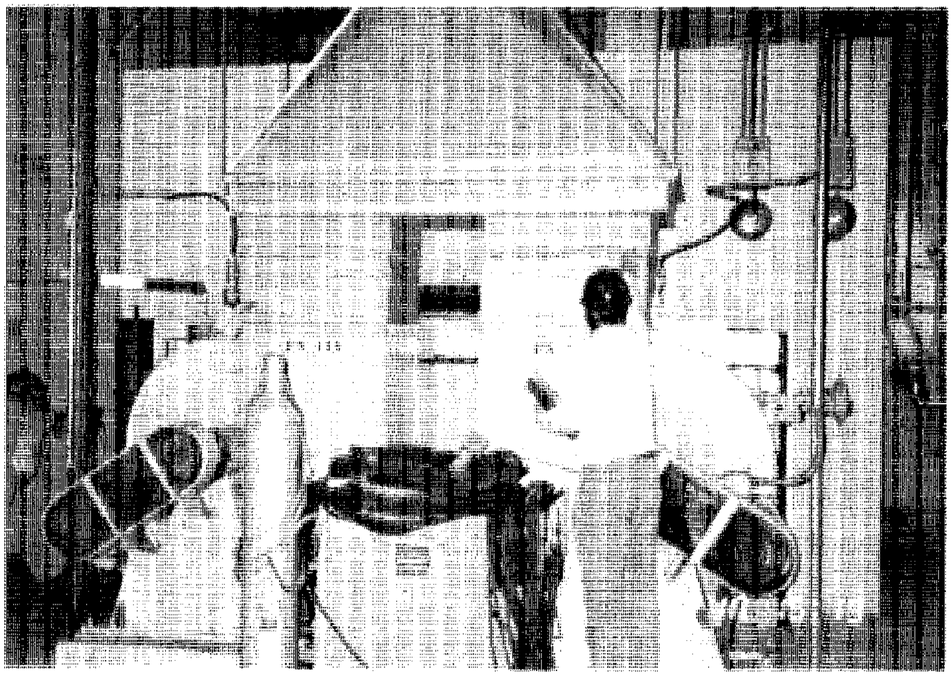


Figure 4.4-9 Proposed Anchorage Modification for the Maine Yankee Containment Spray Fans.



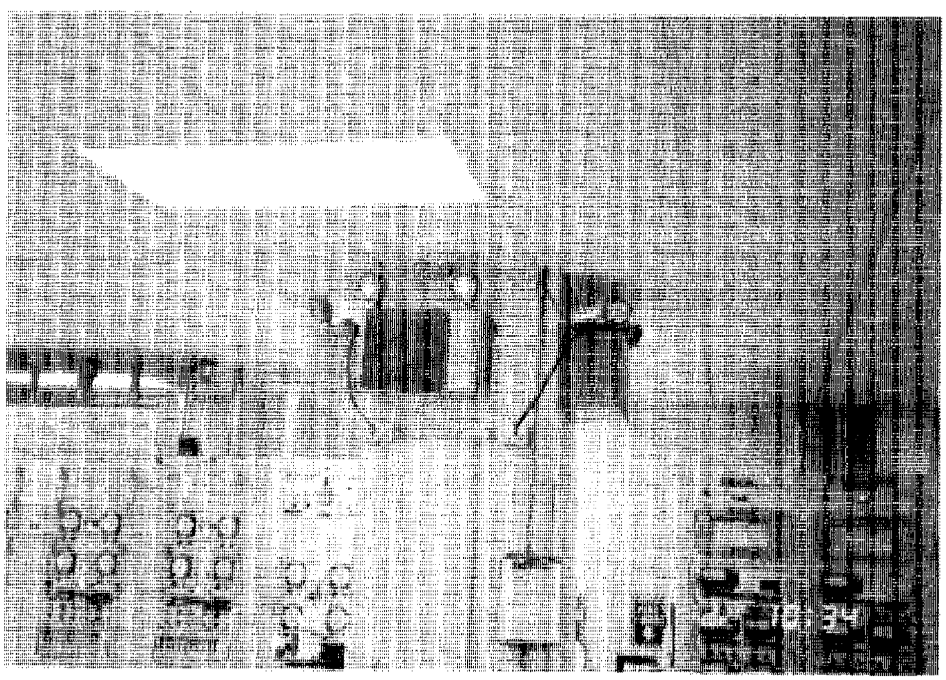


Figure 4.4-10 Photographs of the Unstrapped Emergency Lighting Units in the Control Room and Welding Cart Near the Containment Spray Fans.

CHAPTER 5

EVALUATION OF SEISMIC CAPACITIES OF COMPONENTS AND PLANT

5.1 Review of Structural Models

In-structure response spectra for Maine Yankee were generated by Cygna specifically for use in the trial plant application. Input consisted of acceleration time-histories matching the median NUREG/CR-0098 ground response spectrum. The dynamic analyses were performed with structural models recently developed for use in other Maine Yankee analyses. These models were reviewed to verify that they are adequate to predict responses for the review level earthquake. The floor response spectra generated by Cygna were judged to be adequate for this margin review. Since seismic analyses of structures and components developed for the original design were generally not used to calculate HCLPF capacities, the original design structure dynamic models were not obtained or reviewed.

[Cygna, 1982] describes the dynamic models generated for the following structures:

- o Containment structure
- o Containment internal structure
- o Containment spray pumphouse
- o Main steam valve house
- o Primary auxiliary building
- o Turbine/service building

This report contains the following information:

- o Computer programs used
- o General modeling assumptions
- o Description of individual structure modeling considerations
- o Sketches of seismic load resisting elements
- o Overall structure mass properties
- o Calculated frequencies, modal participation factors, and mode shapes

This information was reviewed to verify the adequacy of the dynamic models in the following manner:

1. Review overall approach and assumptions.
2. Review individual structure models.
   - 2.1 Review general model layout.
   - 2.2 Verify that the major load paths are included.
   - 2.3 Verify accuracy of the overall masses.
   - 2.4 Review reasonableness of the eigen solutions.

The overall approach and assumptions were judged to be consistent with practices within the nuclear industry and generally adequate to predict overall seismic response. All structures were modeled as being linear elastic with fixed base boundary conditions.

The containment was analyzed by a three-dimensional finite-element model using shell elements with uniform mass densities. The effects of concrete cracking for the containment analysis were conservatively modeled by using upper and lower bounds on the modulus of elasticity. Accuracy of the containment model was confirmed by an independent analysis described in [Hashimoto, 1984].

The other structures were analyzed using dynamic models with masses lumped at major floor elevations. Stiffness matrices for the lumped mass models were generated from more refined finite-element representations of the load resisting elements. Uncracked stiffness properties were used for concrete shear walls. Floor slabs were modeled as being infinitely stiff in their own planes.

With the possible exceptions noted below, the dynamic models of the structures were found to be consistent with practices used in the nuclear industry and generally adequate to predict overall seismic response. Model descriptions and sketches in [Cygna, 1982] were reviewed to verify that major load-resisting elements were included. Independent, approximate calculations were performed when possible to confirm accuracy of the lumped masses. The eigen solutions were reviewed using engineering judgment to assess whether they were reasonable given the structure configurations.

Review of the turbine/service building model indicates that the diesel generator enclosure walls contribute significantly to the overall stiffness of the lowest story. However, the load path from the rest of the structure to the diesel generator enclosure is relatively flexible. Although decoupling the diesel generator enclosure from the remainder of the structure reduces the overall structure stiffness, the net effect on seismic response is lessened since exclusion of the diesel generator enclosure also reduces the overall structure mass. Rather than develop a new structure model, correction of the dynamic response was estimated in assessing the structural response factor for the fragility evaluations.

Stiffnesses of the shear wall structures other than containment were based upon uncracked properties. On-going scale model testing being conducted for the NRC has indicated that this may result in an overestimation of the actual stiffness. The effect of this issue on the plant HCLPF capacity was assessed by sensitivity studies described in Section 5.5.4.

Group A structures that were not dynamically analyzed by Cygna are the circulating water pumphouse and the fuel oil pumphouse. Review of the circulating water pumphouse indicates that the concrete portions at and below the operating floor can be considered essentially rigid since the structure is supported by several heavy concrete walls and the exterior north, south, and west walls were poured against excavated rock. Dynamic analysis of the steel superstructure supporting the roof slab was independently generated for the fragility evaluation. Dynamic response of the fuel oil pumphouse was not required since no Group A components are housed in or directly attached to it.

5.2 Simplified Analysis and Use of Screening Tools

NUREG/CR-4334 developed a set of initial screening criteria for seismic margin studies as outlined in Section 4.1 of this report. These initial screening criteria are based on the results of past probabilistic risk assessments and on actual earthquake experience data. These initial screens identified classes of equipment and structures which have consistently demonstrated high seismic capacities. Within the remaining classes of equipment and structures which have not been screened out there typically exists a range of seismic capacities. Simplified analysis techniques can be utilized to separate out components within a category which have very high seismic capacities and do not warrant a detailed fragility analysis.

Simplified analyses for the Maine Yankee margin study were conducted using either of two methods:

- o Fragility derivation using conservative response and capacity parameters and an estimate for the variability ($\beta_R + \beta_U = 0.7$)

- o Deterministic evaluation using conservative values similar to the CDFM approach outlined in NUREG/CR-4482

The 120-V ac vital bus 1A through 4A is a good example of a Maine Yankee component whose seismic margin was evaluated on the basis of a simplified analysis. These 120-V buses are wall-mounted panels containing circuit breakers and are located in the control room (service building at +21 ft). NUREG/CR-4334 does not explicitly screen out active electrical equipment and states that anchorage should be verified for the cabinet and for individual components in the cabinet. The walkdown of these 120-V buses showed them to be relatively lightweight components with overdesigned wall anchorage. A simplified conservative fragility analysis was conducted utilizing the following parameters:

- o Weight = 245 lb (specified on drawing)

- o Spectral acceleration = peak spectral value for 7% damping for the appropriate floor spectra (cabinet is judged to be rigid)

- o Center of gravity = 8 in. away from the wall (conservative since the cabinet is only 10 in. deep)

- o Anchor bolt capacity = industry specified allowable of one-quarter of the ultimate strength

- o $\beta_R + \beta_U = 0.7$

The simplified fragility analysis on these panels resulted in a HCLPF capacity greater than 0.5g. Components whose HCLPF capacities were calculated via simplified analysis methods to be greater than 0.3g were automatically screened out, and detailed fragility derivations were judged to be unnecessary.

The conservatisms most commonly utilized in the simplified analyses are:

- o Natural frequency - conservative estimate on the frequency or use of the frequency corresponding to the peak spectral acceleration

- o Capacity - conservative estimate such as the Code allowable, 70% of the ultimate, or failure at the yield strength

- o Combined randomness and uncertainty variability - conservatively estimated based on calculations for similar components at Maine Yankee

5.3 Second Walkdown

The second walkdown was performed to accomplish the following tasks:

- o Obtain additional detailed information on equipment and structures inspected during the first walkdown.

- o Survey components added to the Group A equipment list after the first walkdown was performed.

HCLPF capacities were calculated for certain components immediately following the first walkdown. In determining these capacities, additional information needs were identified. The first task listed above was performed to obtain this information for finalization of component HCLPF capacities. A minor amount of effort was also devoted towards additional documentation of items reviewed in the first walkdown. A limited number of components were added to the Group A equipment list after the first walkdown was completed based upon more detailed system analyses. The second task was performed to obtain information on these added components for screening and/or HCLPF capacity determination. Structures and equipment reviewed in the second walkdown are listed in Table 5.3-1.

5.4 HCLPF Capacity of Components

In the following, the fragility analysis methodology is described with some illustrative examples.

5.4.1 Fragility Analysis Method

5.4.1.1 Methodology

In this method, the component HCLPF capacity is calculated using the fragility of the component. The seismic fragility of a structure or equipment is defined as the conditional probability of its failure at a given value of peak ground acceleration. The methodology for evaluating seismic fragilities of structures and equipment is documented in [Ravindra and Kennedy, 1983], [PRA Procedures Guide, 1983], and [Kennedy and Ravindra, 1984] and has been developed and applied in over 20 seismic PRAs.

Table 5.3-1 Second Walkdown Component Review List

TURBINE BUILDING

COMPONENT	FLOOR ELEVATION
1. Structural details	
2. Block wall TB 21-2	21' - 0"
3. Primary Component Cooling Heat Exchanger E-4A	21' - 0"
4. Primary Component Cooling Heat Exchanger E-4B	21' - 0"
5. Secondary Component Cooling Heat Exchanger E-5A	21' - 0"
6. Secondary Component Cooling Heat Exchanger E-5B	21' - 0"
7. Cooler Supply Temperature Control Valve PCC-T-19	37' - 0"
8. Cooler Bypass Temperature Control Valve PCC-T-20	37' - 0"

SERVICE BUILDING

COMPONENT	FLOOR ELEVATION
1. Structural details	
2. Block Wall SB 21-7	21' - 0"
3. Block Wall SB 21-17	21' - 0"
4. Block Walls SB 35-1 to 4	35' - 0"
5. Block Walls SB 39-1 and 2	39' - 0"
6. Block Wall SB 45-6	45' - 0"
7. Block Wall SB 61-2	61' - 0"
8. Block Wall SB 77-2	77' - 0"
9. General inspection of the Switchgear Room	45' - 6"
10.Protected SWGR. Room Supply Fan FN-31	39' - 0"
11.Computer Room Air Conditioner AC-1A	39' - 0"
12.Computer Room Air Conditioner AC-1B	39' - 0"
13.Lab Air Conditioner AC-2	39' - 0"
14.General inspection of the Control Room	21' - 0"
15.Cable Trays	
16.Control Room Ceiling	21' - 0"

TURBINE BUILDING AUXILIARY BAY

COMPONENT	FLOOR ELEVATION
1. Auxiliary Fuel Oil Supply Tanks Tk-28 A & B	21' - 0"
2. Diesel Fuel Oil Day Tank Tk-62 A & B	21' - 0"
3. DG Compressed Air Tanks Tk-76A & B1-6	21' - 0"
4. DG Room Exhaust Fans FN-20A & B	31' - 0"
5. Diesel Generator Air Intake & Exhaust Dampers	31' - 0"
6. DG-1B Cooler Inlet Temperature Control Valve SCC-T-305	23' - 0"

Table 5.3-1 Second Walkdown Component Review List (Continued)

PRIMARY AUXILIARY BUILDING

COMPONENT	FLOOR ELEVATION
1. Charging Pump Seal Leakage Coolers E-92A & B	11' - 0"
2. Seal Water Heater E-96	
3. Auxiliary Charging Pump P-7 Lube Oil Cooler	11' - 0"
4. Return from Penetration Coolers PCC-A-238	12' - 0"
5. Seal Water Supply Filter FL-34B	
6. PCCW Isolation from the RHR Heat Exchanger PCC-M-90	11' - 0"
7. PCCW Isolation to Containment PCC-M-219	11' - 0"
8. Cable Trays	

STEAM AND FEED WATER VALVE AREA

COMPONENT	FLOOR ELEVATION
1. Structural details	
2. AFW Pump B Turbine Throttle Valve MS-A-173	21' - 0"
3. Turbine Steam Supply Pressure Control MS-P-168	21' - 0"

CONTAINMENT SPRAY PUMP HOUSE

COMPONENT	FLOOR ELEVATION
1. Residual Heat Removal Heat Exchangers E-3A & B	14' - 0"
2. Reactor Coolant Regenerative Heat Exchanger E-67	
3. Safeguards Pumps Seal Leakage Cooler E-91A & B	00 - 0"
4. LPSI Pump Coolers	21' - 0"
5. Containment Spray Pump P-61A, B & S Coolers	14' - 0"
6. Containment Penetration Cooling Lines	
7. Block Wall VE 21-1	21' - 0"

FUEL BUILDING

COMPONENT	FLOOR ELEVATION
1. Block Walls	
2. Fuel Pool Heat Exchanger E-25	21' - 0"

Table 5.3-1 Second Walkdown Component Review List (Continued)

MCC ROOM

COMPONENT	FLOOR ELEVATION
1. Cable Trays	

MISCELLANEOUS COMPONENTS

COMPONENT	FLOOR ELEVATION
1. Instrument Racks	various

The objective of fragility evaluation is to estimate the ground acceleration capacity of a given component. This capacity is defined as the peak ground acceleration value at which the seismic response of a given component located at a specified point in the structure exceeds the component's resistance, resulting in its failure. The ground acceleration capacity of the component is estimated using information on plant design bases, responses calculated at the design-analysis stage, as-built dimensions, and material properties. The ground acceleration capacity is a random variable which can be described completely by its probability distribution. However, there is uncertainty in the estimation of the parameters of this distribution, the exact shape of this distribution, and in the appropriate failure model for the component. For any postulated failure model and set of parameter values and shape of the probability distribution, a fragility curve depicting the conditional probability of failure as a function of ground acceleration can be obtained. Hence, for different models and parameter assumptions, one could obtain different fragility curves. A satisfactory way to consider these uncertainties is to represent the component fragility by means of a family of fragility curves obtained as above; a subjective probability value is assigned to each curve to reflect the analyst's degree of belief in the model that yielded the particular fragility curve. When represented in this fashion, the fragility curves need not appear to be smooth S-shaped curves, approximately parallel to each other; they could intersect each other and they may not even be nondecreasing functions of peak ground acceleration. The only requirement is that fragility being a probability should be between 0 and 1 (see Figure 5.4-1).

At any acceleration value, the component fragility (i.e., conditional probability of failure) varies from 0 to 1; this variation is represented by a subjective probability distribution. On this distribution we can find a fragility value (say, 0.01) that corresponds to the cumulative subjective probability of 5%. We have 5% cumulative subjective probability (confidence) that the fragility is less than 0.01. Similarly, we can find a fragility value for which we have a confidence of 95%. Note that these statements can be made without reference to any probability model. Using this procedure, the median and high (95%) and low (5%) confidence fragility curves can be drawn. On the high confidence curve, we can locate the fragility value of 5%; the acceleration corresponding to this fragility on the high confidence curve is the so called HCLPF capacity of the component. By characterizing the component fragility through a family of fragility curves, the analyst has expressed all his knowledge about the seismic capacity of the component along with the uncertainties. Given the same information, two analysts with similar experience and expertise would produce approximately the same fragility curves. Development of the family of fragility curves using different failure models and parameters for a large number of components in a seismic margin review or seismic PRA is impractical if it is done as described above. Hence, a simple model for the fragility was proposed as described in the above cited references. In the following this fragility model is described.

Fragility Model

The entire fragility family for an element corresponding to a particular failure mode can be expressed in terms of the best estimate of the median ground acceleration capacity, A_m, and two random variables. Thus, the ground acceleration capacity, A, is given by

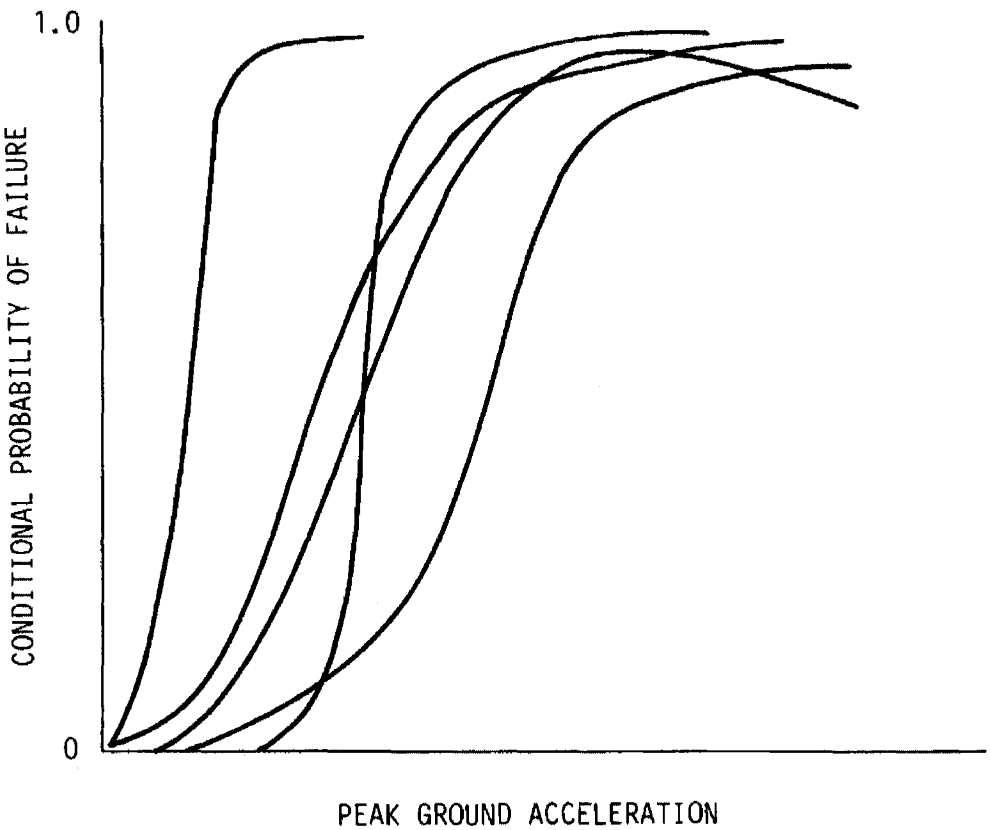


Figure 5.4-1 Fragility Curves.

$$A = A_m \epsilon_R \epsilon_U \ , \qquad (5.4\text{-}1)$$

in which β_R and β_U are random variables with unit medians, representing, respectively, the inherent randomness about the median and the uncertainty in the median value. In this model, we assume that both ϵ_R and ϵ_U are lognormally distributed with logarithmic standard deviations, β_R and β_U, respectively. The formulation for fragility given by Eq. (5.4-1) and the assumption of lognormal distribution allow easy development of the family of fragility curves which appropriately represent fragility uncertainty. For the quantification of fault trees in the plant system and accident sequence analyses, the uncertainty in fragility needs to be expressed in a range of conditional failure probabilities for a given ground acceleration. This is achieved as explained below:

With perfect knowledge (i.e., only accounting for the random variability, ϵ_R), the conditional probability of failure, f_0, for a given peak ground acceleration level, a, is given by

$$f_0 = \Phi\left[\frac{\ell n\left(\frac{a}{A_m}\right)}{\beta_R}\right] \qquad (5.4\text{-}2)$$

where $\Phi(\cdot)$ is the standard Gaussian cumulative distribution function. The relationship between f_0 and a is the median fragility curve plotted in Figure 5.4-2 for a component with a median ground acceleration capacity A_m = 0.90g and β_R = 0.30. For the median conditional probability of failure range of 5% to 95%, the ground acceleration capacity would range from 0.55g to 1.48g.

When the modeling uncertainty ϵ_U is included, the fragility becomes a random variable (uncertain). At each acceleration value, the fragility f can be represented by a subjective probability density function. The subjective probability, Q (also known as "confidence") not exceeding a fragility f' is related to f' by

$$f' = \Phi\left[\frac{\ell n\left(\frac{a}{A_m}\right) + \beta_U \Phi^{-1}(Q)}{\beta_R}\right] \qquad (5.4\text{-}3)$$

where

Q = P[f < f' / a] i.e., the subjective probability (confidence) that the conditional probability of failure, f, is less than f for a peak ground acceleration a

$\Phi^{-1}(\cdot)$ = the inverse of the standard Gaussian cumulative distribution function.

For example, the conditional probability of failure f' at acceleration 0.4g that has a 95% nonexceedance subjective probability (confidence) is obtained from Eq. (5.4-3) as 0.22. The 5% to 95% probability (confidence) interval on the failure at 0.4g is 0 to 0.22. Subsequent computations are made easier by discretizing the random

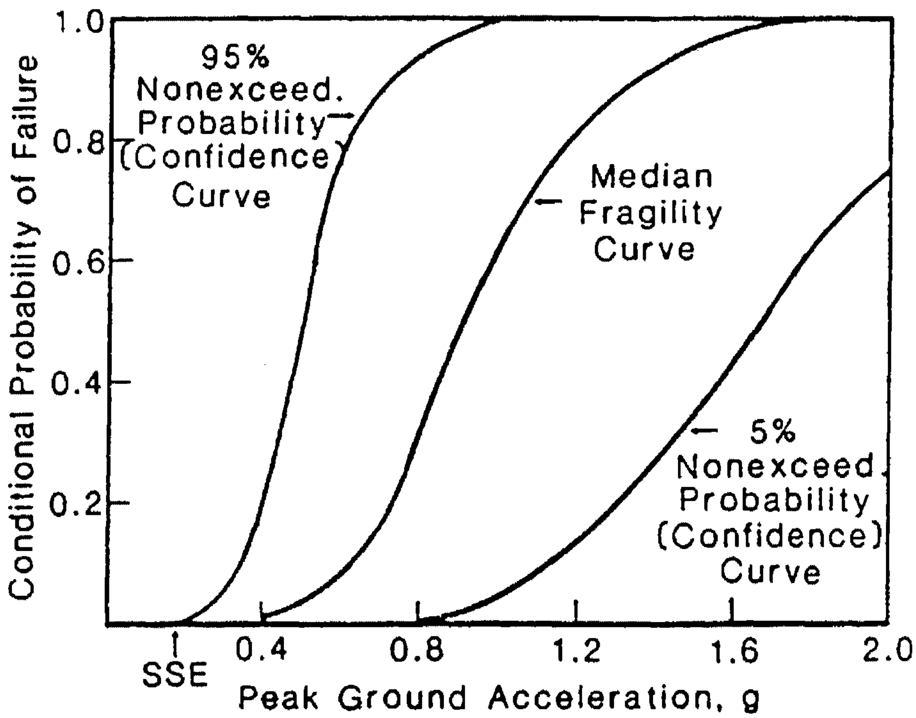


Figure 5.4-2 Median, 5% Nonexceedence, and 95% Nonexceedence Fragility Curves for a Component.

variable probability of failure f into different intervals and deriving probability q_i for each interval (Figure 5.4-3). Note that the sum of q_i associated with all the intervals is unity. The process develops a family of fragility curves, each with an associated probability q_i.

The median ground acceleration capacity A_m, and its variability estimates β_R and β_U are evaluated by taking into account the safety margins inherent in capacity predictions, response analysis, and equipment qualification, as explained below.

Failure Modes

The first step in generating fragility curves such as those in Figure 5.4-3 is to develop a clear definition of what constitutes failure for each of the critical elements in the plant. This definition of failure must be agreeable to both the structural analyst generating the fragility curves and the systems analyst who must judge the consequences of component failure. Several modes of failure (each with a different consequence) may have to be considered and fragility curves may have to be generated for each of these modes. For example, a motor-actuated valve may fail in any of the following ways:

1. Failure of power or controls to the valve (generally related to the seismic capacity of the cable trays, control room, and emergency power). Since they are not related to the specific item of equipment (i.e., motor actuated valve) and are common to all active equipment, such failure modes are most easily handled as failures of separate systems linked in a series to the equipment.

2. Failure of the motor.

3. Binding of the valve due to distortion and, thus, failure to operate.

4. Rupture of the pressure boundary.

It may be possible to identify the failure mode most likely to be caused by the seismic event by reviewing the equipment design and considering only that mode. Otherwise, fragility curves are developed based on the premise that the component could fail in any one of all potential failure modes.

Identification of the credible modes of failure is largely based on the analyst's experience and judgment. Review of plant design criteria, calculated stress levels in relation to the allowable limits, qualification test results, seismic fragility evaluation studies done on other plants, and reported failures (in past earthquakes, in licensee event reports and fragility tests) are useful in this task.

Structures are considered to have failed functionally when they cannot perform their designated functions. In general, structures have failed functionally when inelastic deformations under seismic load are estimated to be sufficient to potentially interfere with the operability of safety-related equipment attached to the structure, or fractured sufficiently so that equipment attachments fail. These failure modes represent a conservative lower bound of seismic capacity since a

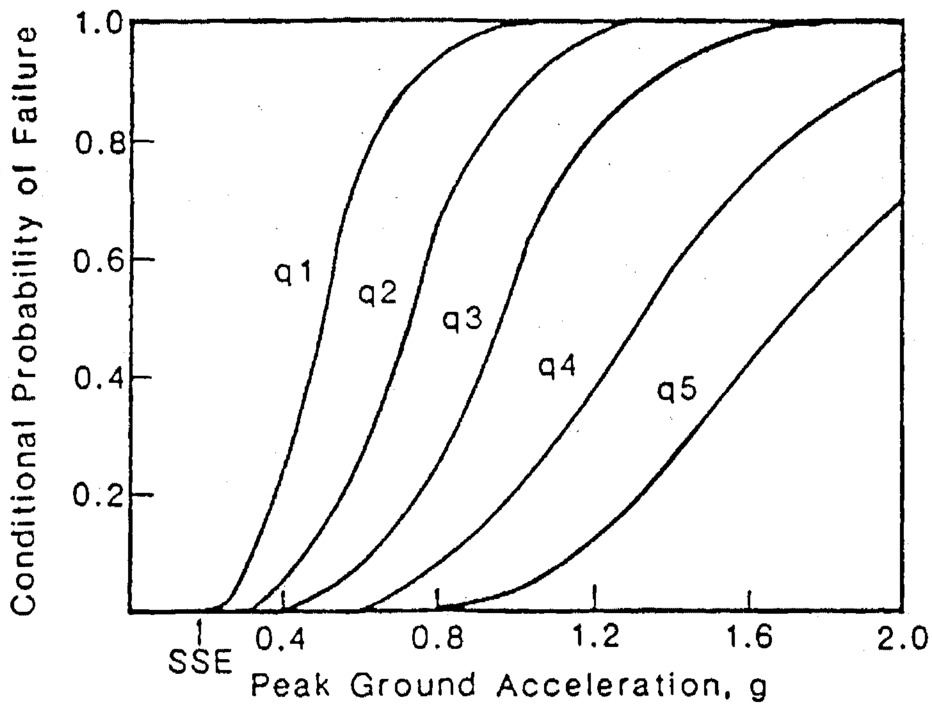


Figure 5.4-3 Family of Fragility Curves for a Component.

larger margin of safety against collapse exists for nuclear structures. Also, a structural failure has been generally assumed to result in a common cause failure of multiple safety systems, if these are housed in the same structure. For example, the service water pumps in Zion were assumed to fail when the crib house pump enclosure roof collapses.

For piping, failure of the support system and plastic collapse of the pressure boundary are considered dominant failure modes. Failure modes of equipment examined may include structural failure modes (e.g., bending, buckling of supports, anchor bolt pullout, etc.), functional failures (binding of valve, excessive deflection), and relay trip or chatter.

Consideration should also be given to the potential for soil failure modes (e.g., liquefaction, toe bearing pressure failure, base slab uplift, and slope failures). For buried equipment (i.e., piping and tanks), failure due to lateral soil pressures may be an important mode. Seismically induced failures of structures or equipment under impact of another structure or equipment (e.g., a crane) may also be a consideration. Seismically induced failures of dams, if present, resulting in either flooding or loss-of-cooling-source, should also be investigated.

Estimation of Fragility Parameters

In estimating fragility parameters, it is convenient to work in terms of an intermediate random variable called the factor of safety. The factor of safety, F, on ground acceleration capacity above the safe shutdown earthquake level specified for design, A_{SSE}, is defined as follows:

$$A = F\,A_{SSE}$$

$$F = \frac{\text{Actual seismic capacity of element}}{\text{Actual response due to SSE}}$$

$$= \frac{\text{Actual capacity}}{\text{Calculated capacity}}$$

$$\times \frac{\text{Calculated capacity}}{\text{Design response due to SSE}}$$

$$\times \frac{\text{Design response due to SSE}}{\text{Actual response due to SSE}}$$

F is further simplified as:

$$F = \frac{\text{Actual capacity}}{\text{Design response due to SSE}}$$

$$\times \frac{\text{Design response due to SSE}}{\text{Actual response due to SSE}}$$

$$F = F_C F_{RS}. \qquad (5.4\text{-}4)$$

Note F can also be defined with reference to a different earthquake such as the review earthquake level in this margin study.

The median factor of safety, F_m, can be directly related to the median ground acceleration capacity, A_m, as:

$$F_m = \frac{A_m}{A_{SSE}} \qquad (5.4\text{-}5)$$

The logarithmic standard deviations of F, representing inherent randomness and uncertainty, are identical to those for the ground acceleration capacity A.

For structures, the factor of safety can be modeled as the product of three random variables:

$$F = F_S \, F_\mu \, F_{RS} \qquad (5.4\text{-}6)$$

The strength factor, F_S, represents the ratio of ultimate strength (or strength at loss-of-function) to the stress calculated for A_{SSE}. In calculating the value of F_S, the nonseismic portion of the total load acting on the structure is subtracted from the strength as follows:

$$F_S = \frac{S - P_N}{P_T - P_N}, \qquad (5.4\text{-}7)$$

where S is the strength of the structural element for the specific failure mode, P_N is the normal operating load (i.e., dead load, operating temperature load, etc.) and P_T is the total load on the structure (i.e., sum of the seismic load for A_{SSE} and the normal operating load). For higher earthquake levels, other transients (e.g., SRV discharge, and turbine trip) may have a high probability of occurring simultaneously with the earthquake; the definition of P_N in such cases should be extended to include the loads from these transients.

The inelastic energy absorption factor (ductility), F_μ, accounts for the fact that an earthquake represents a limited energy source and many structures or equipment items are capable of absorbing substantial amounts of energy beyond yield without loss-of-function. A suggested method to determine the deamplification effect resulting from inelastic energy dissipation involves the use of ductility modified response spectra [Newmark, 1977]. The deamplification factor is primarily a function of the ductility ratio defined as the ratio of maximum displacement to displacement at yield. More recent analyses [Riddell and Newmark, 1979] have shown the deamplification factor to be a function of system damping. One might estimate a median value of for low-rise concrete shear walls (typical of auxiliary building walls) of 4.0. The corresponding median F_μ value would be 2.4. The variabilities in the inelastic energy absorption factor, F_μ, are both estimated as $\beta_R = 0.21$ and $\beta_U = 0.21$, taking into account the uncertainty in the predicted relationship between F_μ, μ, and system damping.

The structure response factor, F_{RS}, recognizes that in the design analyses structural response was computed using specific (often conservative) deterministic response parameters for the structure. Because many of these parameters are random (often with wide variability) the actual response may differ substantially from the design analyses calculated response for a given peak ground acceleration.

The structure response factor, F_{RS}, is modeled as a product of factors influencing the response variability:

$$F_{RS} = F_{SA} F_{\varphi} F_{\delta} F_M F_{MC} F_{EC} F_{SD} F_{SS} , \qquad (5.4\text{-}8)$$

where

F_{SA} = spectral shape factor representing variability in ground motion and associated ground response spectra

F_{φ} = direction factor representing the variability in the two earthquake direction response spectral values about the mean value

F_{δ} = damping factor representing variability in response due to difference between actual damping and design damping

F_M = modeling factor accounting for uncertainty in response due to modeling assumptions

F_{MC} = mode combination factor accounting for variability in response due to the method used in combining dynamic modes of response

F_{EC} = earthquake component combination factor accounting for variability in response due to the method used in combining earthquake components

F_{SD} = factor to reflect the reduction with depth of seismic input

F_{SS} = factor to account for effect of soil-structure interaction.

The median and logarithmic standard deviations of F are expressed as:

$$F_m = F_{Sm}\, F_{\mu m}\, F_{SAm}\, F_{\varphi m}\, F_{\delta m}\, F_{Mm}\, F_{MCm}\, F_{ECm}\, F_{SDm}\, F_{SSm} \qquad (5.4\text{-}9)$$

and

$$\beta_F^2 = \left(\beta_S^2 + \beta_\mu^2 + \beta_{SA}^2 + \ldots + \beta_{SS}^2\right)^{\frac{1}{2}} \qquad (5.4\text{-}10)$$

The logarithmic standard deviation β_F is further divided into random variability, β_R, and uncertainty, β_U. To obtain the median ground acceleration capacity A_m the median factor of safety, F_m, is multiplied by the safe shutdown earthquake peak ground acceleration.

For equipment and other components, the factor of safety is composed of a capacity factor, F_C; a structure response factor, F_{RS}; and an equipment response (relative to the structure) factor, F_{RE}. Thus,

$$F = F_C\, F_{RE}\, F_{RS} \qquad (5.4\text{-}11)$$

The capacity factor F_C for the equipment is the ratio of the acceleration level at which the equipment ceases to perform its intended function to the seismic design level. This acceleration level could correspond to a breaker tripping in a switchgear, excessive deflection of the control rod drive tubes, or failure of a steam generator support. The capacity factor for the equipment may be calculated as the product of F_S and F_μ . The strength factor, F_S, is calculated using Eq. (5.4-7). The strength, S, of equipment is a function of the failure mode. Equipment failures can be classified into three categories:

1. Elastic functional failures
2. Brittle failures
3. Ductile failures

Elastic functional failures involve the loss of intended function while the component is stressed below its yield point. Examples of this type of failure include the following:

- o Elastic buckling in tank walls and component supports
- o Excessive blade deflection in fans
- o Shaft seizure in pumps

The load level at which functional failure occurs is considered the strength of the component.

Brittle failure modes are those which have little or no system inelastic energy absorption capability. Examples include the following:

- o Anchor bolt failures
- o Component support weld failures
- o Shear pin failures

Each of these failure modes has the ability to absorb some inelastic energy on the component level, but the plastic zone is very localized and the system ductility for an anchor bolt or a support weld is very small. The strength of the component failing in a brittle mode is therefore calculated using the ultimate strength of the material.

Ductile failure modes are those in which the structural system can absorb a significant amount of energy through inelastic deformation. Examples include the following:

- o Pressure boundary failure of piping
- o Structural failure of cable trays and ducting
- o Polar crane failure

The strength of the component failing in a ductile mode is calculated using the yield strength of the material for tensile loading. For flexural loading, the strength is defined as the limit load or load to develop a plastic hinge.

The inelastic energy absorption factor, F_μ, for a piece of equipment is a function of the ductility ratio, μ. The median value F_μ is considered close to 1.0 for brittle and functional failure modes. For ductile failure modes of equipment that respond in the amplified acceleration region of the design spectrum (i.e., 2 to 8 Hz):

$$F_\mu = \epsilon (2\mu - 1)^{\frac{1}{2}} , \qquad (5.4\text{-}12)$$

where ϵ is a random variable reflecting the error in Eq. (4-16) and has a median value of 1.0 and a logarithmic standard deviation, β_U, ranging from 0.02 to 0.10 (increasing with the ductility ratio). For rigid equipment, F_μ is given by

$$F_\mu = \epsilon_\mu^{0.13} \qquad (5.4\text{-}13)$$

Again, ϵ is a random variable of median equal to 1.0 and logarithmic standard deviation ranging from 0.02 to 0.10.

The median and logarithmic standard deviation of ductility ratios for different equipment are calculated considering recommendations of [Newmark, 1977]. This reference gives a range of ductility ratios to be used for design. The upper end of this range might be considered to represent approximately the median value, while the lower end of the range might be estimated at about two logarithmic standard deviations below the median.

The equipment response factor F_{RE}, is the ratio of equipment response calculated in the design to the realistic equipment response; both responses are calculated for design floor spectra. F_{RE} is the factor of safety inherent in the computation of equipment response. It depends upon the response characteristics of the equipment and is influenced by some of the variables listed under Eq. (5.4-8). These variables differ according to the seismic qualification procedure. For equipment qualified by dynamic analysis, the important variables that influence response and variability are as follows:

- o Qualification method (QM)
- o Spectral shape (SA) - including the effects of peak broadening and smoothing, and artificial time history generation
- o Modeling (affects mode shape and frequency results) (M)

- o Damping (δ)
- o Combination of modal responses (for response spectrum method) (MC)
- o Combination of earthquake components (EC)

For rigid equipment qualified by static analysis, all variables, except the qualification method, are not significant. The equipment response factor is the ratio of the specified static coefficient divided by the zero period acceleration of the floor level where the equipment is mounted. If the equipment is flexible and was designed via the static coefficient method, the dynamic characteristics of the equipment must be considered. This requires estimating the fundamental frequency and damping, if the equipment responds predominantly in one mode. The equipment response factor is the ratio of the static coefficient to the spectral acceleration at the equipment fundamental frequency.

Where testing is conducted for seismic qualification, the response factor must take into account the following:

- o Qualification method (QM)
- o Spectral shape (SA)
- o Boundary conditions in the test versus installation (BC)
- o Damping (δ)
- o Spectral test method (sine beat, sine sweep, complex waveform, etc.) (STM)
- o Multi-directional effects (MDE)

The overall equipment response factor is the product of these factors of safety corresponding to each of the variables identified above. The median and logarithmic standard deviations for randomness and uncertainty are estimated following Eqs. (5.4-9) and (5.4-10).

The structural response factor, F_{RS}, is based on the response characteristics of the structure at the location of component (equipment) support. The variables pertinent to the structural response analyses used to generate floor spectra for equipment design are the only variables of interest to equipment fragility. Time-history analyses using the same structural models used to conduct structural response analysis for structural design are typically used to generate floor spectra. The applicable variables are as follows:

- o Spectral shape
- o Damping
- o Modeling
- o Soil-structure interaction

For equipment with a seismic capacity level that has been reached while the structure is still within the elastic range, the structural response factors should be calculated using damping values corresponding to less than yield conditions (e.g., about 5% median damping for reinforced concrete). The combination of earthquake components is not included in the structural response since the variable is to be addressed for specific equipment orientation in the treatment of equipment response.

Median F_m and variability β_R and β_U estimates are made for each of the parameters affecting capacity and response factors of safety. These median and variability estimates are then combined using the properties of lognormal distribution in accordance with Eqs. (5.4-6), (5.4-8), and (5.4-11) to obtain the overall median factor of safety F_m and variability β_R and β_U estimates required to define the fragility curves for the structure or equipment. For each variable affecting the factor of safety, the random (β_R) and uncertainty (β_U) variabilities must be separately estimated. The differentiation is somewhat judgmental, but it can be based on general guidelines. Essentially, β_R represents variability due to the randomness of the earthquake characteristics for the same acceleration and to the structural response parameters which relate to these characteristics. The dispersion represented by β_U is due to factors such as the following:

- Our lack of understanding of structural material properties such as strength, inelastic energy absorption, and damping.
- Errors in calculated response due to use of approximate modeling of the structure and inaccuracies in mass and stiffness representations.
- Usage of engineering judgment in lieu of complete plant-specific data on fragility levels of equipment capacities, and responses.

Information Sources

For structures such as concrete shear walls, prestressed concrete containment, steel frames, masonry walls, field-erected tanks, and buried structures, the fragility parameters are generally estimated using plant-specific information. For major passive equipment (e.g., reactor pressure vessel, steam generator, reactor coolant pump, recirculation pump, major vessels, heat exchangers, and major piping), it is preferable to develop plant-specific fragilities using original design analyses. Because of the large quantities of other types of passive equipment (e.g., piping and supports, cable trays and supports, HVAC ducting and supports, conduit, and miscellaneous vessels and heat exchangers), it is generally necessary to use generic fragilities. For active equipment, use of a combination of generic and plant-specific information is needed to develop fragilities.

Several sources of information are utilized in developing plant-specific and generic fragilities for equipment. These sources include the following:

1. Seismic qualification design reports
2. Seismic qualification test reports
3. Plant safety analysis reports
4. Seismic qualification review team (SQRT) submittals
5. Seismic qualification report summaries
6. Past earthquake experience and expert opinion
7. United States Corps of Engineers shock test reports
8. Specifications for the seismic design of equipment

Sources 1 to 5 are plant-specific; sources 6 to 8 are generic data collected for similar types of equipment.

Equipment fragility development is accomplished by grouping equipment into a number of categories.

In seismic margin studies, a relevant quantity is the HCLPF acceleration capacity of the component. This quantity considers both the uncertainty and randomness variabilities and is the acceleration value for which we have 95% confidence that the conditional probability of failure is less than 5%. That is, it is an acceleration value for the component for which we are highly confident there is only a small chance of failure given this ground acceleration level.

$$\text{HCLPF capacity} = A_m \exp(-1.65\,(\beta_U + \beta_R)) \qquad (5.4\text{-}14)$$

Spectral Shape Factor - F_{SA}

The review earthquake level is specified in this study as the NUREG/CR-0098 median ground response spectrum (rock) anchored to 0.3g pga. It is interpreted as the 84% nonexceedance spectrum for Maine Yankee. For calculating the median capacity and variability, we need to estimate the median spectrum for Maine Yankee and the variability in the spectral ordinates. It is interpreted that the review earthquake level (target spectrum) is a 84% nonexceedance level, in that a future earthquake will exceed the target spectrum only if 16% of the spectral ordinates exceed the target over the frequency range of interest.

In calculating the spectral shape factor F_{SA} we need to consider the peak-to-peak response spectrum variability which reflects the observation that response spectra in each horizontal direction for real earthquakes have hills and valleys relative to a mean spectrum. In other words, the spectral ordinates at a given frequency will occur randomly either above or below the mean spectrum.

The peak-to-peak variability for multidegree of freedom structure is different from a single-degree of freedom structure. In general, the logarithmic standard deviation due to peak-to-peak variability is less for a multidegree of freedom structure.

In fragility analysis, the response spectrum logarithmic standard deviation, assuming independent dynamic modes, is given by the following equation [Reed et al., 1986]:

$$\beta_x = \left[\sum \left(\frac{R_{im}}{R_{xm}}\right)^4\right]^{\frac{1}{2}} \beta_{pp} \qquad (5.4\text{-}15)$$

where

β_x = logarithmic standard deviation for building response in the x-direction (similar for y-direction)

R_{im} = median building response for the i^{th} mode for an earthquake in the x-direction

R_{xm} = median building response combined for all modes for an earthquake in the x-direction

When a single mode is dominant,

$$\beta = \beta_{pp}$$

The value of β_{pp} has been estimated to be 0.20 [Reed, et al., 1986].

In this present study, β_{pp} is conservatively assumed to be 0.20. Since the review level spectrum is assumed to be at a 84% nonexceedance value, the median spectrum is obtained by multiplying the target spectrum by exp($-\beta_{pp}$).

Therefore, the median spectral shape factor for structural response factor calculation is given by

$$F_{SA_m} = \exp(0.20) = 1.22$$

An additional factor to be considered is the horizontal earthquake direction variability. This occurs since at a given frequency the response spectrum value in one horizontal direction is different from the direction 90 degrees away. If the response of the critical structural element depends equally on the two horizontal earthquake components, then the directional variability has little effect on the response. In contrast, if one direction dominates, then direction variability may significantly effect response variability.

When co-linear responses for two horizontal earthquake directions are combined, the logarithmic standard deviation for total response due to peak-to-peak and direction variability, becomes for the SRSS peak response rule:

$$\beta = \left[\left(\frac{R_{xm}}{R_m}\right)^2 \beta_x^2 + \left(\frac{R_{ym}}{R_m}\right)^2 \beta_y^2 + \left(\frac{R_{xm}^2 - R_{ym}^2}{R_m^2}\right)^2 \beta_\varphi^2\right]^{\frac{1}{2}} \quad (5.4\text{-}16)$$

where

β_φ = logarithmic standard deviation for direction variability from a statistical analysis of the ratios of spectral accelerations for a fixed direction (e.g., north/south) to the corresponding geometric mean values. Since we are assuming that the maximum of the two horizontal components is used in specifying the review earthquake, the resulting distribution of the ratio will be approximately lognormal with a median larger than 1.0.

In the absence of such a detailed study, β_ϕ due to directional effect is judged to be between 0.10 and 0.20 and is taken as 0.15 in this margin study. The total variability due to peak-to-peak variability and directional effects is taken as

$$\beta_{SA} = \sqrt{\beta_{PP}^2 + \beta^2} = \sqrt{0.20^2 + 0.15^2} = 0.25$$

It is assumed that these variabilities (peak-to-peak and direction) are removed from any hazard analysis leading to selection of margin earthquake selected for review. It should be noted that the issue of exceedance of the target response spectrum (e.g., 16% assumed in the Maine Yankee specific spectrum) is not the same as the confidence levels associated with the target spectrum as determined from the hazard curves. In the case of the hazard curves, the confidence is associated with the uncertainty of the underlying geophysical parameters, while the exceedance of the target response spectrum is a randomness consideration associated with the variabilities of earthquake time history "signatures" [Reed et al., 1986].

5.4.1.2 Steel Structures

HCLPF capacities were determined for the following steel framed structures:

- o Circulating water pumphouse steel superstructure
- o Turbine/service building, steel framed portions
- o Main steam valve house steel interior structure

The circulating water pumphouse (Figure 5.4-4) houses the service water pumps and other components of the service water system. This structure is constructed of both reinforced concrete and structural steel framing. In the vicinity of the structure, grade is located at El. 20'-0" and the rock line is at approximately El. 0'-0". The portion of the structure at and below El. 21'-0" consists of thick reinforced concrete walls and slabs. The exterior north, south, and west walls were poured against rock and the exterior east wall faces the Back River. As discussed in Section 4.3.4.1, the concrete portion of the pumphouse is screened out for the 0.3g review level earthquake.

The 12-inch-thick concrete roof slab at El. 37'-0" is supported by structural steel beams and columns. The columns are typically anchored to the concrete floor at El. 21'-0". The steel superstructure is enclosed by metal siding. Resistance to lateral seismic loads is provided primarily by diagonal bracing (Figure 5.4-5). The diagonal braces are built up from two channel sections arranged to form a T shape. A single diagonal brace is provided at each side of the building.

Review guidelines are not established for steel framed structures. Consequently, the circulating water pumphouse steel superstructure was analyzed to determine its fragility. There are no Group A components attached directly to the steel framing. Damage to the Group A service water system components can only occur if the roof slab collapses.

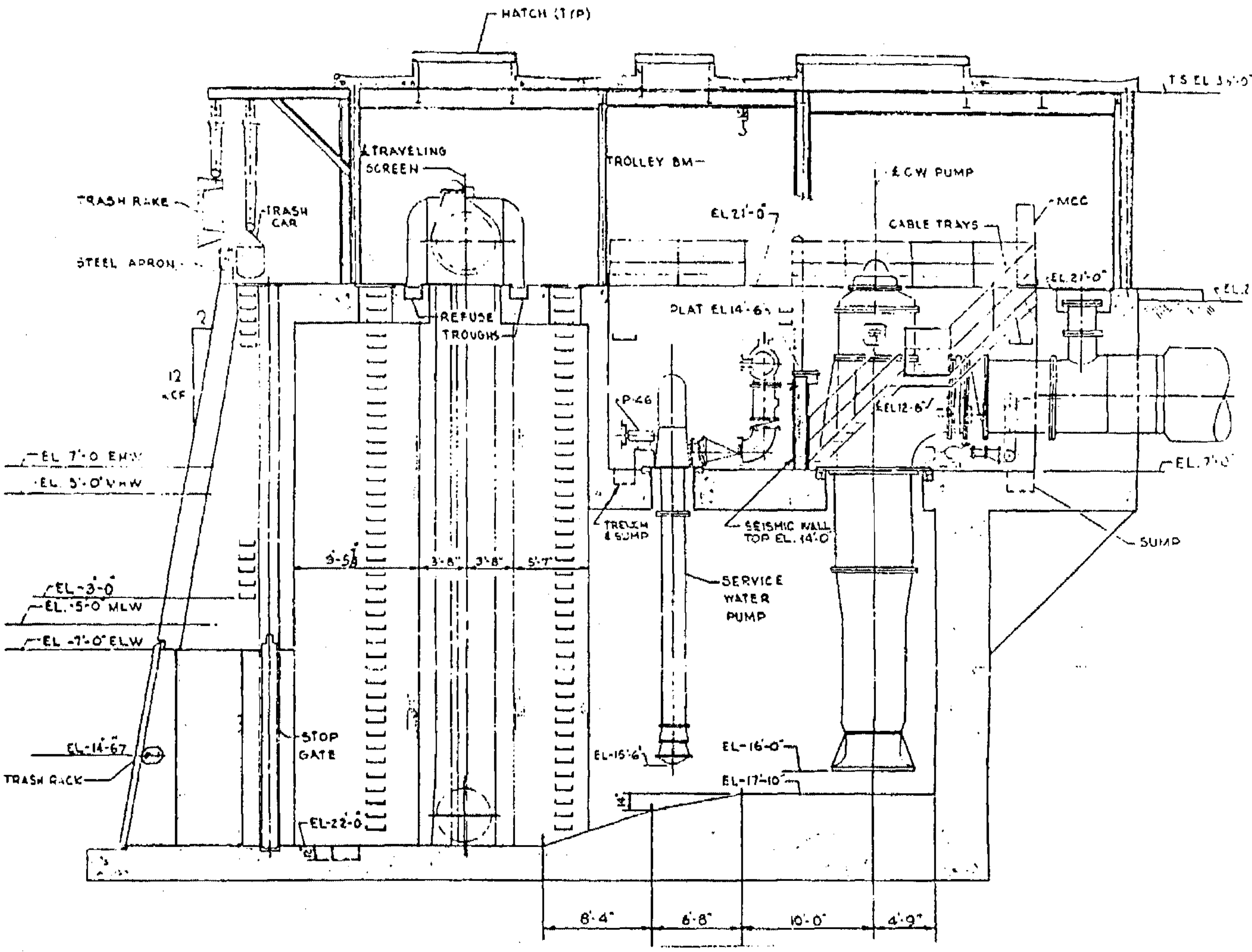
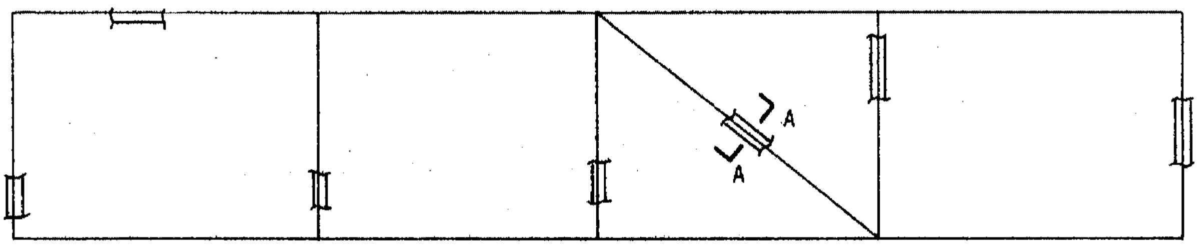
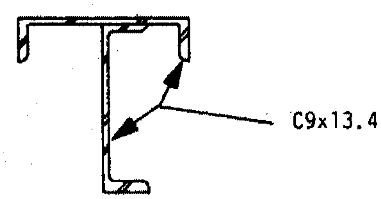


Figure 5.4-4 Circulating Water Pumphouse.





SECTION A-A

Figure 5.4-5 Circulating Water Pumphouse, Typical Braced Frame.

Seismic analysis of the steel superstructure above El. 21'-0" was performed using a single-degree-of-freedom dynamic model. The concrete portion of the pumphouse was considered to be essentially rigid since it is built into rock and the shear walls are very stiff. Mass lumped at the roof level included contributions from the slab, steel framing, siding, and attached equipment. The roof slab was treated as a rigid diaphragm. Horizontal stiffnesses for the dynamic model were based on the stiffnesses associated with the braced frames. Seismic responses in the two horizontal directions were analyzed independent of each other since little coupling is expected.

Fundamental frequencies in the two horizontal directions were both estimated to be about 3.7 Hz. Seismic input to the steel superstructure consisted of the specified review level earthquake ground response spectrum since the concrete structure is essentially rigid. Median damping was estimated to be approximately 15% based upon the recommendations of NUREG/CR-0098 for bolted steel structures at or near the yield point. Resulting overall seismic inertial loads were distributed to the braced frames in proportion to their relative rigidities. Accidental torsional moments were based upon a minimum 5% eccentricity.

A number of potential failure modes were considered in the evaluation of the circulating water pumphouse, including the following:

- o Diagonal brace buckling
- o Column buckling
- o Diagonal brace connection failure
- o Column base connection failure
- o Roof diaphragm shear failure

Buckling of the diagonal braces was found to be the initial failure mode. The results of testing on carbon steel compression members are compiled in [Hall, 1981]. This study derives empirical equations to determine average buckling stresses and associated variabilities. These data were used to estimate the buckling capacity of the diagonal braces and its logarithmic standard deviation. Inelastic energy absorption due to ductile response was not included since the structure is essentially elastic until buckling occurs. A HCLPF capacity of 0.19g was calculated for initial diagonal brace buckling.

As previously noted, damage to the service water system components can occur only if the roof slab collapses. Known damage to steel framed structures in past earthquakes with estimated peak ground accelerations in the range of 0.5g is uncommon. There have only been a few instances of gross collapse of steel framed structures in actual earthquakes of magnitudes larger than expected for the Maine Yankee site. The overall evidence on the performance of steel framed structures in past earthquakes indicates that the collapse capacity should be at least 0.3g. Furthermore, the results of static and dynamic testing demonstrate that additional reserve capacity past buckling is provided by the following two sources:

- o Even in the buckled state, a steel member can still resist axial compression by developing a plastic hinge at midlength. Although the capacity degrades with increased axial displacement, significant displacements in excess of that at buckling are still attainable.

- o The reversing nature of the dynamic seismic input causes displacements to cycle rather than monotonically increase to a total loss of stability.

Based upon past earthquake performance of steel structures and test data, the HCLPF capacity for collapse of the pumphouse steel structure is judged to be at least 0.30g; corresponding estimated fragility parameters are shown in Table 5.5-1.

The turbine/service building (Figures 5.4-6 and 5.4-7) was built as a single integral structure. Most of the Group A components housed in this building are located in the control/switchgear building and the diesel generator enclosure. These portions of the structure are constructed of heavy reinforced concrete shear walls and slabs which have been screened out for the review level earthquake of 0.3g.

Only the steel framed portions of the turbine/service building that could fail with resulting damage to Group A components were subjected to detailed evaluation. These areas included the following:

- o Service building floor at El. 39'-0", bounded by column lines 1/2, 7, C, and F.
- o Service building roof at El. 61'-0"
- o Turbine building floor at El. 35'-0" over the PCC/SCC system

Approximate comparisons of initial load distributions and load carrying capacities indicated that the turbine building diagonal braces may buckle. However, even if this occurs, overall structural integrity of the turbine building will be maintained since lateral displacements will be limited by the massive concrete turbine pedestals once the one inch separation gap with the operating floor is closed.

The portion of the service building floor at El. 39'-0" bounded by column lines 1/2, 7, C, and F was evaluated since its failure may result in damage to the PCC lines running to the chillers. Resistance to seismic load in the N-S direction is provided by diagonal bracing, W36 building columns on Column Line C, diesel generator enclosure shear walls, and control/switchgear building shear walls. The capacity of the load path to the diesel generator enclosure is limited by the shear capacity of the adjacent roof deck. Because the El. 39'-0" floor is discontinuous at Column Line 7, the capacity of the load path to the control/switchgear building is limited by the weak axis bending capacity of the columns on Line 7 which must transmit E-W seismic loads to the floors at El. 35'-0" and El. 45'-0". Accounting for the available load paths and their ductilities, a HCLPF capacity of 0.38g was conservatively estimated.

The steel building roof at El. 61'-0" over the chillers and the turbine building steel framing over the PCC/SCC components at grade both have relatively low masses. They were evaluated using conservative methods. Steel framing in these two areas were found to be capable of maintaining their integrity at peak ground acceleration levels well in excess of 0.3g.

The steel framed structure within the main steam valve house provides support for the main steam lines before they enter containment and associated components. The structure consists primarily of platforms with metal grating supported by steel beams and columns. It is essentially independent from the exterior concrete structure with the exception of three horizontal struts spanning between the steel framing and the east exterior concrete wall. These struts were included as pipe supports only.

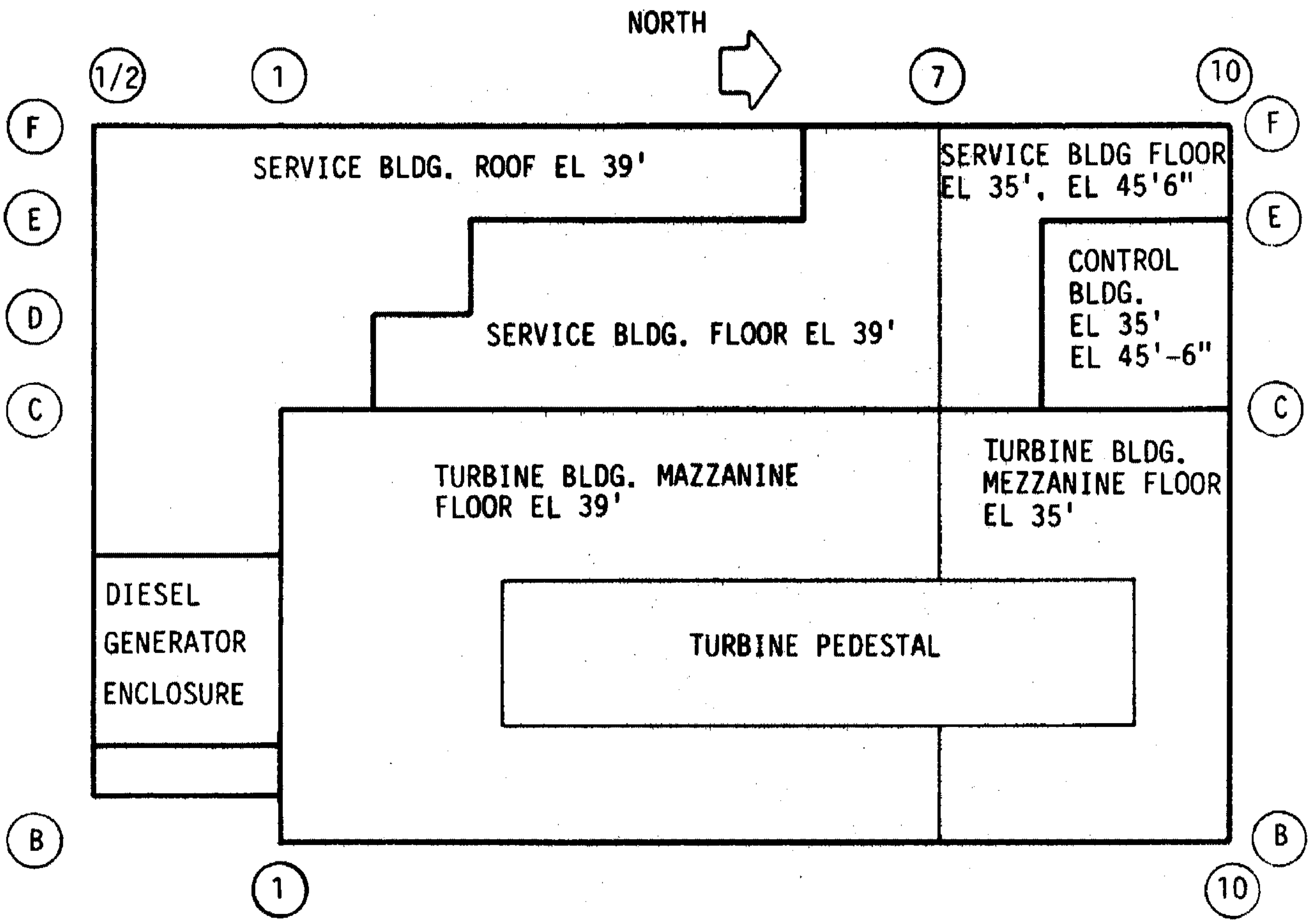


Figure 5.4-6 Turbine/Service Building, El. 35'-0"/El. 39'-0".

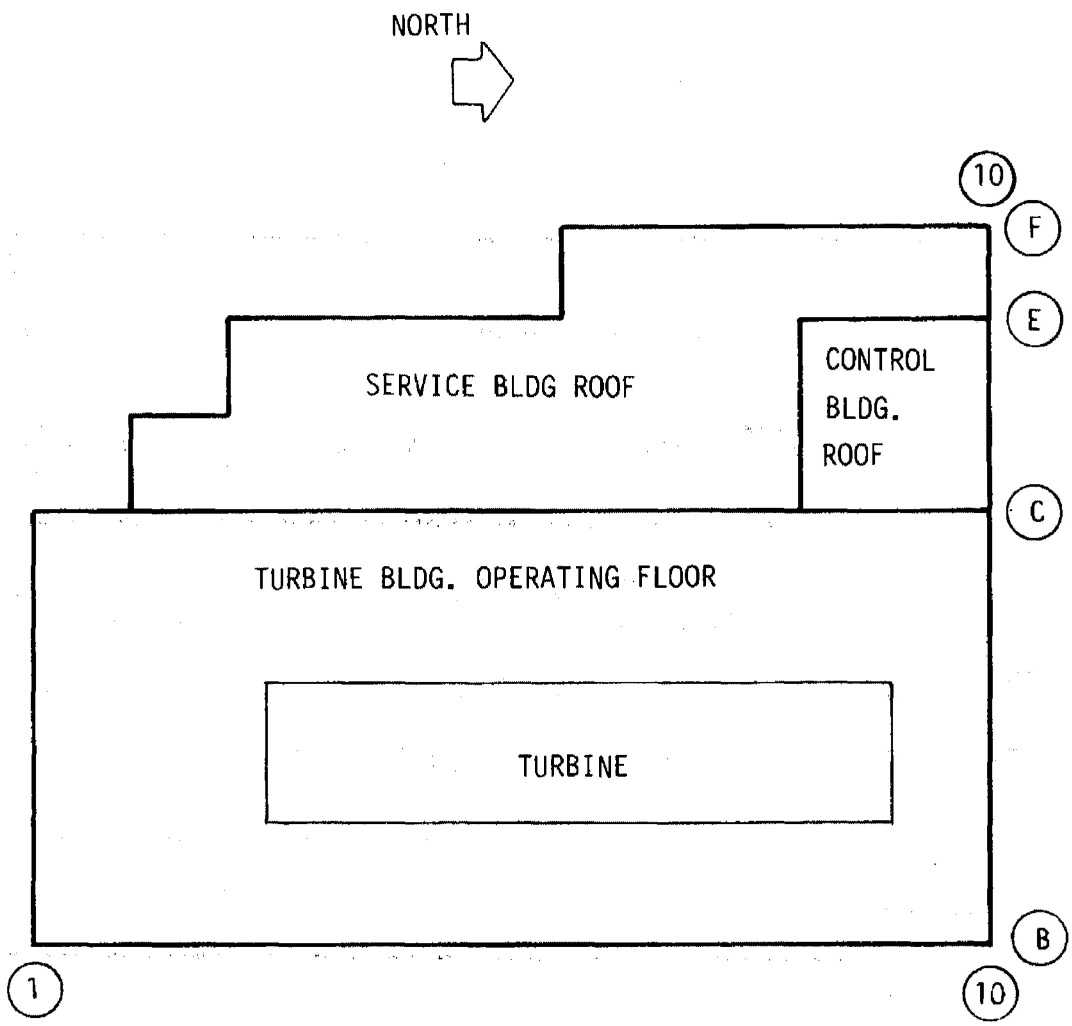


Figure 5.4-7 Turbine/Service Building, El. 61'-0".

Resistance to lateral seismic load is provided primarily by diagonal bracing at each side of the structure. This bracing consists of either double angles or composite channel sections welded together to form a T shape. During the first walkdown, it was noted that the diagonal brace for the north side of the structure at the lowest story was missing. This member was apparently removed due to interference with the access door.

Reevaluation of the steel structure in the as-built condition was performed by Cygna. Results consisted of overall seismic loads from the dynamic model and localized load distributions to individual structural members from more detailed static models. The following potential failure modes were evaluated using conservative methods:

- o Column failure under combined axial load and bending
- o Diagonal brace buckling
- o Brace connection failure
- o Column base connection failure
- o Beam failure due to biaxial bending
- o Pullout of pipe support anchorage

Particular attention was given to members in the vicinity of the missing brace since their loads may be increased to accommodate the load path discontinuity. The HCLPF capacity for this structure was found to be greater than 0.3g based upon conservative methods.

5.4.1.3 Block Walls

A list of block walls supporting or adjacent to Group A components is presented in Chapter 7. These walls are separated into two categories:

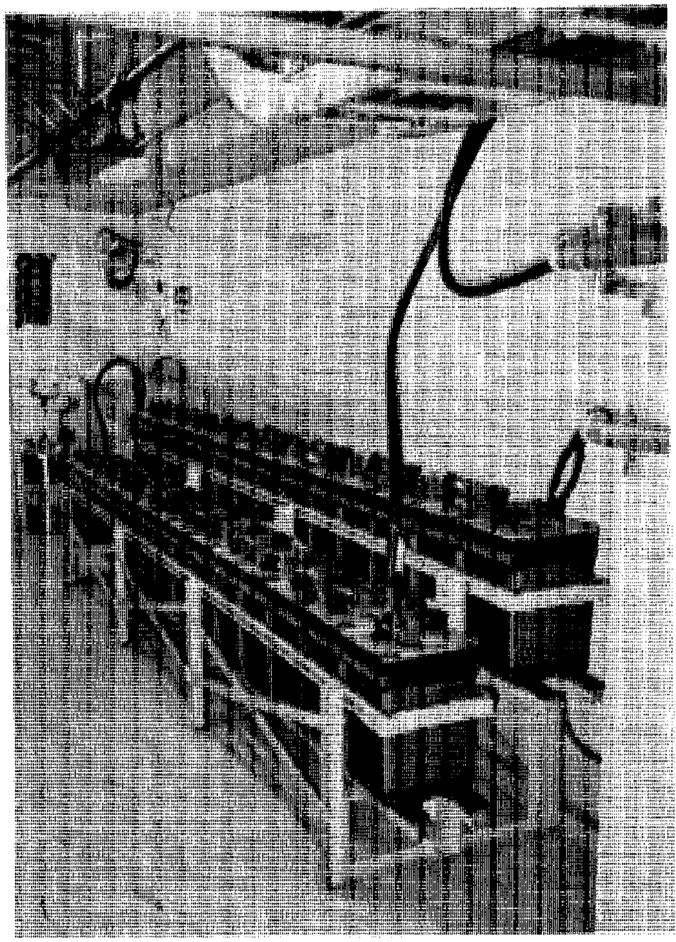
- o Walls that were screened out on the basis that their failure will not damage Group A components.
- o Walls for which HCLPF capacities were determined.

Evaluation of the Group A walls did not identify any with HCLPF capacities less than 0.3g. This can be attributed to the seismic retrofits already installed or to be added during the next outage. Verification that the block wall capacities are greater than 0.3g was performed either by determining the actual fragilities or by utilizing simplified, conservative analyses. As an example, the fragility evaluation of Wall SB 35-3 is described in the following.

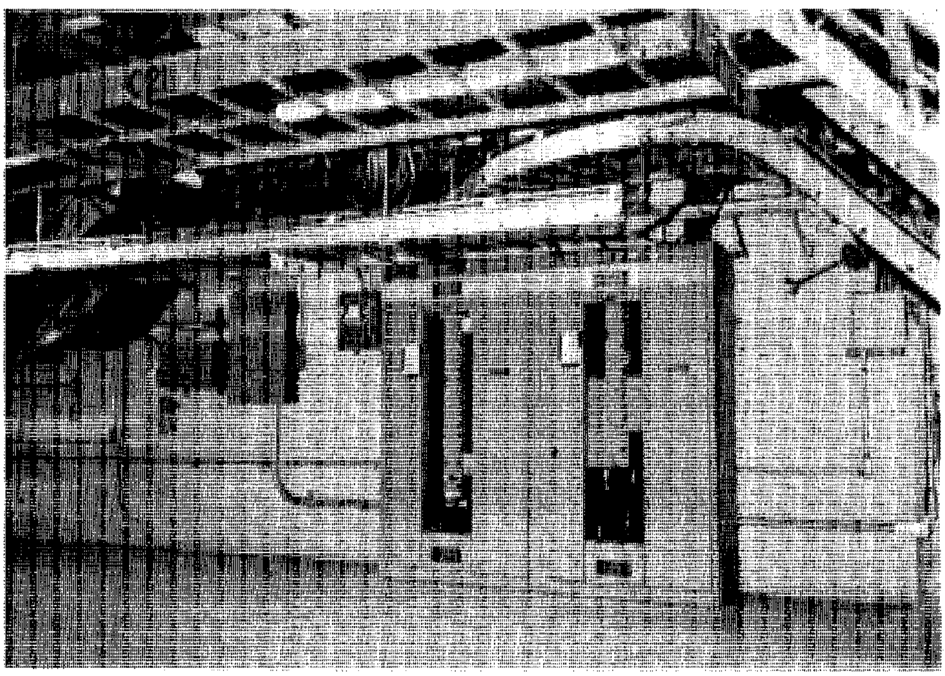
Wall SB 35-3 (Figure 5.4-8) is located at the west side of the service building battery room at El. 35'-0". Out-of-plane collapse of this wall will result in damage to Battery Group 3 mounted on the floor next to the wall. It is built of 12-inch-thick, hollow block units which are assumed to be normal weight. The wall is 7'-0" high by 19'-8" wide. It is mortared up into the overhead concrete beam.

Original block wall construction specifications for Maine Yankee were available. Concrete block units were specified to meet ASTM C90 requirements. The mortar was specified to meet C270, Type M or N requirements. Testing on a limited number of unit and mortar samples taken from the battery room at El. 45'-6" determined the following average compressive strengths:

- o Block unit, f_m= 3800 psi on the net area



a. Looking from the Battery Room



b. Looking from the Cable Spreading Room

Figure 5.4-8 Block Wall SB35-3.

o Mortar, m_o = 2700 psi

The mortar strength would imply the use of Type M mortar at the El. 45'-6" battery room. Because use of Type M mortar throughout the plant could not be confirmed, Type N was assumed for other walls. An average Type N mortar strength of 825 psi was estimated by scaling the results from the test samples.

This wall was analyzed to determine its resistance against out-of-plane seismic response. Because of its low height-to-width ratio, it was analyzed as a one-way member spanning vertically. The top and bottom boundary conditions were represented as simple supports. The fundamental out-of-plane response frequency was calculated using uncracked stiffness properties and found to be into the rigid response range of the floor spectra. The maximum stress due to out-of-plane response was based upon inertial loads due to the floor ZPA and the elastic section modulus. This stress was combined with those due to in-plane and vertical seismic inertial loads by SRSS. Additional in-plane loads may be developed by relative structure displacements between the top and bottom of the wall. However, these loads are stiffness dependent and will tend to dissipate once the wall cracks. Relative structure displacements are small and are considered insufficient to cause wall collapse.

Initial cracking was evaluated by comparing the calculated elastic stress against the median mortar modulus of rupture, including additional resistance provided by dead load compression. Conservative allowable stresses for use in the design of masonry structures are presented in ACI 531. The commentary to ACI 531 notes that factors of safety in masonry are generally held at a minimum of three. Review of available test data indicated that the average factor of safety is greater than this value. The median cracking stress was estimated to be three times the ACI 531 allowable value to account for potential reduction in capacity due to differences between laboratory and field workmanship. A relatively large coefficient of variation of 0.3 was assigned to the cracking stress to account for scatter in the test data, workmanship effects, etc. Inelastic energy absorption effects were not considered since behavior prior to cracking is essentially linear. A HCLPF capacity of 0.60g was estimated for initial cracking of Wall SB 35-3.

Even if the wall cracks through at midheight, stability can be maintained by arching action which has been exhibited in testing of confined walls. The wall will tend to behave as two rigid bodies pivoting about the midheight crack. This rotation causes the wall to push outward against the top and bottom supports, thus developing compression stresses at the points of contact which oppose the transverse loading. The resistance developed by arching action may be limited by compression or shear in the wall, or by bending of the floor beams above and below. An approximate analytical model was developed to account for arching action. While this model was not correlated against test results, it demonstrated that, even using very conservative assumptions, this wall could potentially develop a HCLPF capacity well in excess of 0.3g through arching action.

5.4.1.4 Flat Bottom Storage Tanks

The following flat bottom storage tanks were evaluated:

- o Refueling water storage tank, TK-4 (RWST)
- o Demineralized water storage tank, TK-21 (DWST)

- o Primary water storage tank, TK-16 (PWST)
- o Spray chemical addition tank, TK-54 (SCAT)

While the SCAT is not actually part of the Group A systems, its failure could result in the failure of interconnecting piping to the RWST. Fragility parameters for the RWST, DWST, and PWST are shown in Section 5.5-1. Calculated HCLPF capacities for these tanks are listed in Chapter 7. These capacities were determined using essentially the same methodology for all tanks. A detailed description of the evaluation of the RWST follows.

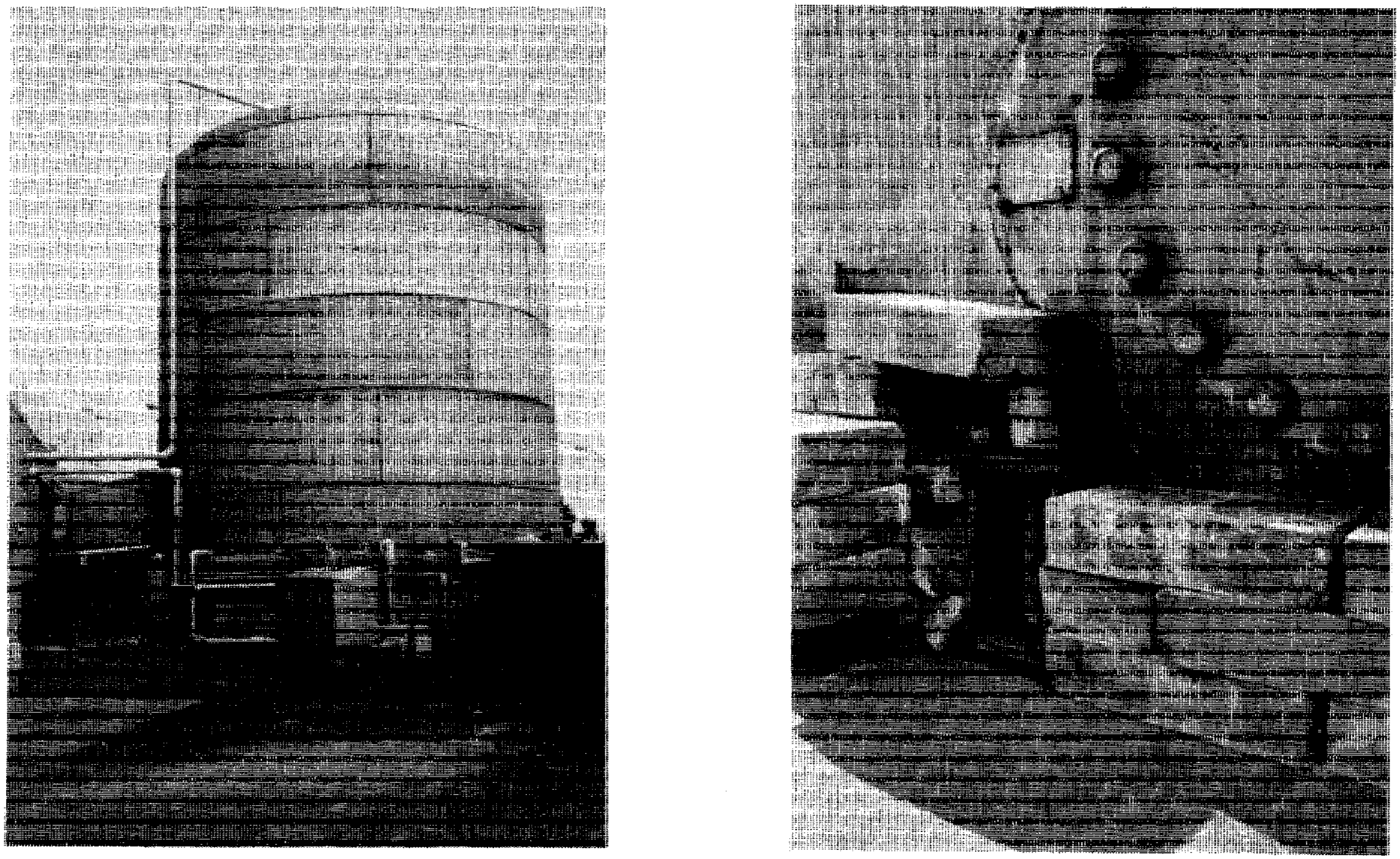
An overall view of the RWST and a typical anchor bolt detail are shown in Figure 5.4-9. The RWST has an inside diameter of 40'-0". The shell has a total height of 38'-0" to the springline and varies in thickness from 3/16 in. to 3/8 in. The spherical shaped head has a total rise of 6'-10" and is 5/16 in. thick. The bottom plate is 1/4 in. thick. The shell, head, and bottom plate are fabricated from A240 Type 304 stainless steel. The RWST is anchored to its concrete ring wall foundation by a total of 8 anchor bolts spaced at 45 degrees. These bolts are 2 inches in diameter and are fabricated from A307 steel. They are attached to the tank shell by chairs built up from stainless steel plate (Figure 5.4-9).

Seismic response of the RWST was calculated following the recommended approach described in NUREG/CR-1161. The following response modes were included:

- o Horizontal impulsive
- o Horizontal convective (sloshing)
- o Vertical fluid
- o Vertical tank

Effective masses and mass centroids to determine overall tank seismic loads due to the horizontal impulsive and convective modes were calculated from equations in NUREG/CR-1161. The impulsive mode fundamental frequency of 5.6 Hz was based upon coefficients developed by [Haroun, 1981]. The frequency of the convective mode was found using equation for rigid tanks. The median impulsive mode damping of 7% was based upon the recommendations of NUREG/CR-0098 for welded steel structures at or near the yield point. Damping of 0.5% was assigned to the convective mode. Overall tank seismic loads due to the impulsive and convective modes were found by factoring the effective masses, mass centroid heights, and spectral accelerations corresponding to the modal frequencies. Contributions from the two horizontal modes were then combined by SRSS. Tank response due to vertical excitations accounted for amplified fluid response associated with tank shell flexibility in the breathing mode and vertical response of the tank itself.

Failure of the RWST is controlled by buckling of the shell due to overall seismic moment at the tank base. Conventional design practice for anchored tanks utilizes a stress distribution at the tank base derived from elementary beam theory. In this stress distribution, plane sections are assumed to remain plane with the neutral axis at the tank centerline under pure bending. This distribution results in unrealistic seismic capacities and is considered to be overly conservative. When the vertical



a. Overall View

b. Anchor Bolt Chair

Figure 5.4-9 Refueling Water Storage Tank.

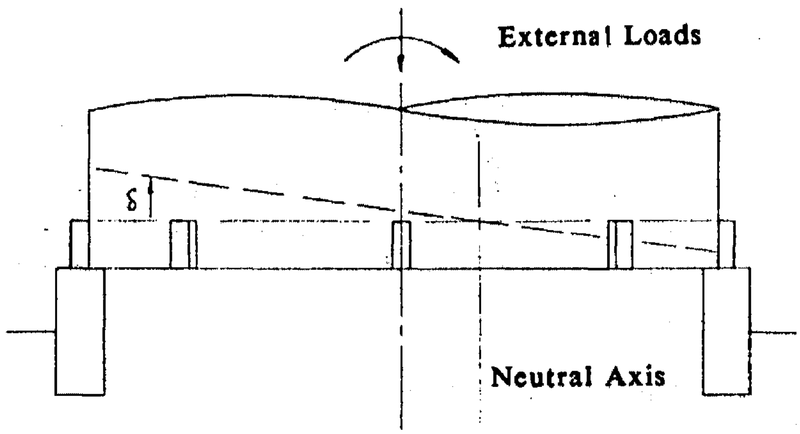
force provided by the tank weight is overcome by the base moment, the tank will uplift and mobilize anchor bolt resistance. Because the anchor bolts are not as stiff as the shell, the neutral axis will shift toward the compression side to maintain equilibrium between the applied and resisting forces. Also, if the anchor bolts or their chairs can deform in a ductile manner, additional load redistribution will occur as uplift increases and anchor bolts around the tank perimeter are progressively stressed past yield. This behavior is acceptable since ductile deformation of the anchor bolts or their chairs will not directly cause a loss of tank contents, unless local deformations become excessive.

To more accurately determine a realistic tank capacity against overall seismic base moment, the analytical model shown in Figure 5.4-10 was developed. Tank deformations at the base and at the elevation through the top of the anchor bolt chairs are assumed to be linearly distributed. This is appropriate since the foundation and the portion of tank above the top of the chairs are stiff compared to the anchor bolts. The distribution of forces at the base of the tank resulting from this assumption is shown in Figure 5.4-10. Compressive strains in the shell are found by distributing the displacements over the height of shell between the tank base and the top of the anchor bolt chairs. Shell compressive stresses are found by factoring the strains by the elastic modulus. So long as the anchor bolts or their chairs are elastic, anchor bolt strains are determined by assuming that the total uplift displacement results in a uniform strain over the length of the bolt from the top of the chair to the anchor plate embedded in concrete.

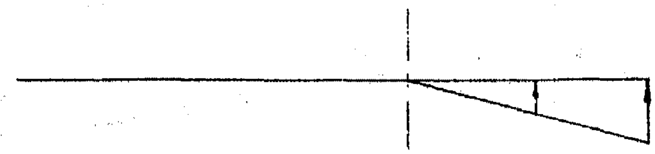
Uplift of the tank base also mobilizes some additional resistance associated with the contained fluid bearing on the tank bottom plate. The fluid hold-down force provided by the tank bottom plate is accounted for in the design of unanchored tanks by modeling the plate as a beam subjected to a vertical displacement at one end and uniform pressure along its length by the fluid weight. A plastic mechanism is assumed to develop in this beam. While this representation is considered to be conservative for unanchored tanks, it may be unconservative for an anchored tank whose uplift displacements may not be sufficient to develop this plastic mechanism. Accordingly, the fluid holdown force was based on the same beam representation for unanchored tanks described in [Wozniak, 1978], but limited to elastic behavior only (Figure 5.4-11). The magnitude of the hold-down force varies around the tank perimeter according to the distribution of uplift displacements. The overall tank base moment capacity is found by limiting the maximum shell compressive stress to the value causing buckling and solving for force equilibrium between the applied loads and resisting forces.

Design buckling stresses for cylindrical shells subjected to bending moment were developed in [NASA, 1966]. The NASA design approach was adapted to predict the median RWST buckling stress. The design buckling stress coefficients in [NASA, 1966] are values having a 90% confidence of exceedance. Based upon buckling stress data reported in [Weingarten, 1965] and [Gerard, 1957], the design coefficients for buckling due to bending are estimated to have a median factor of safety of 1.4. Increase in the tank buckling stress due to internal pressurization by the contained fluid was included. Stainless steel does not have a sharply defined yield plateau, and instead exhibits a rounded stress-strain curve. To account for potential inelastic buckling, the plasticity correction factor was derived from a Ramberg-Osgood-Hill representation of the material stress-strain curve.

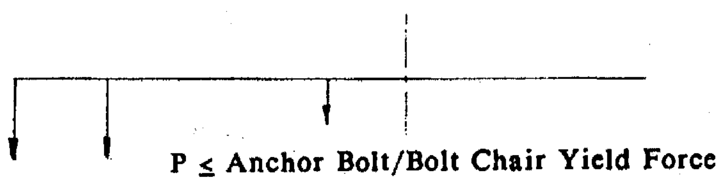
The tensile capacity of the anchor bolts may be limited by any of the following:



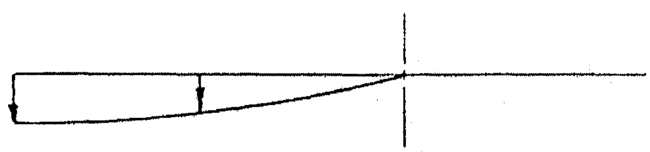
a. Assumed Displacement Distribution



b. Distribution of Shell Compressive Stresses



c. Distribution of Anchor Bolt Forces



d. Distribution of Fluid Holddown Force on Tank Bottom Plate

Figure 5.4-10 Model for Determination of Tank Resistance Against Seismic Base Moment.

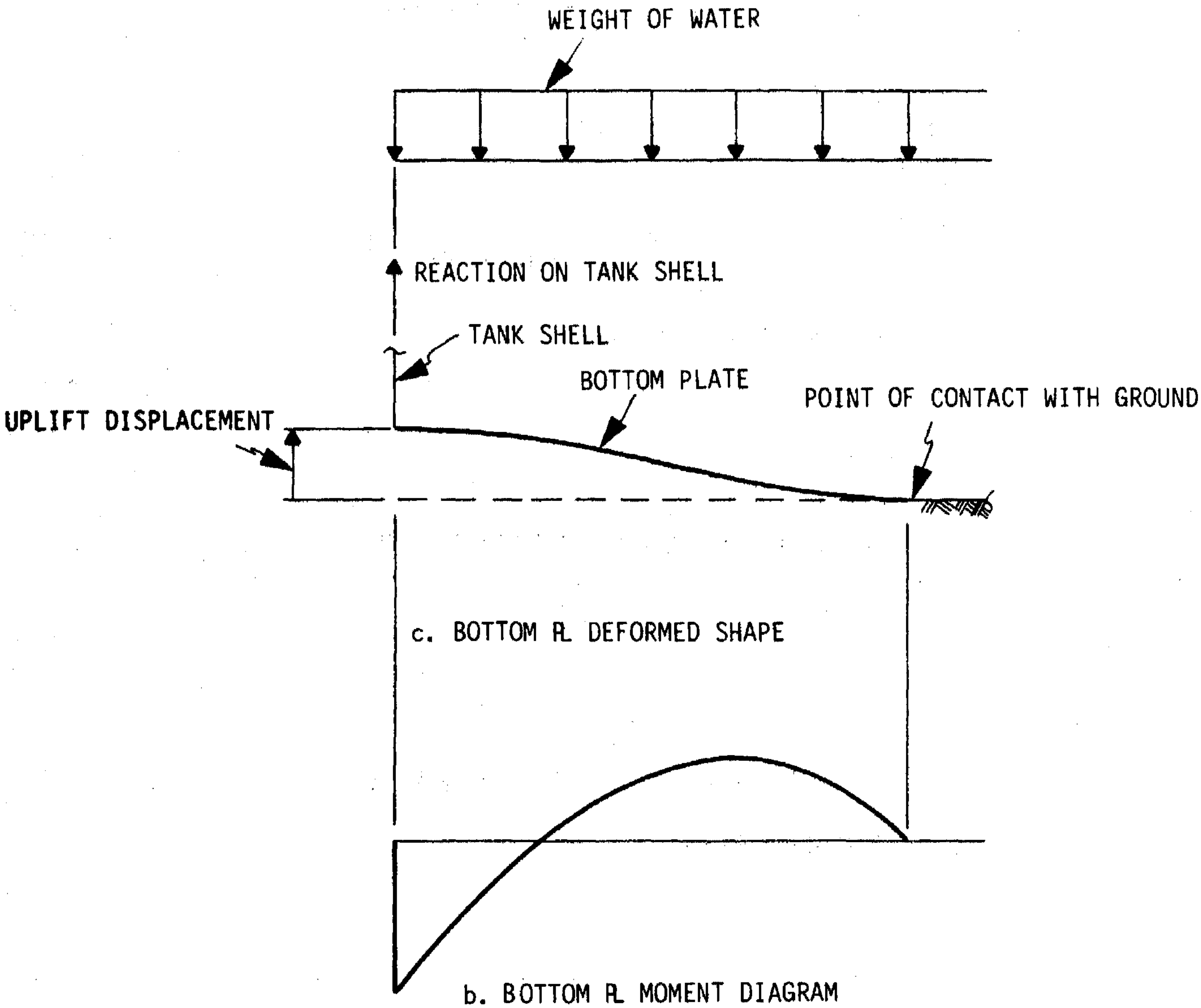


Figure 5.4-11 Beam Model of Tank Bottom Plate.

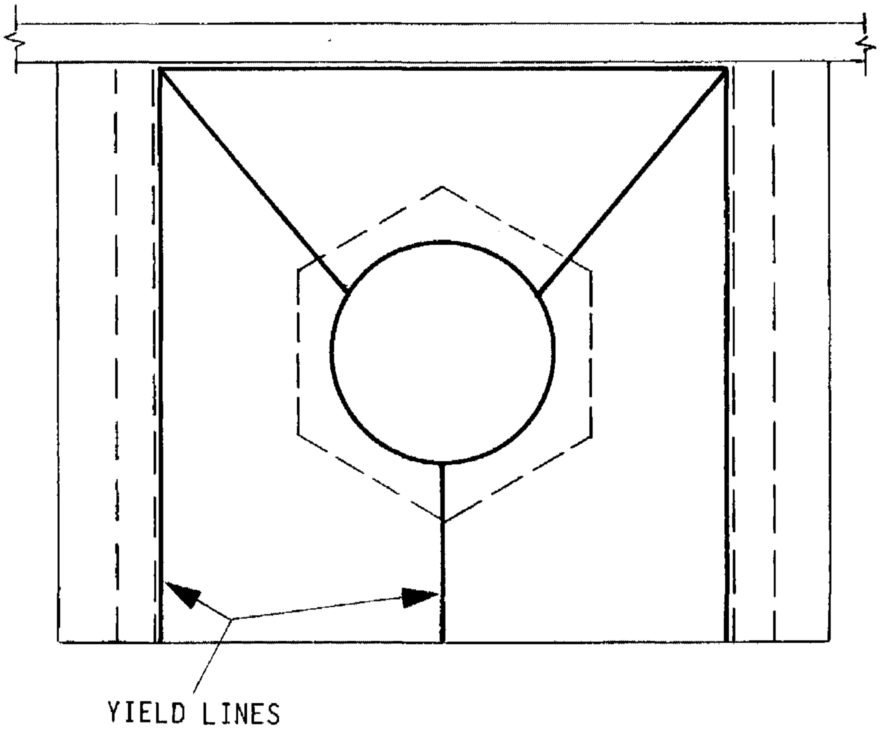
- o Anchor bolt yield
- o Anchor bolt fracture at the threads
- o Anchor bolt pullout from the concrete foundation
- o Yield of the tank shell locally at the anchor bolt chair
- o Bending of the anchor bolt chair top plate
- o Failure of anchor bolt chair welds
- o Stability of the anchor bolt chair stiffener plates

Based upon a review of these items, capacity of the anchor bolts was found to be limited by bending of the anchor bolt chair top plate. Under applied loading imposed by the anchor bolt, the top plate is subjected to bending as it spans between its transverse supports consisting of the vertical stiffener plates and the tank shell. The available resistance was based upon yield line theory with yield lines developing as shown in Figure 5.4-12. At the supports, hinge moment capacities are governed by the bending capacities of the vertical stiffener plates and the tank shell since they are thinner than the top plate itself.

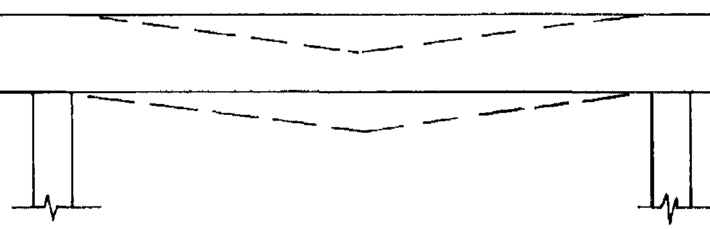
A240 Type 304 stainless steel has a minimum specified yield strength at 0.2% offset of 30 ksi and a minimum specified tensile strength of 75 ksi. The minimum elongation is specified to be 40%. Based upon values reported by [Smith, ASME] median yield and tensile strengths are estimated to be 37 ksi and 84 ksi, respectively. Because stainless steel does not exhibit a sharply defined yield plateau characteristic of carbon steel and because of the large strain ductility available, derivation of plastic moment capacities from the yield strength would be overly conservative. Instead, an effective yield stress was estimated. The moment-curvature relationship was derived for a flat plate using a Ramberg-Osgood-Hill fit to the median strength properties. From a review of the resulting curve, an effective yield stress of 69 ksi was judged to provide a reasonable approximation to the moments at the relatively high curvature demands to which the top plate is likely to be subjected. The median anchor bolt tensile capacity as limited by bending of the chair top plate was estimated using this effective yield stress in conjunction with the yield line representation.

Following this methodology, the RWST was found to have a HCLPF capacity of 0.21g against shell buckling. This capacity included a median system ductility of 1.1 to account for some limited benefit of inelastic energy absorption due to nonlinear response of the tank as the anchor bolt chair top plates yield around the perimeter. Because the seismic response and capacity of the tank is the same in any horizontal direction, the prescribed ground spectrum which is the larger of the two horizontal components is the one that will lead to tank failure. Consequently, values for F_{φ} and β_{φ} of 1.0 and 0, respectively, were used. Other potential tank failure modes considered and found to have greater capacity than shell buckling included sliding at the base and yield of the shell due to hoop stress induced by hydrostatic and hydrodynamic pressure.

The other flat bottom storage tanks were evaluated using essentially the same methodology. Although the DWST is smaller than the RWST and has the same type of anchorage, its HCLPF capacity is lower because it is fabricated from B209-5052-O aluminum. The PWST is similar to the DWST, but it is anchored by twenty anchor bolts, one inch in diameter.



a. YIELD LINE PATTERN



b. DEFORMED SHAPE

Figure 5.4-12 Yield Line Model of Anchor Bolt Chair Top Plate.

5.4.1.5 Inverter

The Maine Yankee inverters are located in the protective switchgear room of the turbine/service building at elevation 45'-6" (25 ft. above grade). The inverters are 12-kVA static inverters manufactured by Solid State Controls Inc. (SCI). They are constructed of an all welded steel tubular frame with bolted shear panels on three sides and bolted doors on the front. The back panels have ventilated slits to provide cooling for the inverter internals. SCI provided the initial qualification of the inverters by comparing the Maine Yankee 12-kVA units to a larger 15-kVA unit shake table tested as part of the qualification program for the Three Mile Island Nuclear Station (reference Maine Yankee FSAR Amendment No. 35, Volume II). The tested 15-kVA inverter was similar in configuration and size to the Maine Yankee inverter, but approximately 1000 pounds heavier. Specific test data was not provided such that a fragility evaluation could be extracted from the tests. The results however, did confirm that the critical failure mode of the inverter is its anchorage. The test results further confirmed the high capacity of the internals and attached components. The inverters were also compared to the GERS, October 1986 draft which supported the capacity of the internals and attached components.

The Maine Yankee inverters are one electrical component which has undergone an anchorage upgrade since its original installation. The original installation (which is still present) bolted the inverter to four base channels which were in turn was expansion bolted to the supporting floor. In 1983 Maine Yankee installed an anchorage upgrade to the inverters increasing the capacity. Figure 5.4-13 shows two photographs of the inverter and anchorage addition. A conservative static analysis was performed by Maine Yankee in evaluating the anchorage addition. To arrive at a more realistic capacity, a fragility evaluation was performed in the present study using available data and as-built details collected during the walkdowns.

The anchorage of the inverter to the concrete floor was identified as the critical failure mode. The failure mechanism was defined as the existing anchorage in combination with the anchorage upgrade.

Capacity Factor. The failure of the existing anchorage to the concrete floor was determined to be the most critical failure mode. A flexibility study was performed to demonstrate the existing anchorage was much stiffer than the anchorage upgrade. Calculations determined the existing anchorage was 99% stiffer than the anchorage upgrade. The primary area of flexibility found in the anchorage upgrade was the long extension of the angle required to clear the base channels in order to bolt the entire assembly to the floor.

The method used to arrive at the inverter capacity consisted of computing two bounding cases for anchorage failure:

- o Lower bound: The existing 1/2 in. diameter Red Head self-drilling anchors resist the entire seismic load.

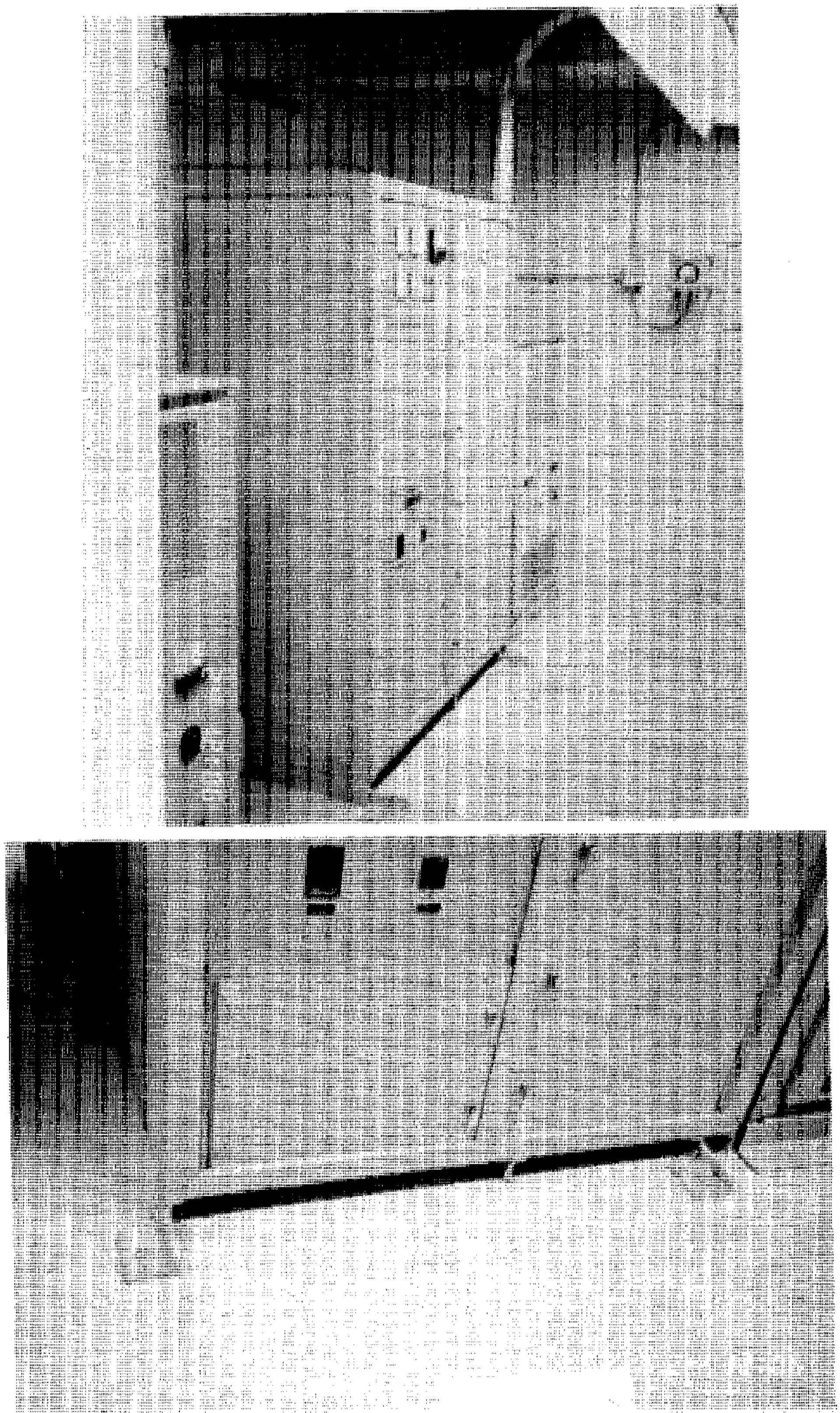


Figure 5.4-13 Maine Yankee 12 kVA Inverter and Battery Charger, and a Close-up of the Anchorage Addition.

- o Upper bound: The existing Red Head anchors combined with the anchorage upgrade fail at the same time.

The actual failure mechanism will lie between these two bounding cases. The theorized failure mechanism will be the existing anchors loading up first since this load path is more stiff. However, some slippage in the existing anchors will occur as ultimate load is reached. This condition will begin to transfer more load into the anchorage upgrade as the displacement increases until failure occurs. It is difficult to ascertain how much slippage (thus load transfer) will occur; consequently, the second bounding condition was assumed. The median failure capacity was assumed to lie more closely to the lower bound, since this is the stiffer load path. Consequently, median capacity was assumed to be defined by the following:

- o A_{upper} represents a $+3\beta$ on A_m (5.4-17)
- o A_{lower} represents a -1β on A_m (5.4-18)

Strength Factor. The inverter was modeled as a single-degree-of-freedom system where the strength factor was computed using static analysis techniques. The turbine/service building floor spectra at 7% damping generated from the 0.18g NUREG/CR-0098 50th percentile ground response spectra was used to obtain the spectral accelerations at the inverters fundamental frequency.

A conservative horizontal fundamental frequency was calculated using only the welded frame and neglecting the bolted shear panels. This yields a lower bound on the median horizontal fundamental frequency. A horizontal fundamental frequency of approximately 13 Hz was computed. This was used for both horizontal directions as the side to side direction was not believed to be significantly stiffer. The rear shear panels were observed to have ventilating slits and the front doors had cut-outs for instruments.

The vertical fundamental frequency of the inverters was judged to be greater than 30 Hz. Vertical response spectra was not developed for points in the floor slab. In order to account for the amplification of the floor spectra at points in the slab, the amplification factors developed from a study by [Structural Mechanics Associates, 1983] were used. The amplification factors are based on the the floor vertical fundamental frequency. A conservative floor vertical fundamental frequency of 18 Hz was computed for the slab in the protective switchgear room which yielded an amplification factor of 1.5. The factor was applied to the ground response spectra at the inverters vertical fundamental frequency to arrive at the spectral acceleration. Since the inverters vertical fundamental frequency was judged greater than 30 Hz, the 1.5 factor was applied to the 0.18 ZPA of the ground response spectra.

The following presents the strength factor computed for the inverter HCLPF capacity determination based on the the combined failure mode of the existing anchorage and the anchorage upgrade. The strength factor discussion below presents the strength factor determination for the existing anchorage, the anchorage upgrade, and the combined strength factor resulting from both.

Existing anchorage strength factor. The existing anchorage consists of the 1/2 in. Red Head self-drilling anchors bolted through the inverter base channels into the concrete floor. The lower bound strength factor determination assumed this failure mode to transfer the entire seismic loading. The seismic loads were applied to the C.G. of the inverter assumed located at the geometric center of the cabinet. This is slightly conservative because the majority of the inverter weight is bolted to the bottom base framing (the heaviest component, transformer, was observed to be positively bolted to the inverter base framing). The seismic loadings were applied in the three orthogonal directions considering 100-40-40 as median centered to result in the most severe loadings. One hundred percent of the east-west seismic accelerations yielded the highest bolt loads. The loads were resolved using standard static techniques to obtain bolt shear and tension loads.

Criteria for determining the strength factor for expansion anchors was that recommended by [URS Corporation/John A. Blume & Associates, Draft 1986]. Shear-tension interaction was based on the quadratic formulation as the inverter bolt loadings were in the area of high tension and low shear ($V/V_{max} < 0.4$). In this range of high tension and low shear the quadratic and the linear formulation produce negligible differences in the results. The median anchor bolt values tabulated in the Blume report were considered median. These values were used only if the Blume criteria was satisfied regarding edge distance requirements and anchor spacing. The uncertainty was computed based on a factor of 2 on the median values as representing a 95% confidence level.

Anchorage Upgrade Strength Factor. The critical failure mode identified for the anchorage upgrade was shear in the fillet weld attaching the angle to the inverter base channels. The fillet weld is heavily stressed due to the configuration of the anchorage upgrade. The angle attaching to the inverter base channels extends out away from the inverter, thus creating a long lever arm which induces shear in the weld.

The strength factor for the weld was computed based on considering median shear strength as 70% of the weld materials ultimate tensile capacity. Median ultimate tensile capacity for material strength was judged to be 20% higher than the code value.

Combined Strength Factor. The combined strength factor considered both of the anchorage systems as resisting the seismic loadings. As discussed previously, the existing anchorage is stiffer than the added anchorage upgrade; consequently, will begin to transfer the loading initially. After slippage occurs in the existing expansion anchors, the anchorage addition will begin to share in the load transfer until idealistically both anchorage systems will fail simultaneously. This represents an upper bound on the actual failure mechanism. The strength factor for this failure mode was computed by adding the individual strength factors computed previously.

The realistic failure mode lies between the lower bound, the existing anchor bolts, and the upper bound where the combined anchorage systems fail together. The median strength factor was computed from Eqs. (5.4-17) and (5.4-18).

Ductility. Inelastic energy absorption was considered as part of the strength factor. Median ultimate material strength properties were used in computing the strength factor. The inverter overall capacity factor, random variability and uncertainty computed are:

$$F_C = 12.26$$
$$\beta_R = 0.00$$
$$\beta_U = 0.46$$

Equipment Response Factors

Qualification Factor. The inverters respond in a single-degree-of-freedom manner. The static analysis used to compute the strength factor was considered median centered with no uncertainties or random variabilities.

Spectral Shape Factor. The unsmoothened and broadened floor response spectra were not provided for the margin study. There is typically conservatism, particularly in the amplified regions of the spectra, in the smoothing and broadening. No attempt was made to recover this conservatism; however, conservatism in the development of the synthetic time history for input to the dynamic structural models for the computation of floor response spectra did occur. Cygna performed the structural modeling and subsequent floor response spectra generation. A synthetic time history was developed such that the response spectra enveloped the 0.18g NUREG/CR-0098 50th percentile ground response spectra. Cygna computed the statistical results of the conservatism in the enveloped spectra over the NUREG spectra. This conservatism was assumed to be included in the floor response spectra. The spectral shape factor used and associated random variabilities and uncertainties derived from the statistical results are:

$$F_{SS} = 1.15$$
$$\beta_R = 0.066$$
$$\beta_U = 0.00$$

Damping Factor. Seven percent damping was considered median centered for welded and bolted structures. The inverter analysis used 7% damping. Five percent damping was considered as representing a minus one beta uncertainty on the median. The random variability was considered to be two tenths of the uncertainty. The resulting values used are:

$$F_D = 1.0$$
$$\beta_R = 0.01$$
$$\beta_U = 0.05$$

Modeling Factor. A median horizontal fundamental frequency was computed in determining the spectral accelerations for the inverter analysis. To establish the uncertainty on the computed median frequency a range of frequencies was considered. A range of 10 to 20 Hz was considered in order to compute the uncertainty. The highest spectral acceleration in the frequency range was considered to represent a 95% confidence on the median spectral acceleration. The random variability was considered to be equal to zero. The modeling factor and the associated random variability, and uncertainty are:

$$F_M = 1.0$$
$$\beta_R = 0.00$$
$$\beta_U = 0.17$$

Mode Combination. The inverter was modeled as a single-degree-of-freedom system. There is some uncertainty as to the contribution of higher modes to the overall response of the inverter not considered in the simplified static analysis. An uncertainty of 0.10 was assumed to reasonably represent the contribution of higher modes. The mode combination factor, random variability, and uncertainty are:

$$F_{MC} = 1.0$$
$$\beta_R = 0.00$$
$$\beta_U = 0.10$$

Earthquake Component Combination Factor. 100-40-40 rule was used to compute the strength factor. This was considered median centered. Random variability based on past PRA studies, considering median coupling and both horizontal directions contributing to failure equal to 0.10 was used. The factor, random variability, and uncertainty are:

$$F_{ECC} = 1.0$$
$$\beta_R = 0.10$$
$$\beta_U = 0.00$$

The combined equipment response factors for the inverter are:

$$F_{ER} = 1.15$$
$$\beta_R = 0.12$$
$$\beta_U = 0.20$$

Structural Response Factors

The structural response factors for the turbine/service building included:

- Soil-structure interaction
- Modeling
- Damping
- Spectral shape
- Directional effects affecting equipment response

Refer to Section 5.4.1.1 for a discussion of the structural response factors. The combined structural response factors used for the inverters are:

F_{RS} = 1.35
β_R = 0.25
β_U = 0.24

<u>Inverter Ground Acceleration Capacity.</u> Combining the capacity factor, equipment response factor, and the structural response factor the resulting median ground acceleration capacity and HCLPF capacity for the inverter was computed as:

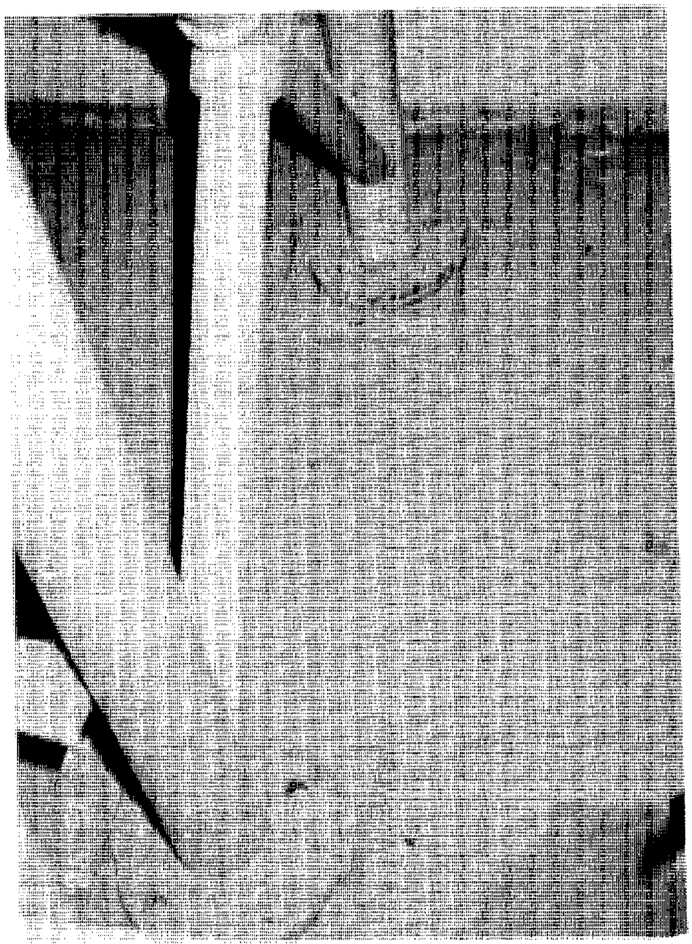
A_m = 3.43g
β_R = 0.28
β_U = 0.59
HCLPF Capacity = 0.82g

5.4.1.6 Diesel Fuel Oil Day Tank

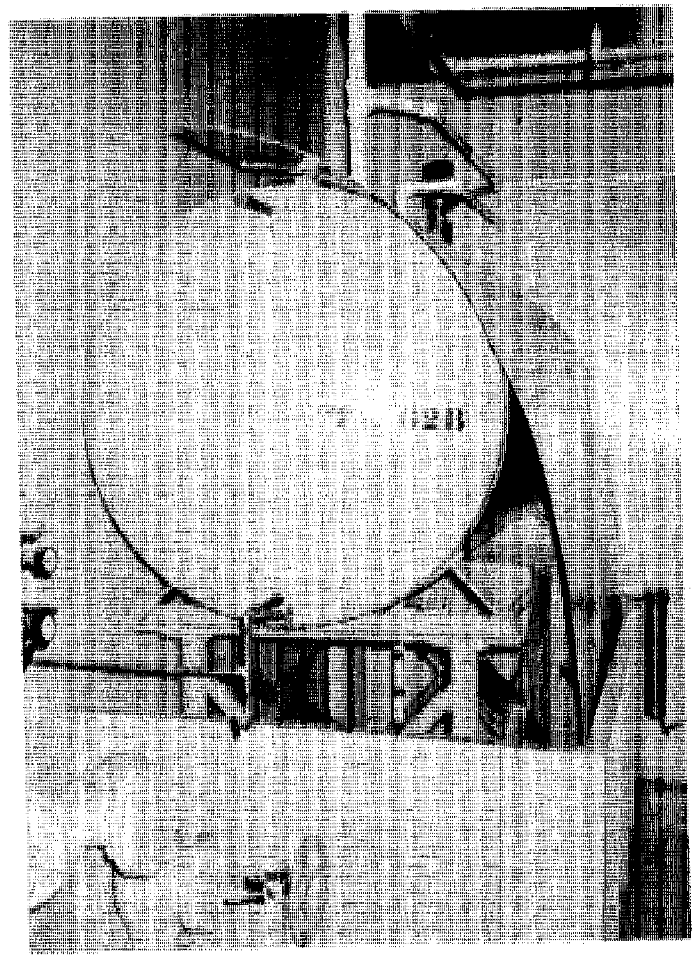
The Maine Yankee diesel fuel oil day tank is a horizontal cylindrical tank supported about 5 feet off the floor by a braced frame support structure. The day tank is located at grade in the diesel generator building. Figure 5.4-14 shows photographs of the tank taken during the first walkdown. The critical failure mode evaluation for the tank was based on a previous finite element analysis [Impell, 1985]. The results of that analysis identified the seven most critical areas of the tank/support system and compared the resulting seismic stresses to ASME Code allowables. Table 5.4-1 specifies the factor of safety (allowable stress divided by the seismic stress) for the seven critical areas assessed. Figures 5.4-15 and 5.4-16 are outline drawings of the day tank and show locations for these seven critical areas which were analyzed. The most critical day tank failure mode is shown from the factors of safety to be the anchor bolts, while the second most critical failure mode is the one-half-inch bolts on the 2" x 2" x 1/4" bracing. The demonstrated factor of safety on the bracing bolts is over three times the factor of safety on the anchor bolts. Thus, the anchor bolts are judged to be the primary failure mode and the contribution to risk of all other failure modes is judged to be negligible due to their much higher capacity. It should be noted that in addition to the areas analyzed in the Impell analysis, the fuel piping (threaded) exiting the day tank bottom was evaluated for displacement induced loads and found to have a relatively high seismic capacity.

<u>Capacity Factor</u>

The capacity factor for the day tank anchor bolts is determined by the ratio of the bolt ultimate failure threshold to the response at the bolt due to the review level earthquake. The capacity factor and its associated variability are described in detail in Section 5.4.1.1. The ultimate failure threshold is based on the anchor bolt test data presented in [URS, 1986]. Close inspection of the Maine Yankee day tank anchor bolts revealed that the expansion anchors utilized for the original plant design were not fully embedded into the floor slab (see Section 4.4 for complete description). Maine Yankee engineers devised a retrofit to ensure that the full strength of these anchor bolts would be developed by installing eight new Hilti Kwik bolts (two per leg) as shown in Figure 5.4-17 (On one tank only six bolts could be installed; The calculated capacity is still greater than 0.3g). The day tank fragility is based on this retrofitted condition.



Day Tank Support Legs
and Original Anchor Bolts



Day Tank with Attached Piping
and Splash Shield

Figure 5.4-14 Diesel Fuel Oil Day Tank (TK-62B).

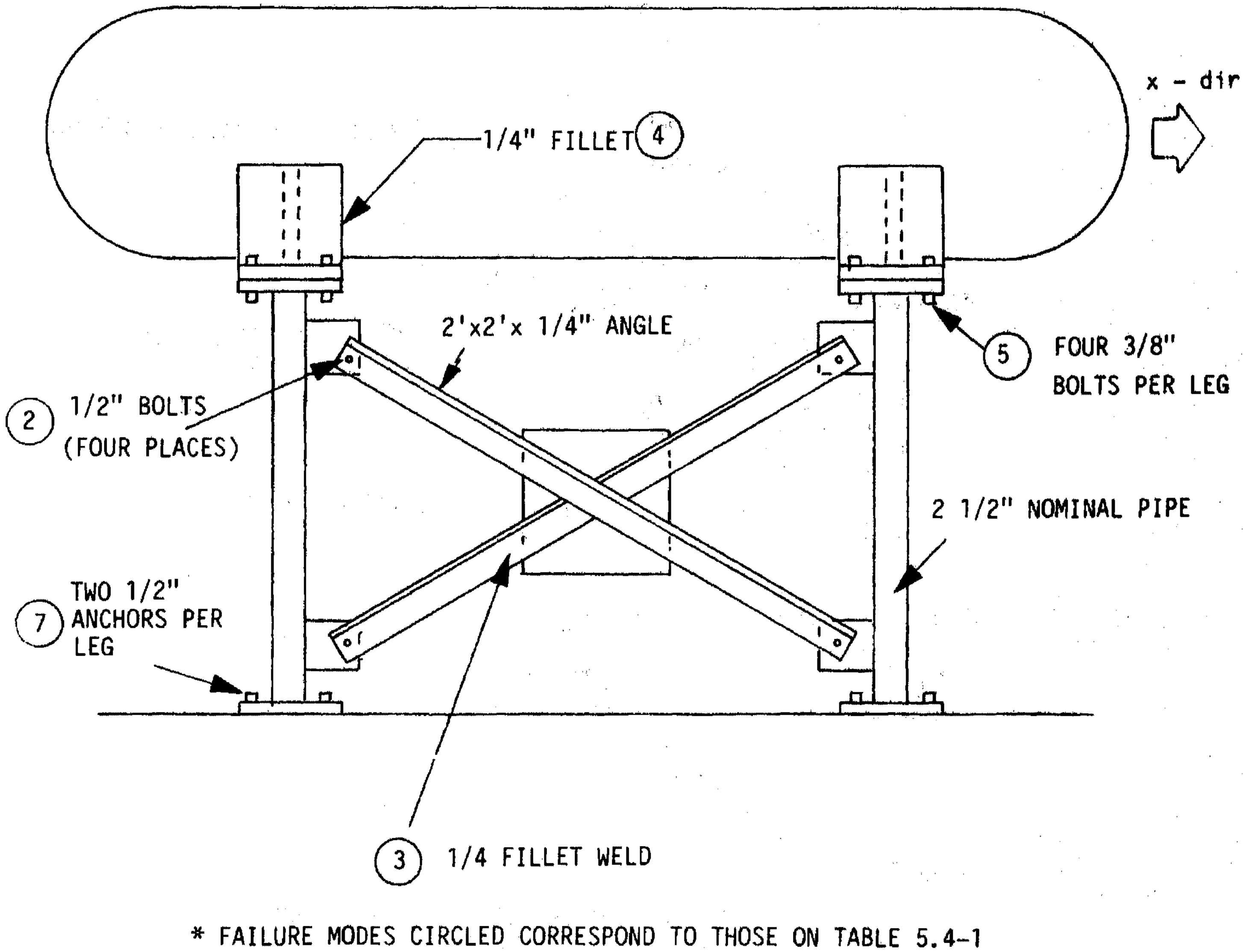


Figure 5.4-15 Diesel Day Tank Critical Areas.

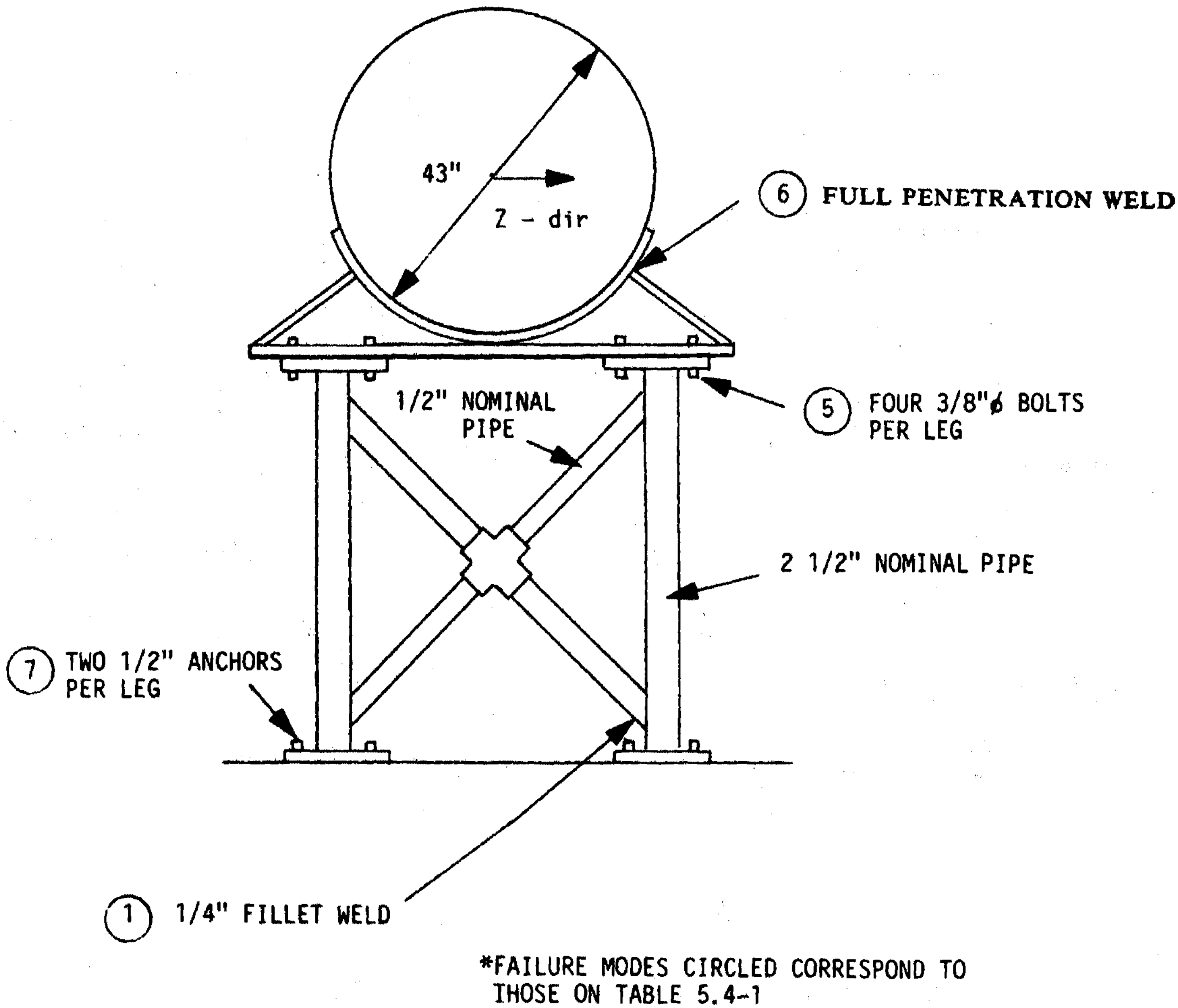


Figure 5.4-16 Diesel Day Tank Critical Areas.

FUEL OIL DAY TANK (TK - 62A,B)

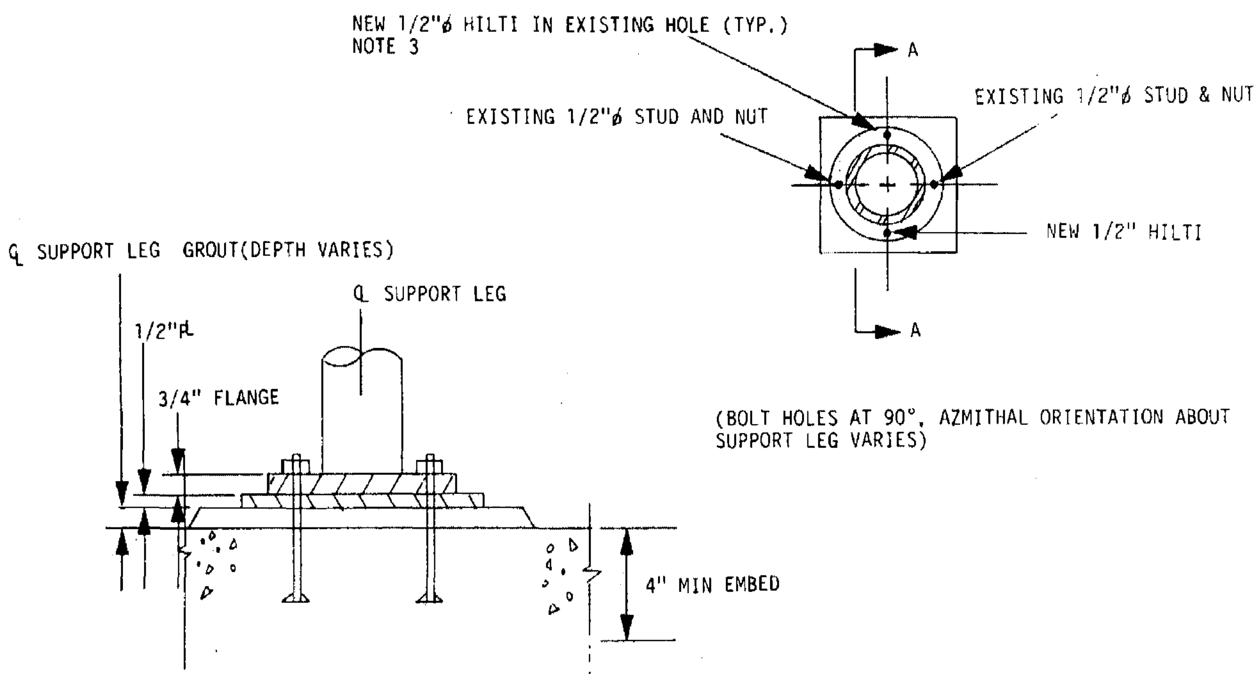


Figure 5.4-17 Retrofit Anchorage Design for the Day Tank.

Table 5.4-1 Diesel Fuel Oil Day Tank Critically Stressed Area

Critical Area	Factor of Safety
1. Welding on 1 1/2" Pipe Braces	5.95
2. 1/2" Diameter Bolts at 2"x2"x1/4" Longitudinal Bracing	3.28
3. Fillet Weld Attaching 2"x2"x1/4" Bracing to the Center Stiffener Plate	High
4. Weld of Tank to Saddle Plates	High
5. Bolting of Vertical 2 1/2" Pipe Posts to the Saddle Plates	2.44
6. Welds at top of Tank Saddle	7.14*
7. Anchor Bolts	1.03

* Conservatively based on a 1/4" fillet weld instead of the actual full penetration weld.

The individual median capacities of the Hilti Kwik bolts are reported in [URS 1986] to be 5.97 kips in tension and 8.80 kips in shear. These values are reportedly based on the 2.25-inch minimum embedment for these bolts. Although the Maine Yankee day tank anchors have a specified minimum embedment of 4 inches, the 2.25-inch embedment strengths are conservatively judged to be applicable for this situation. The reason for the use of the 2.25-inch embedment data is that the relatively close proximity between anchors (4.5 inches between centers) on each day tank leg will lead to some degradation of the rated strength as embedment goes past this 2.25-inch embedment depth. Although the derated strength of a four inch embedded bolt in this situation is greater than the 2.25-inch embedment strength, the 2.25-inch embedment strength has conservatively been assumed for this fragility analysis.

The uncertainty on the median tensile and shear strength values stipulated above is quantified by estimating the 95% probability of exceedance level to be at 50% of the median values. This factor of one-half was judged appropriate as a method of quantifying the various uncertainties associated with expansion anchor capacities (i.e., concrete pours, installation procedures, drill tolerances, dynamic loads and material properties). The shear and tensile loads have been combined using the method recommended in [URS, 1986]. Section 5.4.1.5 describes this interaction equation for the Inverter fragility description.

Response Factors

The day tank natural frequencies were rederived from those presented in the [Impell, 1985] analysis because the weight of the tank contents was neglected in the frequency calculation. This error is significant in calculating the tank frequency and subsequently in deriving the seismic accelerations since the weight of the contents is much greater than the weight of the tank itself. The actual tank frequencies shifted out of the ZPA region and into the amplified section of the spectra (14 Hz and 17 Hz). Spectral accelerations for all three earthquake components were based on the most recent floor spectra (0.18g NUREG/CR-0098) scaled to the 0.3g review earthquake level. Seven percent median damping was used with 5% as the 95% probability of exceedance value. Earthquake components were combined using the median centered 100%, 40%, 40% technique. The remaining equipment and structural response factors were calculated using the methodology presented in Section 5.4.1.1.

Ground Acceleration Capacity

The median ground acceleration capacity of the day tank was calculated to be 1.41g based on the failure of the anchor bolts. The HCLPF capacity is calculated to be 0.43g. Since this HCLPF capacity is greater than the 0.3g review level earthquake for Maine Yankee, the conservative assumption of 2.25-inch embedded anchors is considered acceptable.

5.4.1.7 Containment Spray Fans

The Maine Yankee containment spray fans are located in the ventilation equipment room at elevation 21'-0" (grade). The fans are Westinghouse Silent Vane, size 8030, 13500 cfm units. The fan units are supported from vibration isolators with seismic restraints installed in all directions. The seismic restraints are to be installed during the next Maine Yankee refueling outage (see Section 4.4). Figure 4.4-8 and Figure 4.4-9 show photographs of the fans in the unmodified condition and a sketch of the proposed seismic restraints, respectively. The fragility evaluation was based on this modified condition. The entire fan assembly is anchored to a reinforced raised concrete pad.

The anchorage of the fans was identified as the critical failure mode to investigate and will be presented below. However, another possible failure mode was identified during the walkdown. The fan units have cantilevered shafts. It was postulated that the cantilevered fan blade on the end of the shaft could fail or cause premature bearing failure as a result of seismic excitations. The concern for the cantilever shaft on the containment spray fans was alleviated based on numerous similar configurations which had survived large magnitude earthquakes. Several vendors also verified the fact that this cantilever variety fan is very common and represents approximately 50% of the fans used in power plant applications. These cantilever fans are designed to withstand the high operating loads, and, the vendor felt the safety factor on the operating loads was more than adequate to offset the seismic load.

The containment spray fans anchorage HCLPF capacity was computed using a conservative simplified analysis as outlined in Section 5.2. Several conservative assumptions were used in the analysis to show a high HCLPF capacity. These included:

- The existing anchorage was conservatively assumed to carry no loads.
- Peak spectral acceleration values were used since the vibration isolation system frequency was not known.
- Conservative dimensional data for the anchorage modification was used. Only a conceptual sketch was provided to the fragility team by Maine Yankee for the anchorage modification. Figure 4.4-9 shows the sketch received from Maine Yankee for the anchorage modification. This modification will be verified during the third walkdown.
- Overturning was assumed to be resisted by the outer anchor bolts alone.

Even with these conservative assumptions the fragility evaluation for the containment spray fans showed a HCLPF capacity greater then 0.5g pga which is greater than the margin earthquake for Maine Yankee.

5.4.2 CDFM Method

For the purpose of comparison, HCLPF capacities were determined using the CDFM method for the following selected components:

- o Circulating water pumphouse
- o Block wall SB 35-3
- o Refueling water storage tank

Application of the CDFM method was based on the recommendations in NUREG/CR-4482. HCLPF capacities found by the CDFM and fragility analysis methods are compared in the following discussion. One major difference between the two methods common to all of the examples is the use of the median NUREG/CR-0098 ground response spectrum to determine structure loads and in-structure response spectra for the CDFM evaluation. For the fragility analysis, conservatism in this spectrum was accounted for in the structure response factor described in Section 5.4.1.1. The structure damping value of 7% that was included in the generation of in-structure response spectra is judged to be conservative for the structures considered. Other major differences between the two approaches in their application to the selected examples are discussed below.

5.4.2.1 Circulating Water Pumphouse

Differences between the CDFM and fragility analysis methods for the circulating water pumphouse included the following:

- o Damping
- o Diagonal brace buckling capacity
- o Effective inelastic energy absorption factor

Seismic response of the steel superstructure was determined using 10% damping. This value is recommended by NUREG/CR-0098 as being a conservative value for bolted steel structures at or near the yield point. The diagonal brace buckling capacity was determined using two different formulations:

- o Factored AISC allowable
- o Conservatively biased value based on test data

The AISC allowable buckling capacity for working stress design was factored by 1.7 as permitted by Section 2 of the AISC specifications for plastic design. Alternatively, the test data in [Hall, 1981] used to establish the median capacity was reviewed to establish the 84% exceedance value. These two formulations were found to provide essentially the same conservative capacity.

The HCLPF capacity against initial brace buckling was calculated to be 0.22g. This value is higher than the corresponding capacity of 0.19g determined by the fragility analysis method.

As noted in Section 5.4.1.2, the HCLPF capacity for collapse of the pumphouse steel structure is judged to be greater than 0.3g. This corresponds to an increase factor accounting for the reserve capacity against collapse of 1.35.

5.4.2.2 Block Wall SB 35-3

Conservative seismic response of Wall SB 35-3 for the CDFM method was based upon in-structure response spectra generated for the median NUREG/CR-0098 ground spectrum. A conservative estimate of block wall damping prior to initial cracking was not required since elastic response of this wall occurs in the high frequency range of the floor spectra. The primary source of conservatism in the HCLPF capacity is the selection of a conservative modulus of rupture. The Standard Review Plan permits the use of ACI 531 allowable block wall stresses with certain increase factors for extreme load combinations. However, no increase is permitted for the modulus of rupture perpendicular to the bed joint. This is considered to be overly conservative since ACI 531 itself permits a one-third increase in working stress allowables when stresses due to earthquake loading are included. Review of available test data indicates that the increase factor should be at least 1.33. The modulus of rupture perpendicular to the bed joint was conservatively based upon the ACI 531 allowable value increased by a factor of 1.33. The HCLPF capacity using the CDFM method was determined to be 0.67g. This value is greater than the capacity of 0.57g found by the fragility analysis method.

5.4.2.3 Refueling Water Storage Tank

Major differences between the CDFM and fragility analysis methods in the determination of HCLPF capacities for the refueling water storage tank were associated with the following quantities:

- o Damping
- o Shell buckling stress
- o Material and component strengths
- o Effective ductility

A conservative damping value of 5% was estimated for the horizontal impulsive mode response. This is the value recommended by NUREG/CR-0098 for welded steel structures at or near the yield point. The same analytical model developed in the fragility analysis to determine the resistance against overall seismic base moment was used. As in the fragility analysis, input to this analytical model consisted of the shell buckling stress, bending capacity of the anchor bolt chair top plate, and fluid holdown force on the tank bottom plate. The shell buckling stress was determined using the NASA design coefficients which have a 90% confidence of exceedance. Plastic moment capacities for the anchor bolt chair top plate were based on an effective yield stress taken as the average of the minimum specified yield and tensile strengths. This value corresponds to an equivalent elastic-perfectly plastic representation of the material stress-strain relationship. Additional conservative bias was introduced into the top plate capacity to account for uncertainty in the location of the loading imposed by the anchor bolt nut and washer. Additional seismic capacity implied by nonlinear behavior of the tank as it uplifts and yields the bolt chairs prior to shell buckling was conservatively neglected. A HCLPF capacity of 0.21g was determined for the RWST using the CDFM method. This value is the same as the capacity determined by the fragility analysis method.

[Manos, 1986] has developed an empirical method for the seismic design of unanchored tanks. Manos' comparison of his approach with data compiled on damage and undamaged tanks from actual earthquakes indicates that this method is CDFM equivalent. HCLPF capacities for the RWST, DWST, and PWST based upon Manos' approach are listed in Chapter 7. For example, a capacity of 0.24g is determined for the RWST. This capacity is greater than the 0.21g HCLPF capacity found by the fragility analysis and CDFM methods. Intuitively, the seismic capacity of an anchored tank would be expected to be greater than that of an unanchored tank. However, the mechanics of seismic resistance of unanchored tanks is not well understood at this time. Because the nature of seismic resistance of anchored tanks may be much different than the source of resistance for unanchored tanks, it is not possible to derive firm conclusions on the HCLPF capacities of the Maine Yankee tanks based upon Manos' work.

5.5 HCLPF Capacity of Plant

5.5.1 Accident Sequences

Following the review of plant information, a list of the components that make up the front-line and support systems required to perform the plant safety functions (Group A functions) was developed. This list is given in Chapter 7 and provided the basis for the remainder of the margin review. After the plant walkdowns and subsequent analyses by the systems analysts and fragility analysts, the list of components was reduced by screening out those components that were found to have HCLPF capacities greater than 0.30g pga. The components for which fragilities and HCLPF capacities were determined are discussed in Section 5.4.

These remaining components were used in the development of the event trees and fault trees for the seismic induced core damage accident sequences as described in Volume 2. The event trees and fault trees were analyzed to determine the cut sets for each accident sequence that could lead to core damage. From these cut sets, the Boolean expressions for the accident sequences were developed. The component failures that are significant to these accident sequences are given in Tables 5.5-1 and 5.5-2. Table 5.5-1 gives the seismic induced failures along with the fragility parameters used to represent their capacity. Table 5.5-2 gives the non seismic failures and their failure probability. Note that the component items and the non seismic failure events have been numbered consecutively; the missing numbers represent the items that were screened out in the final pruning of the event and fault trees.

The Boolean expressions for the two dominant accident sequences are:

Small LOCA

$$= 4 + 7 + 20 \qquad (5.5\text{-}1)$$

No LOCA

$$= (4 + 20) * (8 + 15 + 17 + 22)$$

$$+ 8 * (14 + 16) + 7 * 15 \;, \qquad (5.5\text{-}2)$$

where each number in these expressions corresponds to the failure of a component given in Table 5.5-1 or Table 5.5-2. In the above expressions, the notation "+"

Table 5.5-1 Component Seismic Fragility Parameters

Item No.	Item	$A_m(g)$	β_R	β_U	HCLPF Capacity(g)
4	Transformers	0.84	0.30	0.32	0.30
7	RWST	0.45	0.20	0.25	0.21
8	DWST	0.36	0.20	0.26	0.17
20	Circulating Water Pumphouse	0.69	0.24	0.27	0.30
21	PWST*	0.57	0.20	0.26	0.27

* See Page 5-67

Table 5.5-2 Probabilities for Nonseismic Failures

Item No.	Description	Median Unavailability (per demand)	Error Factor*
10	Operator Failure to Close PCC Isol. Valves	8.0E-02	2
11	Random Failure of DG-1B	4.2E-02	5
12	Random Failure of DG-1A	4.2E-02	5
13	Operator Failure to Place AFW Pump Train B in Service Locally	1.5E-01	2
14	Nonseismic Common Cause Failure of DGS	1.6E-03	5
15	Nonseismic Common Cause Failure of AFW	1.2E-04	5
16	Operator Failure to Refill DG Fuel Tanks by Opening Valve or Running P-33A,B	8.0E-03	3
17	Operator Failure to Place AFW Pump Train B in Service from MCR	4.0E-02	3
22	Random Failure of the Turbine Driven Aux. Feedwater Pump	3.0E-02	5

* Error factor equals (95% Confidence Value/Median Value).

denotes probabilistic addition (union) and "*" indicates probabilistic multiplication (intersection).

Because the impulse lines inside the containment could not be inspected to confirm that small LOCA cannot occur, the Boolean expression for small LOCA and No LOCA were combined in two different ways.

The first method uses split fractions to express the conditional probability of a seismic induced small LOCA given the seismic event. The two Boolean Eqs. (5.5-1) and (5.5-2) are combined using the split fraction p, and sensitivety studies are performed on this split fraction.

Core Damage

= p. [Small LOCA] +

(1 - p) . [No LOCA] (5.5-3)

The second method uses an additional term in the small LOCA Boolean expression that represents the initiating event of seismic induced small LOCA and the other terms in Eq. (5.5-1) represent the failure of mitigating systems. The core damage Boolean expression is then the logical combination of the Boolean expressions for two accident sequences i.e., [Small LOCA] * [Small LOCA Boolean Expression] + [No LOCA] * [No LOCA Boolean Expression]. The HCLPF capacity of the plant against core damage is a function ofthe HCLPF capacity against small LOCA (initiating event) if this capacity is less than 0.21g. The plant level core damage HCLPF capacity is governed by the HCLPF capacity of RWST. If the small LOCA HCLPF capacity is larger than 0.21g, the plant level HCLPF capacity is controlled by the small LOCA HCLPF capacity. The HCLPF capacity against small LOCA was not determined since we could not access containment. However, the plant level HCLPF capacity is expected to be greater than 0.21g.

In the following, the quantification of these Boolean expressions for the purposes of deriving the plant HCLPF capacities is discussed.

5.5.2 Probabilistic Method

Following the rules of Boolean algebra, the component fragilities are combined using the numerical procedure proposed by Kaplan (1981). For this purpose, the component fragilities are discretized into a family of fragility curves with a subjective probability estimated for each curve as discussed in Section 5.4.1. The probability assigned to each curve is developed based on the uncertainty distribution on the median capacity. Each fragility curve is assumed to be completely described by the median capacity and the value of β_R. The fragility curve is not truncated in either the lower or upper tail.

The discrete probability distribution (DPD) for a component, say 4 , is expressed as:

$$\{ < q_i^4 , f_i^4 , (a) > \} \qquad (5.5\text{-}4)$$

and is shown graphically in Figure 5.5-1. Here, a is the peak ground acceleration, f_i is the conditional probability of failure of the component under acceleration a and q_i is the subjective probability that the value of f_i is in the range $f_i - \Delta f_i/2$ and $f_i + \Delta f_i/2$. For example, the aggregation of fragilities for two components 4 + 7 is given by a convenient numerical integration scheme [Kaplan, 1981]:

$$\{ < q_{ik}^{4+7} , f_{ik}^{4+7} , (a) > \}, \tag{5.5-5}$$

where

$$q_{ik}^{4+7} = q_i^{4} \; q_k^{7}$$

$$f_{ik}^{4+7} = 1 - (1-f_i^{4})(1-f_k^{7}) \text{ or } \max \{f_i^{4}, f_k^{7}\}.$$

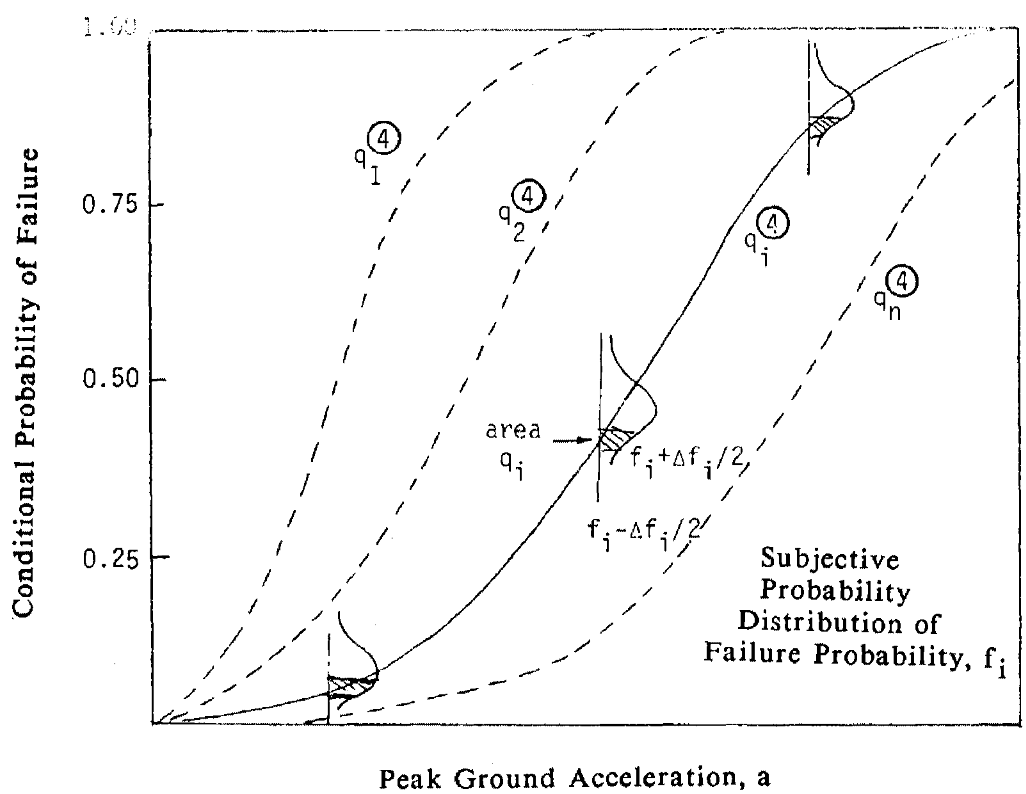


Figure 5.5-1: Seismic Fragility Curves of a Component

In the expression for f_{ik}, two bounds are given corresponding to perfect independence or perfect dependence between component failure events. These bounds are used to assess the significance of correlation between failure events. Expressions similar to Eq. (5.5-5) are derived for the case of joint events 4 * 8 . By taking two components at a time, the plant level fragility is developed

$$\{ <q_i^P, f_i^P, (a)> \}. \qquad (5.5\text{-}6)$$

Figure 5.5-2 shows a plot of the plant level (small LOCA core damage) fragility curves in which the family of fragility curves is reduced to the 5%, 50%, and 95% confidence fragility curves. From this plot, the HCLPF capacity (defined as peak ground acceleration corresponding to 5% probability of failure at 95% confidence) for small LOCA core damage is obtained as 0.21g.

Inspection of the small LOCA core damage Boolean expression [Eq.5.5-1] given above indicates that the dominant components are the singletons. The singleton component with the lowest HCLPF capacity is the refueling water storage tank (RWST) with a HCLPF capacity of 0.21g. Failure of this tank results in no coolant being available for reactor vessel injection following a LOCA.

The other singleton components have HCLPF capacity equal to 0.30g. The capacity of the transformer is estimated based on the proposed upgraded condition.

The seismic fragility of the circulating water pumphouse is conservatively estimated. However, its failure has a less significant effect on the small LOCA core damage HCLPF capacity because of the smaller β_R and β_U.

The HCLPF capacity for No LOCA core damage accident sequence was estimated using the Boolean expression given in Equation 5.5-2 as 0.32g (Figure 5.5-3). The higher capacity against this sequence compared to the small LOCA sequence is because of the absence of low capacity singletons in the expression. Although DWST, i.e., component 8, with its HCLPF capacity of 0.17g appears in this sequence, its failure has to occur simultaneously with one of the higher capacity components i.e., transformer, and circulating water pumphouse.

Core Damage HCLPF Capacity

As explained above, the core damage HCLPF capacity calculation requires a knowledge of the split fraction between the two accident sequences (i.e., small LOCA and Transient). For different assumed split fraction values, the core damage HCLPF capacities were obtained as shown in Table 5.5-3.

The conclusion regarding the dominance of RWST failure in the HCLPF capacity estimation (displayed in the small LOCA accident sequence) is a function of the split fraction assumed. If the plant HCLPF capacity needs to be increased, it is not necessary to concentrate only on RWST. A walkdown and review of small impulse lines within the containment may be performed to estimate their fragilities in order to assign a realistic split fraction. By this procedure, the plant HCLPF capacity may be concluded to be higher without the necessity of any upgrading of the components.

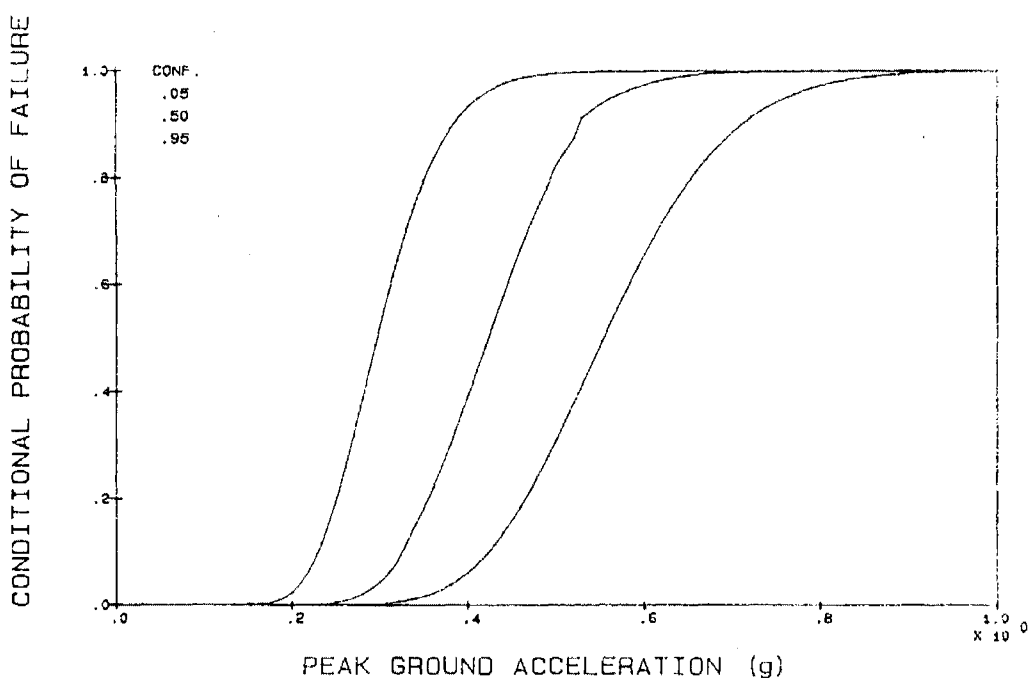


Figure 5.5-2 Fragility Curves for Small LOCA Core Damage.

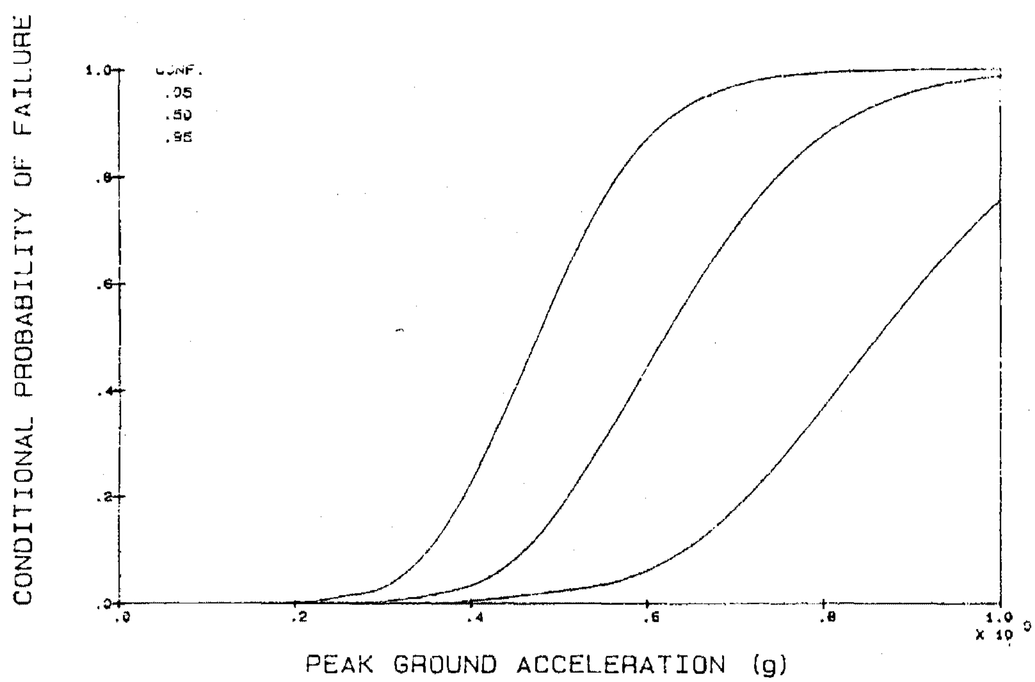


Figure 5.5-3 Seismic Fragility Curves for No LOCA Core Damage.

Table 5.5-3 Summary of Plant Level HCLPF Capacities

Case	Description		HCLPF Capacity (g)
1	Small LOCA		0.21
	- Independent Seismic Failures		
2	Small LOCA		0.21
	- Dependent Seismic Failures		
3	No LOCA		0.30
	- Independent Seismic Failures with Nonseismic Failures		
4	No LOCA		0.30
	- Independent Seismic Failures without Nonseismic Failures		
5	Core Damage		
	- split fraction	p = 0.01	0.30
		p = 0.10	0.28
		p = 0.50	0.23
6	Small LOCA		
	- RWST Reduced Fluid Level		0.26

Effect of Nonseismic Failures

The overall plant HCLPF capacity was calculated using the Boolean expressions for core damage accident sequences which contained both seismic and nonseismic failures. In the following we describe the reasons for including the nonseismic failures in this calculation.

There are two kinds of unavailability. The first kind is that the component fails to start on demand due to random failure, common cause failure or operator error. The other is that the component is unavailable due to maintenance or failure to run. The first is expressed as failure rate per demand; the second is in terms of unavailability per unit time (e.g., year).

For the seismic event, the component fragilities may be considered as failure rate per demand (varies with earthquake level). Hence component fragilities and nonseismic failure on demand rates may be added probabilistically. For the nonseismic unvailabilities, the technique is to convert them into conditional probabilities by multiplying the failure rate by a time interval during which the component is needed to perform.

Example is that the diesels are expected to start on seismic induced loss of offsite power and continue to operate for some hours.

Probability of failure of diesel system = $P_{f_s} + \lambda . t + P_f(a)$

where P_{f_s} = probability of failure to start

λ = rate of failure to run

t = required time for the diesels to be running

$P_f(a)$ = seismic fragility of diesels

The probabilities for nonseismic failures given in Table 5.5-2 include the probability of failure to start, operator error and the probability of failure to run. The uncertainties in the unavailability (per demand) is expressed in terms of the error factor which is equal to the ratio of the 95% confidence value to the median unavailability. The unavailability is modeled by a lognormal probability distribution.

Need for Consideration of Nonseismic Failures in Seismic Margin Studies

In the seismic margin studies, we are interested in estimating the largest earthquake that the plant can withstand with a high degree of confidence. A particular component may be able to withstand this earthquake. Assume that the plant will not suffer core damage if any one of the two components survives this earthquake. Conservatively, we may treat the component failures as statistically dependent. Hence, the plant level HCLPF capacity may be considered to be equal to the higher of the two component capacities. However, the plant HCLPF capacity will be different if one of the components is affected by high nonseismic failure rate. One component may fail in an earthquake, yet the other component may not be available for the plant to survive the earthquake. Therefore, it is important to consider nonseismic failures in estimating seismic margins. Yet, any nonseismic failure that does not occur simultaneously with a seismic failure of a component is not of particular significance in a seismic margin study.

The effect of including nonseismic failures is observed in the No LOCA core damage sequence in that the HCLPF capacity for the sequence changes from 0.33g (when the nonseismic failures are not included) to 0.32g (when the nonseismic failures are included). Since the components with HCLPF capacities >0.3g have been screened out, these plant HCLPF capacities have been represented in Table 5.5-3 as >0.3g. In the small LOCA sequence, there are no nonseismic failures.

Correlation between Seismic Failures

The above calculations were performed assuming perfect independence between seismic failures of different components; i.e., the seismic capacities are assumed to be statistically independent both in randomness and uncertainty. This is a realistic assumption because the components involved in the core damage Boolean expression are yard tanks (RWST, and DWST), transformer, and circulating water pumphouse. These are dissimilar items of structures and equipment, their locations in the plant and within the structures are different, and their dynamic characteristics are also different. Hence correlation in the seismic responses and seismic capacities of these components is judged to be minimal.

It is realistic to expect some correlation in the component failures. Assumption of perfect dependence in both uncertainty and randomness is an extreme case. Assumption of perfect dependence in the uncertainties of different component fragilities means that the median ground acceleration capacities of all components are known if the median ground acceleration capacity of one component is known. Since the uncertainty arises from insufficient understanding of structural material properties, approximate modeling of the structure, inaccuracies in the representation of mass and stiffness, and the use of engineering judgment in lieu of plant specific data, it is expected that all components will be affected to some degree by these uncertainties. Therefore, some probabilistic dependence between component median capacities may be expected. Perfect dependence in uncertainty is however an extreme assumption.

Dependence in the randomness arises from a common earthquake generating the responses in different components and common structural/material properties. Assumption of dependence in the randomness means that if the fragility (conditional probability of failure) of a component for a given peak ground acceleration is known, the probability of failure of the other components would be somewhat modified by that knowledge if it were known.

The plant level fragility, and therefore, the HCLPF capacity depends on the degree of dependence in randomness and uncertainty between the component failures. However, the degree of dependence is difficult to estimate. One approach is to bound the core damage HCLPF capacities by assuming perfect dependence as opposed to the case of perfect independence. This calculation was performed for the small LOCA core damage Boolean expression given above. The small LOCA core damage HCLPF capacity is estimated to be 0.21g i.e., governed by the capacity of RWST. The plant level HCLPF is determined to be approximately 0.21g. When the Boolean expression is dominated by singletons, the assumption of perfect independence is more severe than the assumption of perfect dependence between failures if the fragilities are approximately equal; if there is a single component with a very low capacity compared to the rest of the components in the Boolean expression consisting of singletons, both the assumptions give about the same plant level HCLPF capacity.

The question of dependence between failures is important when there are similar components experiencing common seismic excitation. The case in point is the yard tanks (i.e., RWST, DWST, and PWST). By reviewing the cut sets, it was found that a tripleton cut set could lead to core damage. It is

DWST * PWST * RWST .

The HCLPF capacity for this cutset would be higher than the highest of the three tank HCLPF capacities (i.e., 0.27g for PWST).

5.5.3 Deterministic Method

This approach is based on the assumption that the HCLPF capacities of components estimated in Section 5.4 are true lower bound values. The HCLPF capacity of the plant is obtained directly by studying the Boolean expressions for small LOCA and no LOCA:

Small LOCA

$$= 4 + 7 + 20$$

No LOCA

$$= (4 + 20) * (8 + 15 + 17 + 22)$$

$$+ 8 * (14 + 16) + 7 * 15 \ .$$

For calculating the plant level HCLPF capacity in this method, the nonseismic failures are ignored. Also, the cutset that includes a low probability nonseismic failure is also omitted from this estimation. Therefore, the simplified Boolean expression is:

No LOCA

$$= (4 + 20) * 8$$

In this method the HCLPF capacity of a "doubleton" cut set is calculated as the higher of the two component HCLPF capacities. The HCLPF capacity of a "tripleton" cut set is calculated as the highest of the three component HCLPF capacities. The HCLPF capacity of a union of singleton cut sets is estimated to be the lowest of all the component HCLPF capacities. Using this procedure, the HCLPF capacities of the core damage accident sequences are:

HCLPF capacity against small LOCA core damage

= min [0.30, 0.21, 0.30]

= 0.21g

HCLPF capacity against No LOCA core damage

= max [min (0.30, 0.30), 0.17]

= max [0.30, 0.17]

= 0.30g

Note that the failure of DWST with its HCLPF capacity of 0.17g is not governing the plant level HCLPF capacity because DWST has to fail simultaneously with one of two nonseismic failures to lead to core damage. Since these failures have median probabilities of 6 x 10^{-3} and 8 x 10^{-3}, per demand respectively, it is appropriate to ignore the DWST failure in the plant level HCLPF capacity calculation.

5.5.4 Sensitivity Studies

Two sensitivity studies were performed to assess the effect of certain assumptions on the plant HCLPF capacity. These sensitivity studies addressed the following:

1. Effect of shearwall stiffness reduction on structure response and equipment seismic input.

2 Reduction of RWST fluid level.

On-going scale model testing being conducted by the NRC has indicated potential reductions in the stiffness of concrete shear walls of up to a factor of four, from elastically calculated values due to cracking. This would imply a reduction in the elastic structure frequencies of up to 50%.

For the case with an assumed small LOCA, the RWST is the dominant contributor to the plant HCLPF capacity. RWST constructed of stainless steel and is located in the yard on its own foundation. Its HCLPF capacity is not affected by the potential structural frequency shift. For the case No LOCA, the 4160/480 V transformer is the dominant contributor to the plant HCLPF capacity. The transformer has a 12-13 Hz fundamental frequency which corresponds to a spectral acceleration on the downward slope of the floor response spectra. A reduction of the building frequencies would result in a reduction of the seismic input to the transformer with a correspodig increase in its HCLPF capacity. As a further check on the effect of the building frequency shift, a similar evaluation was made on each of the components within the final Boolean expression (Table 5.5-1). The potential building frequency reduction did not lower the HCLPF capacity for any of these components. Thus, for the purpose of the Maine Yankee seismic margin study, the shear wall stiffness reduction and resulting building frequency shift does not affect the plant seismic margin.

Because the seismic capacity of the RWST controls the plant HCLPF capacity for the small LOCA case, the effect of reducing the fluid level within the tank was investigated. A reduction of the fluid level to a total height of 33 ft, which is approximately equal to 10% reduction from the current level of 37 ft, was assumed. This change leads to a reduction in the effective fluid weight, mass centroid height, and overall tank seismic loads. The RWST HCLPF capacity is increased to 0.28g with a corresponding increase of the small LOCA HCLPF capacity to 0.26g.

CHAPTER 6

COMMENTS AND RECOMMENDATIONS ON SEISMIC MARGIN REVIEW METHODOLOGY

6.1 Selection of Review Earthquake Level

In the course of this trial application of the seismic margin review methodology, it was found that sufficient guidance was not included in NUREG/CR-4334 and NUREG/CR- 4482 for the selection of the review earthquake level. NUREG/CR-4334 states that:

> "The Panel has focused its efforts on earthquakes that could occur in the eastern part of the U.S., specifically east of the Rocky Mountains. Because of limited data on large magnitude events, the assessment of component capacities is limited to earthquakes of less than a Richter magnitude of about 6.5, which are characterized by three to five strong motion cycles with a total duration of 10 to 15 seconds. As the Richter magnitude increases above 6.5, the Panel recommendations may be slightly non-conservative. The frequency content of the earthquake round motion is assume to be represented by median broadband response spectra, and the structures are assumed to have fundamental frequencies above 1 Hz. Note that for high frequency, high acceleration, low magnitude earthquakes, the Panel's recommendations are overly conservative."

The report goes on to state that:

> "The margin review must begin with a target earthquake, in order to provide focus. It is assumed for the purposes of the review activity that some external source (perhaps the NRC staff, or the utility) has designated the earthquake level (in terms of peak ground acceleration, pga) which is the level at which the review is aimed".

The Panel in the NUREG/CR-4482 has stated that:

> "The choice of the review level is a critical one since it is used as a basis for screening out components. The review level should be specified in terms of pga and enough spectral information to assure the applicability of the material of Chapter 5 in NUREG/CR-4334 which is partially given in Table 2-1 of this report. If the review earthquake spectral content is not consistent with the assumptions made in NUREG/CR-4434, then this difference needs to be taken into account. Table 2-1 was

> constructed to cover most spectra generated by magnitude 6.5 earthquakes or less. In addition, spectral information will be needed to calculate HCLPF values."

For the seismic margin review of Maine Yankee, the NRC specified that the review earthquake level would be 0.3g with a median Newmark-Hall ground response spectrum as defined in NUREG/CR-0098 for rock sites. Since this spectrum is applicable generically for all rock sites, it was concluded by the NRC that the selected spectrum is a 84% confidence spectrum for Maine Yankee (See Chapter 2). The requirements on the review earthquake level are also somewhat implicit in the Panel's recommendations. Yet, for future applications, it is suggested based on this trial plant review that more explicit guidance be provided.

6.2 Use of Screening Guidelines

6.2.1 Extent of Review Needed

As discussed before, this study has two primary objectives: to confirm the appropriateness and adequacy of guidelines given in the Expert Panel reports and to estimate the seismic margin for Maine Yankee. Because of this dual objective, the study team had to expend many engineering hours in reviewing the components that were supposedly to have been screened out in the initial screening, i.e., those components identified as having seismic capacities generically higher than the review earthquake level and denoted by letter "C". For example, the review by the study team included components such as valves, diesel generators, pumps, HVAC ducting, cable trays and cabling although the Panel screening guidelines state that these components have seismic capacities larger than 0.30g pga. It may be pointed out that the Panel screening guidelines are invariably conservative in that the specific components in a plant may actually have capacities larger than indicated by the Panel. Also, the collective experience and judgment of the Expert Panel in developing the guidelines cannot be expected to be matched by any single team performing the seismic margin reviews in future. Hence, in keeping with the primary objective of minimizing the review effort needed in a seismic margin review, it is recommended that the components identified as having capacities larger than the review earthquake level be screened out with no further review in future seismic margin studies.

6.2.2 Additional Screening Guidelines

The Expert Panel reports have not explicitly discussed the seismic capacities of steel structures. In the trial plant review on Maine Yankee, three steel structures were analyzed to assess their HCLPF capacities. These structures do have capacities in excess of 0.30g. At the same time, they could not be screened out based on a walkdown review because of unusual connection details and structural arrangement.

Therefore, some guidance is necessary in the margin review methodology for steel structures.

6.2.3 Difficulties of Reviewing Certain Components

The majority of the components identified for review in the Maine Yankee seismic margins study were reviewed during the plant walkdowns. Component design reports and seismic qualification analyses were also reviewed on a majority of the Maine Yankee equipment as a part of the fragility assessment. Difficulties in reviewing certain Maine Yankee components fell into one of two categories:

- o Components whose qualification reports were available only from the NSSS supplier
- o Components located inside the containment structure

6.2.3.1 Reactor Internals and CEDM

The reactor internals and the control element drive mechanism (CEDM) are two components requiring fragility assessments for the Maine Yankee margin study. The utility (MYAC) did not have access to the seismic qualification documentation for two reasons:

1. This information is considered proprietary by the NSSS vendor
2. Due to the vintage of the Maine Yankee Plant, seismic qualification was not required for the CEDM and for portions of the reactor internals

The NSSS supplier, Combustion Engineering (CE), is the only credible source of capacity/response information on the internals and the CEDM. Discussions between CE, MYAC and EQE resulted in a contract being set up through LLNL for CE to collect the qualification data from their files and to meet with EQE and Maine Yankee representatives. CE was very cooperative during this meeting and provided the following:

- o Summaries of the qualification report results
- o Similarity assessments for components like the CEDM which had no previous seismic qualification

This interfacing with the NSSS supplier will generally be required on future seismic margin studies, just as it has been required on almost all of the previous seismic PRA's. In addition, for plants where the utility does not take such an active role in the design/margin review, meetings with the architect-engineer to obtain plant/component information may also be required.

6.2.3.2 Equipment Within Containment

Due to radioactivity concerns, critical components inside the Maine Yankee containment are accessible for review only at the time of a plant outage. The margin study schedule precluded the possibility of one of the plant walkdowns coinciding with the March 1987 plant refueling outage. This fact necessitated the use of previous photographs and design drawings for fragility derivation purposes.

Fortunately, Maine Yankee personnel have maintained a relatively complete file of component photographs based on previous seismic integrity studies. The experiences at Maine Yankee point out the need in future margin studies to plan for the review of components within containment. Difficulties in reviewing plant components will be more severe for plants which do not maintain organized documentation on the components within containment.

Impulse line failures were assumed to be the source of a small LOCA at Maine Yankee. This conservative assumption was required due to the tremendous number of hours which would be required to walk down each of these impulse lines and assess potential system interaction problems. These lines originate from the primary pressure boundary (i.e., RPV, steam generator, pressurizer, primary coolant loop piping, etc.) and are field routed to instrument racks inside containment. The amount of work required to demonstrate the seismic margin in each of these lines plus the fact that a walkdown of these lines would have to take place during a plant outage necessitated the assumption of a small LOCA as an initiating event.

6.3 Availability of Qualification Data

In-structure response spectra for the median NUREG/CR-0098 ground response spectrum were generated for use in the trial plant review by Maine Yankee during the course of this study. The structure building models were developed recently for other Maine Yankee programs. The availability of in-structure response spectra generated by current techniques can reduce the amount of effort necessary to quantify component fragilities since correction or regeneration of responses was not necessary. Adequate floor spectra may not always be available for older plants. In such cases, it may be necessary to develop new dynamic models and perform dynamic analysis to define the seismic input to equipment. This will increase the amount of time and funding necessary to perform seismic margin studies.

Most of the Maine Yankee Group A structures were screened out, so detailed seismic load distributions to the structural members were not necessary for the trial plant review. For the screened in structures, seismic load distributions were either available from recent dynamic analyses, or could be estimated with relatively little effort due to simplicity in structure load paths. Consequently, determination of seismic load distributions was a minor consideration. However, this would not have been the case had it been necessary to rely upon load distributions from the original design calculations. The original structure design calculations for Maine Yankee comprised a large volume of information. Determination of the seismic loads and verifying their accuracy within this body of material would have been time-consuming and tedious. Typically, original structure design calculations have not been used in past PRAs for this reason. When necessary, approximate load distributions have been independently developed. Generation of structure load distributions with the level of accuracy appropriate for a seismic margin review could require additional effort.

Detailed information on the Maine Yankee block walls was available. These data were very useful in conducting the trial plant review since the data provided locations of all block walls in the plant and identified any safety related components that could be affected by their failure. In particular, this latter set of information was valuable since it permitted quick identification of several Group A block walls. It is not known if comparable information exists for most older plants.

Seismic qualification information for Maine Yankee equipment was available for selected critical components. Maine Yankee engineers and their consultants have conducted specific qualification tests and analysis as the need to do such an analysis has been identified. There is very little seismic qualification analysis available from the original plant design documentation. This fact stems from the fairly relaxed requirements in the seismic qualification area at the time of the Maine Yankee plant construction. Seismic qualification in many cases consisted of simply a letter from the vendor that his equipment component had adequate seismic capacity without any supporting documentation.

Examples of equipment components for which seismic qualification information was available included the diesel generator day tank, battery racks, RHR heat exchanger, emergency service water pump, component cooling heat exchanger, demineralized water storage tank and the diesel generator remote control panel. Seismic anchorage calculations were also available for all of the electrical equipment which had their anchorage/supports upgraded from their original design (e.g., inverters, motor control centers, control room cabinets, transformers, etc.). Combustion Engineering provided seismic analysis data for some of the reactor internal components, but the remaining reactor internals structures and the control element drive mechanism did not have qualification data because there was no detailed seismic analysis requirement for Maine Yankee vintage plants. Equipment for which seismic qualification data was not available included the majority of the valves, fans, dampers, heat exchangers, small tanks, pumps, and air conditioners at Maine Yankee.

6.4 Walkdown Procedures

Based upon the trial plant review, suggestions on improvements to the review methodology in the area of walkdown procedures would include the following:

- o Third walkdown
- o Access to personnel with specialized expertise
- o Data sheets
- o Level of walkdown for bulk items

During the course of the trial plant review, certain components with relatively low HCLPF capacities were identified. The utility elected to develop conceptual modifications to be installed in the next outage. Plant HCLPF capacities reported for this study reflect the increased component capacities based upon the conceptual modifications. A third walkdown will be necessary to confirm that these modifications are installed, and that they provide seismic capacity at least comparable to the conceptual retrofits upon which the calculated component capacities were based.

During the walkdowns, plant and utility personnel having expertise or specialized knowledge in particular fields were interviewed. Specific areas of expertise included:

- o Fire water systems
- o Electrical
- o HVAC
- o Instrumentation and control
- o Control room personnel
- o Block walls

The insight gained from discussions with these individuals proved to be very useful in assessing the overall state of the plant and resolving any particular seismic issues. Scheduled meetings with plant experts in each of these areas allow the fragility and systems analysts to gain in-depth understanding of plant specific parameters such as design criteria, construction practices, equipment locations and functions, and operator actions in emergency events.

Walkdown data sheets (Section 4.3.3), while not noted in the review guidelines, were developed prior to the first walkdown. These data sheets served as a checklist of items to review during the walkdown and facilitated the gathering of information necessary for HCLPF evaluation. While no set format need be established for future margin reviews, comment on the use of data sheets in the review guidelines should be considered.

Bulk items such as cable trays, piping, and ducting extend throughout the plant, and local configurations and support details can vary. Detailed walkdown of these bulk commodities would be time-consuming. The extent of walkdown required for review level earthquakes less than 0.3g is typically specified in general terms. As an example, the Expert Panel recommends that example cable trays be inspected to verify that they are adequately anchored and braced and that specific vulnerabilities do not exist. In the trial plant review, a general survey of the cable tray systems was performed and supplemented by detailed inspection of a typical system. More guidance in the seismic margins methodology on the level of walkdown for bulk items would be useful.

6.5 Guidance on CDFM Capacity Calculation Procedures

The second report by the Expert Panel [Prassinos et al., 1985] recommends two candidate methods for calculating the HCLPF capacities for components: the conservative deterministic failure margin (CDFM) method and the fragility analysis (FA) method. The fragility analysis method was used in this study as was done in over 20 seismic PRAs. This method, although very familiar to the present analysis team, calls for subjective judgments to estimate parameters such as A_m, β_R and β_U for each variable needed in the seismic capacity estimation. The CDFM method, on the other hand, prescribes the parameter values and procedures to be used in calculating the HCLPF capacities. It is conceptually very attractive because it aims to avoid subjective judgments on the part of the analysts: first, the analysts may not be equipped to make such judgments and second, there may be inconsistencies between the subjective fragility assignments by different analysts with varied background and expertise. For the sake of consistency and repeatability, we need a prescriptive method to estimate the seismic capacities of structures and equipment. Ideally, the CDFM method would fit the bill when it is fully developed. The guidance on the use of CDFM method in NUREG/CR-4482 is not sufficient for this purpose.

In the course of this study, five example components (i.e., refueling water storage tank, steel structure, diesel day tank, inverter, and block wall) were analyzed using the two candidate methods (i.e., CDFM and FA). Our experience was that several judgmental decisions had to be made in arriving at the parameters of the CDFM method. In each case we were not sure whether we met the intent of the method, i.e., conservative estimation of the capacity, yet more liberal than the SRP requirements: In some cases we may have been overly conservative as was pointed out by the Peer Review Group. The difficulties arise because of two factors:

- The CDFM method has not been fully developed for all structures and equipment items.

- The parameters of the CDFM method such as damping, material strength, static capacity equations, system ductility, and methods for floor spectra generation are not explicitly specified; even where they are specified they may be overly conservative. Also, the appropriate conservatism in the selection of the CDFM parameters needs to be determined using calibration methods.

Development of deterministic procedures such as the CDFM method may be accomplished using probabilistic models in such a way that over a large number of components both the CDFM and FA methods yield identical HCLPF capacities. Once such a calibration is achieved, the CDFM method may be confidently used in future seismic margin reviews. The calibration should start with a fragility evaluation of a representative set of components using the models and parameters agreeable to the group (i.e., Seismic Margins Expert Panel). The parameters of the CDFM method are to be selected such that over this representative set of components the HCLPF capacities derived using the CDFM method and the fragility analysis method are approximately equal.

We recommend that such a calibration study be performed.

6.6 Staffing Requirements and Schedule

6.6.1 Staffing Requirements

NUREG/CR-4482 gives estimated staffing requirements for a seismic margin review; for a plant founded on rock with a review earthquake level of 0.30g pga, the estimate to perform the fragility analysis is 19 engineering-months. The actual engineering-months expended in the present study is about 25% more than the Panel's estimate. Several factors need to be considered in this regard:

- The dual objective of the study, i.e., seismic margin review of Maine Yankee and a trial application of seismic margin review methodology, required additional review efforts and interface meetings not to be expected in future studies.

- o Maine Yankee is an older plant designed and constructed before the development of quality assurance programs and seismic qualification methods presently existing in the nuclear industry. Hence, the qualification reports on certain equipment items were not available. Because of the reevaluation efforts, additional information and new structural models have become available. However, the extent of information available is not comparable to that normally available for a modern nuclear power plant (e.g.,NTOL plant). The staffing requirements in the NUREG/CR-4482 should therefore be revised to reflect the availability of design and qualification data in the particular plant.

- o The utility performed the seismic response analysis for structures in the plant and developed the floor response spectra for the selected review earthquake. The engineering hours spent on this effort should be included in the estimates for future margin studies.

- o The plant design information was provided by the utility with some assistance from the NSSS vendor. Also, the utility personnel assigned for the interface were extremely familiar with the plant arrangement, systems and design. Typically, the interface will extend to the plant architect-engineer and constructor. The level of effort to collect plant design information and perform the plant walkdown may be much larger for other plants as more organizations become involved. Our seismic PRA experience indicates that the Maine Yankee interface experience is very fortunate and not typical.

6.6.2 Schedule

This study has been conducted over an eight-month period; the plant walkdowns have also been conducted as per the schedule outlined in the Panel's report [Prassinos et al., 1985]. The schedule appears to be reasonable.

6.7 Applicability of Methodology for Other Plants

The methodology has been developed using the insights obtained from seismic PRAs on six PWRs supplied by Westinghouse and Babcock & Wilcox and a BWR Mark II. This study on Maine Yankee has added to this collective experience in that a PWR supplied by Combustion Engineering has been analyzed. From a fragility standpoint, the CE reactor internals and the control rod drive mechanism have augmented the PRA fragility information base. The screening guidelines may be revised to reflect this additional information.

The trial plant review has been for a plant on a rock site and for a review earthquake level of 0.30g. The methodology, the review guidelines, and the staffing requirements have not been verified for other conditions, i.e., soil conditions and higher review earthquake level. Also, the methodology has not beendeveloped for screening systems in a BWR. In a BWR, the effect of high-frequency hydrodynamic loads on the equipment needs to be considered. The systems and components to be reviewed in a BWR would be different and the level of effort and staffing requirements would vary. The applicability of the Panel's recommendations to these reactor/site/review levels needs to be demonstrated.

CHAPTER 7

RESULTS AND CONCLUSIONS

The primary objective of this study has been to apply the seismic margin review methodology developed by the NRC Expert Panel on a trial basis in order to confirm the applicability of the methodology for future seismic reviews and to provide input to the Panel for modifications in the methodology and review procedures. The other objective of the study has been to determine the seismic margin of the Maine Yankee Atomic Power Station. This report focuses on the fragility aspects of the seismic margin review. The report has described the plant review, screening of the components based on their generic high seismic capacities, plant walkdowns and calculations of the HCLPF capacities of the components, systems, accident sequences, and the plant. During this review, potential seismic vulnerabilities have been identified and for some of them the utility has proposed/incorporated certain modifications.

7.1 Screened Out Systems and Components

Based on the screening guidelines of the Expert Panel, systems (both front-line and supporting sytems) not supporting Group A functions were screened out by Energy International Inc. These systems are discussed in Volume 2 of this Report. Among those components supporting the Group A function systems, some of them were screened out based on their seismic capacities as generically larger than 0.3 g pga, the selected review level earthquake. Table 7.1-1 lists the Group A structures, the screening procedures used, and the HCLPF capacities. Table 7.1-2 provides a summary of Group A block walls; included are the wall ID number, the Group A components that may be affected by the wall failure and the bases for screening and HCLPF capacity. Table 7.1-3 lists the equipment in Maine Yankee by generic categories, the systems they support, their location, the screening procedures used (i.e., screening tables in NUREG/CR-4334, walkdown review, calculation review) and the HCLPF capacity.

7.2 HCLPF Capacities of Screened-in Components

The definition of screened-in components depends on the stage of the reiew. Screening was done at four stages:

- o Preliminary screening based on the plant design information and using the Panel guidelines
- o Preliminary screening confirmed by first walkdown and additional items screened out based on walkdown observation
- o Screening based on conservation capacity calculations using simplified methods
- o Screening based on final capacity calculations that show the HCLPF capacities to exceed 0.30g

Table 7.1-1 Summary of Group A Structures

	Screening				
Structure	NUREG 4334	Walkdown Review	Fragility Evaluation	HCLPF Capacity	Comments
Containment structure	C			>0.3g	Reinforced concrete structure.
Containment internal structure	C			>0.3g	Reinforced concrete structure.
Primary auxiliary building/fuel building	C			>0.3g	Reinforced concrete structure.
Circulating water pumphouse		X	X	>0.30g	Collapse capacity judged to be >0.3g based upon past earthquake performance, test data, and fragility evaluation.
Turbine/service building, concrete portions	C				Reinforced concrete structure.
Turbine/service building, steel framed floor at EL 39'-0"			X	0.38g	HCLPF capacity based upon detailed fragility evaluation.
Turbine/service building, other steel framed portions			X	>0.3g	HCLPF capacity >0.3g based upon conservative fragility evaluation.
Containment spray pumphouse	C			>0.3g	Reinforced concrete structure.
Ventilation equipment room	C			>0.3g	Reinforced concrete structure.
Main steam valve house, exterior concrete structure	C			>0.3g	Reinforced concrete structure.
Main steam valve house, interior steel structure			X	>0.3g	HCLPF capacity >0.3g based upon conservative fragility evaluation.
M.C.C. room	C			>0.3g	Reinforced concrete structure.
Aux feed pumphouse, purge air exhaust area	C			>0.3g	Reinforced concrete structure.
Fuel oil pumphouse	C			>0.3g	Reinforced concrete structure.

Table 7.1-2 Summary of Group A Block Walls

Wall ID No.	Group A Components	Screening	HCLPF Capacity	Comments
C -2-1, C 20-2, 3	Assorted safety class piping and equipment	Out		These walls are loose blocks shields. Adjacent components are protected by shielding consisting of steel framing.
C -2-2, 3	Containment liner	Out		Breach of the containment liner due to impact by the block wall is considered unlikely
C 0.5-1, C 20-1	Pressurizer instrumentation and tubing	In	>0.3g	HCLPF capacity based upon simplified fragility evaluation.
C 46-1, 2 C 61-1 to 3	Steam generator snubbers and tubing	In	>0.3g	Seismic retrofits are installed. HCLPF capacity based upon simplified fragility evaluation.
SB 21-7	PCC and SCC piping, PCC surge line	Out		The PCC and SCC lines are about 24" in diameter. The PCC surge line is about 6" in diameter. These piping systems are of welded construction. Failure due to impact by the block wall is considered highly unlikely.
SB 21-17	Main control board (aux feed panels)	In	>0.3g	Seismic retrofits are installed. HCLPF capacity based upon simplified fragility evaluation.
SB 21-18	Main control board (aux feed panels), aux logic panels	In	>0.3g	Seismic retrofits are installed. HCLPF capacity based upon simplified fragility evaluation.
SB 35-1	Battery groups 3 and 4, cable trays	In	>0.3g	HCLPF capacity judged greater than value for wall SB 35-4.

Table 7.1-2 Summary of Group A Block Walls (Continued)

Wall ID No.	Group A Components	Screening	HCLPF Capacity	Comments
SB 35-2	Battery groups 3 and 4, cable trays	In	>0.3g	HCLPF capacity based on detailed fragility evaluation.
SB 35-3	Battery groups 3 and 4, cable trays	In	>0.3g	HCLPF capacity based on detailed fragility evaluation.
SB 35-4	Battery groups 3 and 4	In	>0.3g	HCLPF capacity based on detailed fragility evaluation.
SB 35-7	PCC surge line, PCC temperature controller	In	>0.3g	HCLPF capacity judged greater than value for wall SB 35-3.
SB 45-1	125V DC distribution cabinets 1 to 4, Battery group #2, Inverters #1 and #2, Battery chargers #1 and #2, Bus 8,	In	>0.3g	Seismic retrofits are installed. HCLPF capacity based on simplified fragility evaluation.
SB 45-2	Battery group #1, MCC 8A, 480V emergency switchgear	In	>0.3g	Seismic retrofits are installed. HCLPF capacity based on simplified fragility evaluation.
SB 45-3	Battery group #1	In	>0.3g	Seismic retrofits are installed. HCLPF capacity based on simplified fragility evaluation.
SB 45-6	PCC surge line	Out		The PCC surge line is shielded from the wall by a seismic retrofit consisting of two steel angles running parallel with the pipe. Failure of both the angles and the surge line due to impact by the block wall is considered highly unlikely.
SB 61-2	PCC surge tank TK-5	Out		The PCC surge tank is shielded from the block wall by a seismic retrofit that consists of welded wire fabric spanning between steel framing. The shielding is judged to be sufficient to restrain the blocks from impacting the tank.

Table 7.1-2 Summary of Group A Block Walls (Continued)

Wall ID No.	Group A Components	Screening	HCLPF Capacity	Comments
SB 77-2	PCC surge tank TK-5, SCC surge tank TK-59	Out		The surge tanks are shielded from the block wall by a seismic retrofit that consists of welded wire fabric spanning between steel framing. The shielding is judged to be sufficient to restrain the blocks from impacting the tanks.
TB 21-2	PCC heat exchangers E-4A and 4B	Out		The heavy construction of the heat exchangers is judged to be capable of withstanding impact by the block wall. No other vulnerable components are exposed.
VE 21-1	Fans 44A and 44B, ducting and filter	In	>0.3g	Seismic retrofits are to be installed in the next outage. Detailed fragility evaluation of conceptual modification sketches indicates that the HCLPF capacity is >0.3g.
VE 21-4	SCC lines	In	>0.3g	HCLPF capacity based upon simplified fragility evaluation.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities

Equipment Item	System	Building and Elevation	Equipment Qualification Screening			HCLPF Capacity	Comments
			NUREG 4334	Walkdown Review	Calculation Review		
TANKS							
1. Refueling Cavity Water Storage Tank TK-4	HPSI	Yd. + 20'			X	FA 0.21g (1) CDFM 0.21g (2) (0.24g) (3)	
2. Primary Component Cooling Surge Tank TK-5	PCC	SB + 61'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
3. Primary Water Storage Tank Tk-16	AFW	Yd. + 20'			X	0.27g (1) (0.25g) (3)	
4. Demineralized Water Storage Tank Tk-21	AFW	Yd. + 20'			X	0.17g (1) (0.25g) (3)	
5. Air Start Receiver Tank Tk-25	AFW	VA + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
6. Auxiliary Fuel Oil Supply Tank (buried) Tk-28A	FO	APR + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Buried Tanks.
7. Auxiliary Fuel Oil Supply Tank (buried) Tk-28B	FO	APR + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Buried Tanks.

(1) FA - HCLPF capacity calculated per the Fragility Analysis approach.
(2) CDFM - HCLPF capacity calculated per the Conservative Deterministic Failure Margin approach.
(3) HCLPF capacity for unanchored vertical fluid storage tanks per Manos' approach.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
TANKS (Continued)							
8. Spray Chemical Addition Tank Tk-54	HPSI	Yd. + 20'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
9. Secondary Component Cooling Surge Tank Tk-59	SCC	SB + 70'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
10. Emergency Diesel Day Tank Tk-62A	FO	AB + 21'			X	FA 0.43g CDFM 0.54g	Reflects the Modified Condition.
11. Emergency Diesel Day Tank Tk-62B	FO	AB + 21'			X	FA 0.43g CDFM 0.54g	Reflects the Modified Condition.
12. DG-1A Compressed Air Tank Tk-76A1	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
13. DG-1A Compressed Air Tank Tk-76A2 (Same supt. as A1)	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
14. DG-1A Compressed Air Tank Tk-76A3 (Same supt. as A1)	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
15. Diesel Starting Air Receiver 1A Tk-76A4	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
16. Diesel Starting Air Receiver 1A Tk-76A5 (Same supt. as A4)	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
TANKS (Continued)							
17. Diesel Starting Air Receiver 1A Tk-76A6 (Same supt. as A4)	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
18. Dg-1B Compressed Air Tank Tk-76B1	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
19. Dg-1B Compressed Air Tank Tk-76B2 (Same supt. as B1)	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
20. Dg-1B Compressed Air Tank Tk-76B3 (Same supt. as B1)	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
21. Diesel Starting Air Receiver 1B Tk-76B4	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
22. Diesel Starting Air Receiver 1B Tk-76B5 (Same supt. as B4)	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.
23. Diesel Starting Air Receiver 1B Tk-76B6 (Same supt. as B4)	DG	AB + 21'			X	>0.3g	Conservative Fragility Calculation Determined HCLPF Capacity is >0.3g.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
PUMPS							
1. Primary Component Cooling Pump P-9A	PCC	TB + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Horizontal Pumps.
2. Primary Component Cooling Pump P-9B	PCC	TB + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Horizontal Pumps.
3. Secondary Component Cooling Pump P-10A	SCC	TB + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Horizontal Pumps.
4. Secondary Component Cooling Pump P-10B	SCC	TB + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Horizontal Pumps.
5. Charging Pump P-14A	HPSI	PAB + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Vertical Pump W/Shaft Length <20'.
6. Charging Pump P-14B	HPSI	PAB + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Vertical Pump W/Shaft Length <20'.
7. Charging Pump P-14S	HPSI	PAB + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Vertical Pump W/Shaft Length <20'.
8. Emergency Feed Pump P-25A	AFW	AF + 21'	C			>0.3g	Horizontal Pump, HCLPF Cap. >0.3g.
Oil Cooler (Skid mounted) E-86A				X		>0.3g	Oil Cooler HCLPF Cap. Judged >0.3g Assessed on Walkdown Review.
9. Emergency Feed Turbine Pump P-25B	AFW	VA + 21'	C			>0.3g	Horizontal Pump, HCLPF Cap. >0.3g.
Oil Cooler (Skid mounted) E-86B				X		>0.3g	Oil Cooler HCLPF Cap. Judged >0.3g Assessed on Walkdown Review.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
PUMPS (Continued)							
10. Emergency Feed Pump P-25C	HPSI	AF + 21'	C			>0.3g	Horizontal Pump, HCLPF Cap. >0.3g.
Oil Cooler (Skid mounted) E-86C				X		>0.3g	Oil Cooler HCLPF Cap. Judged >0.3g Assessed on Walkdown Review.
11. Service Water Pump P-29A	SW	CW + 7'	C		X	>0.3g	Conservative Fragility Calculation Determined Motor to Pump Bolts HCLPF Capacity is >0.3g.
12. Service Water Pump P-29B	SW	CW + 7'	C		X	>0.3g	Conservative Fragility Calculation Determined Motor to Pump Bolts HCLPF Capacity is >0.3g.
13. Service Water Pump P-29C	SW	CW + 7'	C		X	>0.3g	Conservative Fragility Calculation Determined Motor to Pump Bolts HCLPF Capacity is >0.3g.
14. Service Water Pump P-29D	SW	CW + 7'	C		X	>0.3g	Conservative Fragility Calculation Determined Motor to Pump Bolts HCLPF Capacity is >0.3g.
15. Auxiliary Fuel Oil Transfer Pump P-33A (Vertical Pump over Tk-28A)	FO	Yd. + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Vertical Pump W/Shaft Length <20'.
16. Auxiliary Fuel Oil Transfer Pump P-33B (Vertical Pump over Tk-28A)	FO	Yd. + 21'	C			>0.3g	HCLPF Capacity Judged >0.3g for Vertical Pump W/Shaft Length <20'.
17. Containment Spray Pump P-61A	CS	CS + 14'	C		X	>0.3g	Conservative Fragility Calculation Determined Motor to Pump Bolts HCLPF Capacity is >0.3g.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
PUMPS (Continued)							
18. Containment Spray Pump P-61B	CS	CS + 14'	C		X	>0.3g	Conservative Fragility Calculation Determined Motor to Pump Bolts HCLPF Capacity is >0.3g.
19. Containment Spray Pump P-61S	CS	CS + 14'	C		X	>0.3g	Conservative Fragility Calculation Determined Motor to Pump Bolts HCLPF Capacity is >0.3g.
HEAT EXCHANGERS							
1. Residual Heat Removal Heat Exchanger * E-3A	PCC	CS + 14'			X	>0.3g	Conservative Fragility Calculation Determined the Supports and Anchor Bolts HCLPF Capacity is >0.3g.
2. Residual Heat Removal Heat Exchanger * E-3B	SCC	CS + 14'			X	>0.3g	Conservative Fragility Calculation Determined the Supports and Anchor Bolts HCLPF Capacity is >0.3g.
3. Primary Component Cooling Heat Exchanger E-4A	PCC	TB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Supports and Anchor Bolts HCLPF Capacity is >0.3g.
4. Primary Component Cooling Heat Exchanger E-4B	PCC	TB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Supports and Anchor Bolts HCLPF Capacity is >0.3g.
5. Secondary Component Cooling Heat Exchanger E-5A	SCC	TB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Supports and Anchor Bolts HCLPF Capacity is >0.3g.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Walkdown Review	Calculation Review	HCLPF Capacity	Comments
HEAT EXCHANGERS (Continued)							
6. Secondary Component Cooling Heat Exchanger E-5B	SCC	TB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Supports and Anchor Bolts HCLPF Capacity is >0.3g.
7. Fuel Pool Heat Exchanger * E-25	PCC	FB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Supports and Anchor Bolts HCLPF Capacity is >0.3g.
8. Seal Water Heat Exchanger * * E-34	PCC	PAB + 11'				----	Valves PCC-M-90 & 219 Have a HCLPF Capacity >0.3g and will Isolate this Component.
9. Reactor Containment Air Recirculation * Cooler E-54-1	PCC	RC + 46'			X	>0.3g	Conservative Review Determined the Cooler Supports and Anchorage HCLPF Capacity is >0.3g.
10. Reactor Containment Air Recirculation * Cooler E-54-2	PCC	RC + 46'			X	>0.3g	Conservative Review Determined the Cooler Supports and Anchorage HCLPF Capacity is >0.3g.
11. Reactor Containment Air Recirculation * Cooler E-54-3	PCC	RC + 46'			X	>0.3g	Conservative Review Determined the Cooler Supports and Anchorage HCLPF Capacity is >0.3g.
12. Reactor Containment Air Recirculation * Cooler E-54-4	PCC	RC + 46'			X	>0.3g	Conservative Review Determined the Cooler Supports and Anchorage HCLPF Capacity is >0.3g.
13. Reactor Containment Air Recirculation * Cooler E-54-5	PCC	RC + 46'			X	>0.3g	Conservative Review Determined the Cooler Supports and Anchorage HCLPF Capacity is >0.3g.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
HEAT EXCHANGERS (Continued)							
14. Reactor Containment Air Recirculation * Cooler E-54-6	PCC	RC + 46'			X	>0.3g	Conservative Review Determined the Cooler Supports and Anchorage HCLPF Capacity is >0.3g.
15. Reactor Coolant Regenerative Heat * * Exchanger E-67	HPSI	RC + 21'				----	Valves PCC-M-90 & 219 Have a HCLPF Capacity >0.3g and will Isolate this Component.
16. Safeguards Pumps Seal Leakage Cooler * E-91A	SCC	CS - 02'		X		>0.3g	Cooler HCLPF Capacity Judged >0.3g Assessed on Walkdown Review.
17. Safeguards Pumps Seal Leakage Cooler * E-91B	PCC	CS + 00'		X		>0.3g	Cooler HCLPF Capacity Judged >0.3g Assessed on Walkdown Review.
18. Charging Pump Seal Leakage Cooler * E-92A	SCC	PAB + 11'			X	>0.3g	Conservative Fragility Calculation Determined the Supports and Anchor Bolts HCLPF Capacity is >0.3g.
19. Charging Pump Seal Leakage Cooler * E-92B	PCC	PAB + 11'			X	>0.3g	Conservative Fragility Calculation Determined the Supports and Anchor Bolts HCLPF Capacity is >0.3g.
20. Seal Water Heater * * E-96	HPSI	PAB + 11'				----	Valves PCC-M-90 & 219 Have a HCLPF Capacity >0.3g and will Isolate this Component.
21. Auxiliary Charging Pump P-7 Lube Oil * Cooler (on skid)	CH	PAB + 11'		X		>0.3g	Cooler HCLPF Capacity Judged >0.3g Assessed on Walkdown Review.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
HEAT EXCHANGERS (Continued)							
22. Containment Penetration Cooling Lines * (Line integrity only)	PCC	various		X		>0.3g	Cooler HCLPF Capacity Judged >0.3g Assessed on Walkdown Review.
23. LPSI Pump Coolers (coolers are coils mounted on the pumps)	LPSI *	CS + 21'		X		>0.3g	Cooler HCLPF Capacity Judged >0.3g Assessed on Walkdown Review.
24. Containment Spray Pump Coolers * (coolers are coils mounted on the P-61 Pumps)	CS	CS + 14'		X		>0.3g	Cooler HCLPF Capacity Judged >0.3g Assessed on Walkdown Review.
MISCELLANEOUS COMPONENTS							
1. Diesel Generator DG-1A Diesel Generator Heat Exchanger E-82A * DG 1A Engine Control Panel	DG	AB + 22'	C			>0.3g	HCLPF Capacity is Judged >0.3g for the Diesel Gen. and Peripherals.
2. Diesel Generator DG-1B Diesel Generator Heat Exchanger E-82B * DG 1B Engine Control Panel	DG	AB + 22'	C			>0.3g	HCLPF Capacity is Judged >0.3g for the Diesel Gen. and Peripherals.
3. Seal Water Supply Filter * * FL-34B	HPSI	unknown				----	Valves PCC-M-90 & 219 Have a HCLPF Capacity >0.3g and will Isolate this Component.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening			HCLPF Capacity	Comments
			NUREG 4334	Walkdown Review	Calculation Review		
ELECTRICAL DISTRIBUTION SYSTEMS							
1. 4160V Emergency Bus Bus 5	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
2. 4160V Emergency Bus Bus 6	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
3. 480V Emergency Bus Bus 7	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
4. 480V Emergency Bus Bus 8	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
5. 480V Emergency Motor Control Center MCC-7A	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
6. 480V Emergency Motor Control Center MCC-7B	Elec	RMC + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
7. 480V Emergency Motor Control Center MCC-7B1	Elec	CS + 20'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
ELECTRICAL DISTRIBUTION SYSTEMS (Continued)							
8. 480V Emergency Motor Control Center MCC-8A	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
9. 480V Emergency Motor Control Center MCC-8B	Elec	RMC + 36'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
10. 480V Emergency Motor Control Center MCC-8B1	Elec	CS + 20'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
11. 120V AC Vital Bus (Wall mounted) Bus 1A	Elec	SB + 21'		X		>0.3g	Bus HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
12. 120V AC Vital Bus (Wall mounted) Bus 2A	Elec	SB + 21'		X		>0.3g	Bus HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
13. 120V AC Vital Bus (Wall mounted) Bus 3A	Elec	SB + 21'		X		>0.3g	Bus HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
14. 120V AC Vital Bus (Wall mounted) Bus 4A	Elec	SB + 21'		X		>0.3g	Bus HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
15. 125V DC Bus Distribution Cabinet Bus 1	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
ELECTRICAL DISTRIBUTION SYSTEMS (Continued)							
16. 125V DC Bus Distribution Cabinet Bus 2	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
17. 125V DC Bus Distribution Cabinet Bus 3	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
18. 125V DC Bus Distribution Cabinet Bus 4	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
19. Station Battery No. 1 (1) New Lead Calcium Batteries	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
20. Station Battery No. 2 (Lead Antimony)	Elec	SB + 46'			X	(2)	
21. Station Battery No. 3 (1) New Lead Calcium Batteries	Elec	SB + 35'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
22. Station Battery No. 4 (Lead Antimony)	Elec	SB + 35'			X	(2)	

(1) New Lead Calcium Batteries being installed during the next MYC outage (Spring 1987), see Section 4.4.

(2) Existing MYC Lead Antimony batteries not scheduled to be changed out during the next outage. Lacking data on the seismic performance of lead antimony batteries precludes the determination of a HCLPF Capacity, see Section 4.4.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
ELECTRICAL DISTRIBUTION SYSTEMS (Continued)							
23. Battery Charger No. 1 BC-1	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
24. Battery Charger No. 2 BC-2	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
25. Battery Charger No. 3 BC-3	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
26. Battery Charger No. 4 BC-4	Elec	SB + 46'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
27. Inverter No.1 INVR-1	Elec	SB + 46'			X	FA 0.82g CDFM 1.13g	
28. Inverter No.2 INVR-2	Elec	SB + 46'			X	FA 0.82g CDFM 1.13g	
29. Inverter No.3 INVR-3	Elec	SB + 46'			X	FA 0.82g CDFM 1.13g	
30. Inverter No.4 INVR-4	Elec	SB + 46'			X	FA 0.82g CDFM 1.13g	

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
ELECTRICAL DISTRIBUTION SYSTEMS (Continued)							
31. Station Service Transformer X-507 (located adjacent to Bus 7 & 8)	Elec	SB + 46'			X	0.30g	HCLPF Calculated for Modified Anchorage Configuration.
32. Station Service Transformer X-608 (located adjacent to Bus 7 & 8)	Elec	SB + 46'			X	0.30g	HCLPF Calculated for Modified Anchorage Configuration.
33. Diesel Generator 1A 480V Distribution Panel (wall mounted)	Elec	AB + 22'		X		>0.3g	Panel HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
34. Diesel Generator 1B 480V Distribution Panel (wall mounted)	Elec	AB + 22'		X		>0.3g	Panel HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
35. Diesel Generator Control Panel 1A	Elec	AB + 22'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
36. Diesel Generator Control Panel 1B	Elec	AB + 22'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
37. Distribution Cabinet 1 (wall mounted) DC-1	Elec	SB + 21'		X		>0.3g	Cabinet HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
38. Distribution Cabinet 2 (wall mounted) DC-2	Elec	SB + 21'		X		>0.3g	Cabinet HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Walkdown Review	Calculation Review	HCLPF Capacity	Comments
ELECTRICAL DISTRIBUTION SYSTEMS (Continued)							
39. Distribution Cabinet 3 (wall mounted) DC-3	Elec	SB + 21'		X		>0.3g	Cabinet HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
40. Distribution Cabinet 4 (wall mounted) DC-4	Elec	SB + 21'		X		>0.3g	Cabinet HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
41. Main Control Board 120V AC Vital Bus 1-4	Elec	SB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
42. Electrical Control Board DG-1A & 1B Start 1 & 2 Circuits and Control Power	Elec	SB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
43. Auxiliary Logic Cabinets	Elec	SB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
44. ESF Auxiliary Panels A & B	Elec	SB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
45. Air Condition Control Panel ACCP	Elec	SB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
46. Safety Parameter Display System Cabinets	Elec	SB + 21'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening			HCLPF Capacity	Comments
			NUREG 4334	Walkdown Review	Calculation Review		
HVAC							
1. DG-1A Room Exhaust Fan FN-20A	HV	AB + 31'		X		>0.3g	Fan HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
2. DG-1B Room Exhaust Fan FN-20B	HV	AB + 31'		X		>0.3g	Fan HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
3. Computer Room Air Conditioner * AC-1A	SCC	SB + 39'			X	0.38g	HCLPF Calculated for Modified Support Configuration.
4. Computer Room Air Conditioner * AC-1B	PCC	SB + 39'			X	0.38g	HCLPF Calculated for Modified Support Configuration.
5. Lab Air Conditioner * AC-2	SCC	SB + 39'			X	0.38g	HCLPF Calculated for Modified Support Configuration.
6. Diesel Generator Air Intake & Exhaust Dampers	DG	AB + 31'		X		>0.3g	Damper HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
7. Containment Spray Fan FN-44A (Modified Anchorage)	SCC	CS + 20'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.
8. Containment Spray Fan FN-44B (Modified Anchorage)	SCC	CS + 20'			X	>0.3g	Conservative Fragility Calculation Determined the Anchorage HCLPF Capacity is >0.3g.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
VALVES							
1. Aux. Feedwater Regulating Valve AOV AFW-A-101	AFW	AF + 23'	C			>0.3g	Valve within bounds of experience data.
2. Aux. Feedwater Regulating Valve AOV AFW-A-201	AFW	AF + 23'	C			>0.3g	Valve within bounds of experience data.
3. Aux. Feedwater Regulating Valve AOV AFW-A-301	AFW	AF + 23'	C			>0.3g	Valve within bounds of experience data.
4. Block Valve for AFW-A-101 AOV AFW-A-338	AFW	AF + 23'	C			>0.3g	Valve within bounds of experience data.
5. Block Valve for AFW-A-201 AOV AFW-A-339	AFW	AF + 23'	C			>0.3g	Valve within bounds of experience data.
6. Block Valve for AFW-A-301 AOV AFW-A-340	AFW	AF + 23'	C			>0.3g	Valve within bounds of experience data.
7. HPSI Pump B Discharge to charging header CH-A-32 AOV	HPSI	PT + 13'	C			>0.3g	Valve within bounds of experience data.
8. HPSI Pump A Discharge to charging header CH-A-33 AOV	HPSI	PT + 13'	C			>0.3g	Valve within bounds of experience data.
9. Inlet to Charging Header AOV * * CH-F-38	HPSI	PAB + 18'	C			>0.3g	Valve within bounds of experience data.
10. VCT Discharge to HPSI Pumps MOV CH-M-1	CH	PAB + 24'	C			>0.3g	Valve within bounds of experience data.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
VALVES (Continued)							
11. VCT Discharge to HPSI Pumps MOV CH-M-87	CH	PAB + 24'	C			>0.3g	Valve within bounds of experience data.
12. Containment Spray Header Isolation Valve CS-M-1 MOV	CS	CS + 19'	C			>0.3g	Valve within bounds of experience data.
13. Containment Spray Header Isolation Valve CS-M-2 MOV	CS	CS + 19'	C			>0.3g	Valve within bounds of experience data.
14. CS Pump Containment Suction MOV CS-M-91	CS	CS - 08'	C			>0.3g	Valve within bounds of experience data.
15. CS Pump Containment Suction MOV CS-M-92	CS	CS - 08'	C			>0.3g	Valve within bounds of experience data.
16. HPSI Discharge to Loop 1 MOV HSI-M-11	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.
17. HPSI Discharge to Loop 1 MOV HSI-M-12	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.
18. HPSI Discharge to Loop 2 MOV HSI-M-21	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Walkdown Review	Calculation Review	HCLPF Capacity	Comments
VALVES (Continued)							
19. HPSI Discharge to Loop 2 MOV HSI-M-22	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.
20. HPSI Discharge to Loop 3 MOV HSI-M-31	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.
21. HPSI Discharge to Loop 3 MOV HSI-M-32	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.
22. HPSI Discharge to SI Header MOV HSI-M-40	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.
23. HPSI Pump Discharge MOV HSI-M-41	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.
24. HPSI Pump Discharge MOV HSI-M-42	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.
25. HPSI Discharge to SI Header MOV HSI-M-43	HPSI	PAB + 23'	C	X		>0.3g	Valve exceeds experience data bounds by 17%.
26. HPSI Suction from RWST MOV HSI-M-50	HPSI	Yd + 21'	C			>0.3g	Valve within bounds of experience data.
27. HPSI Suction from RWST MOV HSI-M-51	HPSI	Yd + 21'	C			>0.3g	Valve within bounds of experience data.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
VALVES (Continued)							
28. CS Discharge to HPSI Pump MOV HSI-M-54	CS	CS + 19'	C			>0.3g	Valve within bounds of experience data.
29. CS Discharge to HPSI Pump MOV HSI-M-55	CS	CS + 19'	C			>0.3g	Valve within bounds of experience data.
30. RWST Discharge to LPSI MOV LSI-M-40	CS	Yd. + 28'	C			>0.3g	Valve within bounds of experience data.
31. RWST Discharge to LPSI MOV LSI-M-41	CS	Yd. + 28'	C			>0.3g	Valve within bounds of experience data.
32. Decay Heat Release Valve AOV MS-A-162	ASDA	VA + 43'	C			>0.3g	Valve within bounds of experience data.
33. AFW Pump B Turbine Throttle Valve MS-A-173	AFW	VA + 21'	C			>0.3g	Valve within bounds of experience data.
34. Decay Heat Release MOV MS-M-161	SPC	VA + 43'	C			>0.3g	Valve within bounds of experience data.
35. Auxiliary Steam Supply Valve MOV MS-M-255	MS	VA + 43'	C			>0.3g	Valve within bounds of experience data.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
VALVES (Continued)							
36. Turbine Steam Supply Pressure Control MS-P-168	AFW	VA + 21'	C			>0.3g	Valve within bounds of experience data.
37. Turbine Steam Supply Temperature Control MS-T-163 AOV	ASDA	VA + 41'	C			>0.3g	Valve within bounds of experience data.
38. PCCW Supply to P-12A Seal Cooler AOV PCC-A-53	PCC	PAB + 04'	C			>0.3g	Valve within bounds of experience data.
39. Return from Penetration Coolers AOV PCC-A-216	PCC	PT + 12'	C			>0.3g	Valve within bounds of experience data.
40. Return from Penetration Coolers AOV PCC-A-238	PCC	PT + 12'	C			>0.3g	Valve within bounds of experience data.
41. Return from CEA Air Coolers PCC-A-270	PCC	RC + 01'	C			>0.3g	Valve within bounds of experience data.
42. DG-1A Cooling Water Outlet AOV PCC-A-493	PCC	AB + 21'	C			>0.3g	Valve within bounds of experience data.
43. PCCW Outlet from RHR Heat Exchanger PCC-M-43	PCC	CS + 02'	C			>0.3g	Valve within bounds of experience data.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
VALVES (Continued)							
44. PCCW Isolation to BR & LW Coolers MOV PCC-M-90	PCC	PAB + 11'	C			>0.3g	Valve within bounds of experience data.
45. PCCW Isolation to Letdown Heat Exchangers PCC-M-150 MOV	PCC	PAB + 21'	C			>0.3g	Valve within bounds of experience data.
46. PCCW Isolation to Containment PCC-M-219 MOV	PCC	PAB + 11'	C			>0.3g	Valve within bounds of experience data.
47. Cooler Supply Temperature Control AOV PCC-T-19	PCC	TB + 37'	C			>0.3g	Valve within bounds of experience data.
48. Cooler Bypass Temperature Control AOV PCC-T-20	PCC	TB + 37'	C			>0.3g	Valve within bounds of experience data.
49. Pressurizer Safety Valve PR-S-11	SRV	RC + 65'	C			>0.3g	Vendor data review found valve within bounds of experience data.
50. Pressurizer Safety Valve PR-S-12	SRV	RC + 65'	C			>0.3g	Vendor data review found valve within bounds of experience data.
51. Pressurizer Safety Valve PR-S-13	SRV	RC + 65'	C			>0.3g	Vendor data review found valve within bounds of experience data.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Walkdown Review	Calculation Review	HCLPF Capacity	Comments
VALVES (Continued)							
52. Power-Operated Relief Valve PR-S-14	PORV	RC + 66'	C			>0.3g	Vendor data review found valve within bounds of experience data.
53. Power-Operated Relief Valve PR-S-15	PORV	RC + 66'	C			>0.3g	Vendor data review found valve within bounds of experience data.
54. Power-Operated Block Valve MOV PR-M-16	PORV	RC + 64'	C			>0.3g	Vendor data review found valve within bounds of experience data.
55. Power-Operated Block Valve MOV PR-M-17	PORV	RC + 64'	C			>0.3g	Vendor data review found valve within bounds of experience data.
56. Non-Seismic Return Header Stop Valve AOV SCC-A-460	SCC	TB + 21'	C			>0.3g	Valve within bounds of experience data.
57. Non-Seismic Return Header Stop Valve AOV SCC-A-461	SCC	TB + 43'	C			>0.3g	Valve within bounds of experience data.
58. Cooler Bypass Temperature Control AOV SCC-T-23	SCC	TB + 37'	C			>0.3g	Valve within bounds of experience data.
59. Cooler Bypass Temperature Control AOV SCC-T-24	SCC	TB + 37'	C			>0.3g	Valve within bounds of experience data.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments
VALVES (Continued)							
60. DG-1B Cooler Inlet Temperature Control SCC-T-305 AOV	**SCC**	**AB + 23'**	**C**			**>0.3g**	**Valve within bounds of experience data.**
61. RCP Seal Water Inlet AOV * * SL-P-3	**PCC**	**SB + 23'**	**C**			**>0.3g**	**Valve within bounds of experience data.**
PIPING				X		>0.3g	Piping HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
ACTUATIONS							
1. Instrument Racks	various	various		X		>0.3g	Racks HCLPF Capacity Judged >0.3g Assessed During Walkdown Review.
2. Cable Trays & Conduit	various	various		X		>0.3g	Cable Tray & Conduit HCLPF Judged >0.3g Assessed During Walkdown Review.
3. Impulse Lines (EI)							

* Component whose failure may breach critical system pressure boundary.

* * Valves PCC-M-90 and PCC-M-219 have been determined to have a capacity of 0.3 g's or higher, thus, these components do not require a seismic review as the valves will isolate all components downstream from these valves.

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Walkdown Review	Calculation Review	HCLPF Capacity	Comments

Legend:

SYSTEM

AFW	Auxiliary Feedwater
ASDA	Alternate Shutdown Decay Heat Removal
CH	Charging
CS	Containment Spray
DG	Diesel Generator Starting
FO	Fuel Oil
HPSI	High Pressure Safety Injection
HV	Area Heating and Ventilation
PCC	Primary Component Cooling
PORV	Power-Operated Relief Valve
SCC	Secondary Component Cooling

SYSTEM

SL	Seal Water
SPC	Secondary Pressure Control
SRV	Safety Relief Valves
SW	Service Water

Table 7.1-3 Summary of Maine Yankee Equipment Screening and HCLPF Capacities (Continued)

Equipment Item	System	Building and Elevation	Equipment Qualification Screening: NUREG 4334	Equipment Qualification Screening: Walkdown Review	Equipment Qualification Screening: Calculation Review	HCLPF Capacity	Comments

Legend: (Continued)

Building

AB	Turbine Building Auxiliary Bay
AF	Auxiliary Feed Pump House
CS	Containment Spray Pump House
CW	Circulation Water Pump House
PAB	Primary Auxiliary Building
PT	Pipe Tunnel
PV	Purge Air Valve Room
RC	Reactor Coolant
RMC	Reactor Motor Control Center Room
SB	Service Building
TB	Turbine Building
VA	Steam and Feed Water Valve Area
YD	Yard

The screened-in components are the ones that are left in in the final event and fault trees using which the accident sequence Boolean expressions are derived. The seismic fragility parameters and the HCLPF capacities of these components are given in Table 5.5-1 in Chapter 5. The nonseismic unavailabilites are listed in Table 5.5-2 in Chapter 5.

7.3 HCLPF Capacity of Plant

Two dominant accident sequences that could lead to core damage were studied in this review: small LOCA and transient. The Boolean expressions were derived for these sequences consisting of seismic failures and nonseismic unavailabilities. The seismic failures were quantified by the seismic fragilites reported in Table 5.5-1. For the accident sequence of small LOCA, the HCLPF capacity was determined to be 0.21 g pga. This is almost entirely governed by the failure of the refueling water storage tank. For the accident sequence of no LOCA, the HCLPF capacity was determined to be in the range of 0.32 g to 0.33 g depending on whether the nonseismic unavailabilities are considered or not. The component failures contributing to this sequence are the transformers, and circulating water pumphouse.

These accident sequences could not be combined together to estimate the core damage HCLPF capacity since the split fractions for the two dominant accident sequences are not known. Parametric studies were done wherein the different split fractions were assumed. The resulting HCLPF capacities were seen to vary from 0.23 g to 0.32 g for split fraction ranging from 50% to 1% for the small LOCA.

7.4 Identification of Low Capacity Components

During the course of this review, certain items were identifed tobe potential seismic vulnerabilites at Maine Yankee. Anong these are the aged lead antimony batteries, the internal seismic supports for transformers, vibration-isolation supports for containment spray fans, anchorage of diesel day tank, and block wall near the containment spray fans. The utility has proposed that certain modifications would be made for the components in the next outage in March 1987. The seismic capacities of these components have been estimated in this study in their upgraded condition.

7.5 Conclusions

This seismic margin review has been performed with the following assumptions and limitations:

- The review earthquake level was specified by the NRC as the NUREG/CR-0098 median spectrum. We have interpreted this spectrum to be a 84 percent confidence site specific spectrum for Maine Yankee. The spectral values are treated as corresponding to the higher ones from the two orthogonal horizontal directions.

- The structural models and the floor response spectra generated by Maine Yankee have been cursorily reviewed by EQE and judged to be adequate for the purposes of this margin review.

- In general EQE is in agreement with the Panel's recommendations; where the Panel's recommendations were not specific or did not cover a particular item, they have been identified and the review has been accomplished based on our own experience from past seismic PRAs, earthquake damage investigations, and judgment.

- Since the analysis team could not perform the walkdown inside the containment, the seismic capacity of components inside the containment could not be determined. In some instances, Maine Yankee provided photographs for certain equipment items inside the containment. However, we could not confirm the absence of potential system interaction effects that may make the impulse lines inside the containment vulnerable in earthquakes and lead to a small LOCA.

- Our review and walkdowns concentrated on components supporting Group A system functions; however, a review of non-Group A system components (block walls, for example) led us to believe that there are no anomolies at Maine Yankee which would violate the Expert Panel systems screening guidelines.

- In keeping with the Expert Panel philosophy, the screening of components was performed using conservative procedures. For the screened-in components, the seismic capacities have been calculated using conservative methods since the scope did not permit detailed response analysis and thorough investigation of failure modes and their capacities. In all cases, the factors contributing to the seismic margin and their varilabilities are identified and quantified using procedures normally used within the state-of-the-art.

- The plant level HCLPF capacity was calculated based on two accident sequences: small LOCA and no LOCA. Since the small LOCA could not be dismissed at this review earthquake level of 0.30g, it was postulated to occur: the contributing component to this sequence HCLPF capacity is RWST. Sensitivity studies indicate that the plant level HCLPF capacity is higher than the HCLPF capacity of the RWST.

- The HCLPF capacity of the plant has been determined based on the seismic capacities of components in their existing or proposed modified conditions. Maine Yankee has proposed that certain modifications or replacements would be made for station batteries, transformer internal core/coil assembly anchorage, vibration-isolation supports for containment spray fans and air conditioners, anchorage of diesel day tank, and block wall near the containment spray fans. We recommend that a third walkdown of the plant be performed to assure that these modifications have been carried out and that the HCLPF capacities of these components calculated in the present study are still applicable.

Based on this margin review and subject to the above restrictions, EQE confirms that the high confidence low probability of failure capacity of Maine Yankee plant is at least 0.21g refered to the ground response spectrum specified above; it is also our judgment that the actual capacity of the plant is much higher since the assumption of a small LOCA occuring is conservative.

REFERENCES

American Concrete Institute (1981), Building Code Requirements for Concrete Masonry Structures (ACI 531-79) (Revised 1981) and Commentary - ACI 531R-79.

ANCO, Engineers, Inc., Generic Seismic Qualification Using Existing Test Data,

Bernreuter, D.L., J.B. Savy, R.W. Mensing, J.C. Chen, and B.C. Davis (April 1985), Seismic Hazard Characterization of the Eastern United States Volume 1: Methodology and Results for Ten Sites, UCID-20421 Vol. 1. Lawrence Livermore National Laboratory, Livermore, California.

Budnitz, R.J., et al., (July 1985), An Approach to the Quantification of Seismic Margins in Nuclear Power Plants, NUREG/CR-4334.

Campbell, R.D., R.P. Kennedy, and R.P. Kassawara May 1986, Seismic Margin Review Methodology, Draft EPRI Report.

Coats, D.W., May 1980, Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria, NUREG/CR-1161.

CYGNA Energy Services (April 1982), Dynamic Modeling of Class I Structures, Maine Yankee Atomic Power Station, Report BM-Y-MY-80006-5, Rev. 0.

EQE Incorporated (October 1986), The Performance of Cable Tray and Conduit Systems In Actual and Simulated Seismic Motion, Report 82001.05.001 Draft B, prepared for the Seismic Qualification Utility Group.

Gerard, G. and H. Becker (1957), Handbook of Structural Stability, NACA TN 3783.

Hall, D.H. (April 1981), Proposed Steel Column Strength Criteria, Journal of The Structural Division, ASCE.

Haroun, M.A. and G.W. Housner (April 1981), Seismic Design of Liquid Storage Tanks, Journal of The Structural Division, ASCE.

Hashimoto, P.S., R.P. Kennedy, R.B. Narver, and D.A. Wesley (December 1984), Conservative Seismic Capacities of the Maine Yankee Reactor Containment Including and Excluding Design Incident Pressure, NTS/SMA 36001.01, prepared for Maine Yankee Atomic Power Company by Structural Mechanics Associates.

Henries, W.E., and S.R. Boeschen (March 1986), Evaluation of AOVs and MOVs to a Greater Than Design Base Earthquake in the Maine yankee Nuclear Power Plant, Prepared for Maine Yankee Atomic Power Company.

Kaplan, S. (1981), On the Method of Discrete Probability Distribution in Risk and Reliability Calculations, Risk Analysis, Volume 1.

Kennedy, R.P. and M.K. Ravindra (May 1984), Seismic Fragilities for Nuclear Power Plant Risk Studies, Nuclear Engineering and Design, Vol. 79, No. 1, pp. 47-68.

Manos, G.C. (August 1986), Earthquake Tank-Wall Stability of Unanchored Tanks, Journal of the Structural Engineering, ASCE.

National Aeronautics and Space Administration (June 1966), Shell Analysis Manual.

Newmark, N.M. and W.J. Hall (May 1978), Development of Criteria For Seismic Review of Selected Nuclear Power Plants, NUREG/CR-0098.

Prassinos, P.G., M.K. Ravindra, and J.B. Savy (October 1985), Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants, NUREG/CR-4482, Lawrence Livermore National Laboratory, Livermore, California.

PRA Procedures Guide, NUREG/CR-2300, 2 volumes, January 1983.

Ravindra, M.K., and R.P. Kennedy (August 1983), "Lessons Learned from Seismic PRA Studes," Paper M6/4, Proceedings of the Seventh Conference on Structural Mechanics in Reactor Technology, Chicago, Illinois.

Reed, J.W., R.P. Kennedy, and M.K. Ravindra (December 5, 1986), Memo to Fragility - Hazard Interaction Subgroup Joint Working Group on Seismic PRA, ASCE Dynamic Analysis Committee.

Smith, G.V., An Evaluation of the Yield, Tensile, Creep, and Trupture Strengths of Wrought 304, 316, 321, and 347 Stainless Steels At Elevated Temperatures, ASTM Data Series DS 5S2.

Stevenson, J.D., and W.E. Henries (January 1983), Seismic Review of the Maine Yankee Nuclear Power Plant, Volume I, prepared for the Maine Yankee Atomic Power Company.

Structural Mechanics Associates (February 1983), Seismic Margin Review Midland Energy Center Project, Volume I, Methodology and Criteria, prepared for Consumers Power Company.

URS Corporation/John A. Blume & Associates, Engineers, Seismic Anchorage Guidelines for Nuclear Power Plant Equipment, Volume I: Development and Use of Guidelines, prepared for Electric Power Research Institute, Palo Alto, California.

Weingarten, V.I., E.J. Morgan, and P. Seide (March 1965), Elastic Stability of Thin-Walled Cylindrical and Conical Shells Under Axial Compression, AIAA Journal.

Wozniak, R.S. and W.W. Mitchell (May 9, 1978), Basis of Seismic Design Provisions For Welded Steel Oil Storage Tanks, American Petroleum Institute, 43rd Midyear Meeting, Toronto, Ontario, Canada.

APPENDIX A

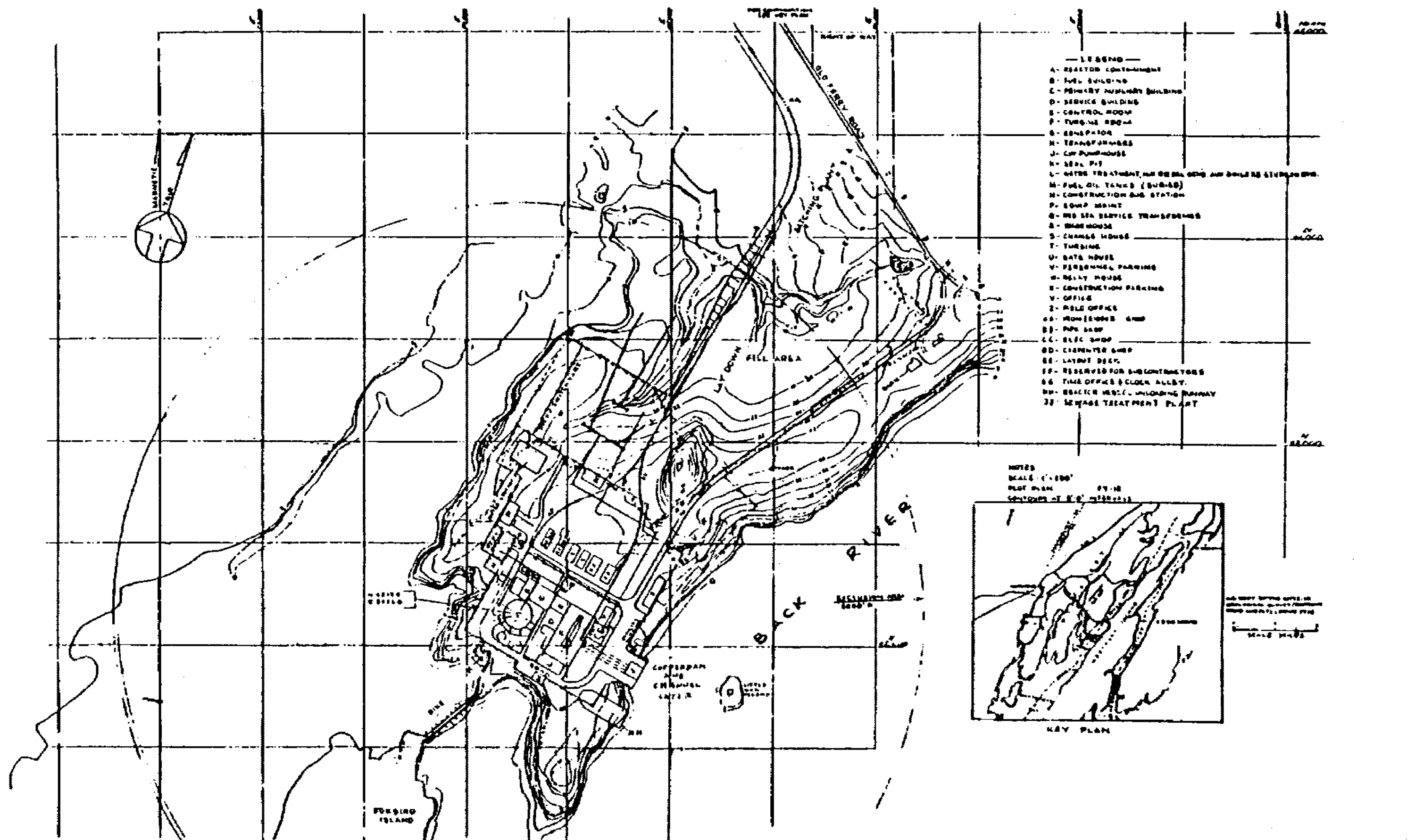
MAINE YANKEE ATOMIC POWER STATION

ARRANGEMENT DRAWINGS

CONTENTS

STRUCTURE

Site Plan A-2
Plot Plan A-3
Reactor Containment Plan, El. 46'-0" A-4
Containment Plan, El. 20'-0" A-5
Primary Auxiliary Building Arrangement A-6
Primary Auxiliary Building Arrangement A-7
Turbine Building El. 21'-0" A-8
Turbine Building El. 39'-0" A-9
Turbine Building Plan A-10
Service Building Addition A-11
Circulating Water Pump House A-12
Turbine Building El. 21'-0" A-13



REV. 1

FIG. 1.3-1
SITE PLAN
MAINE YANKEE ATOMIC POWER COMPANY

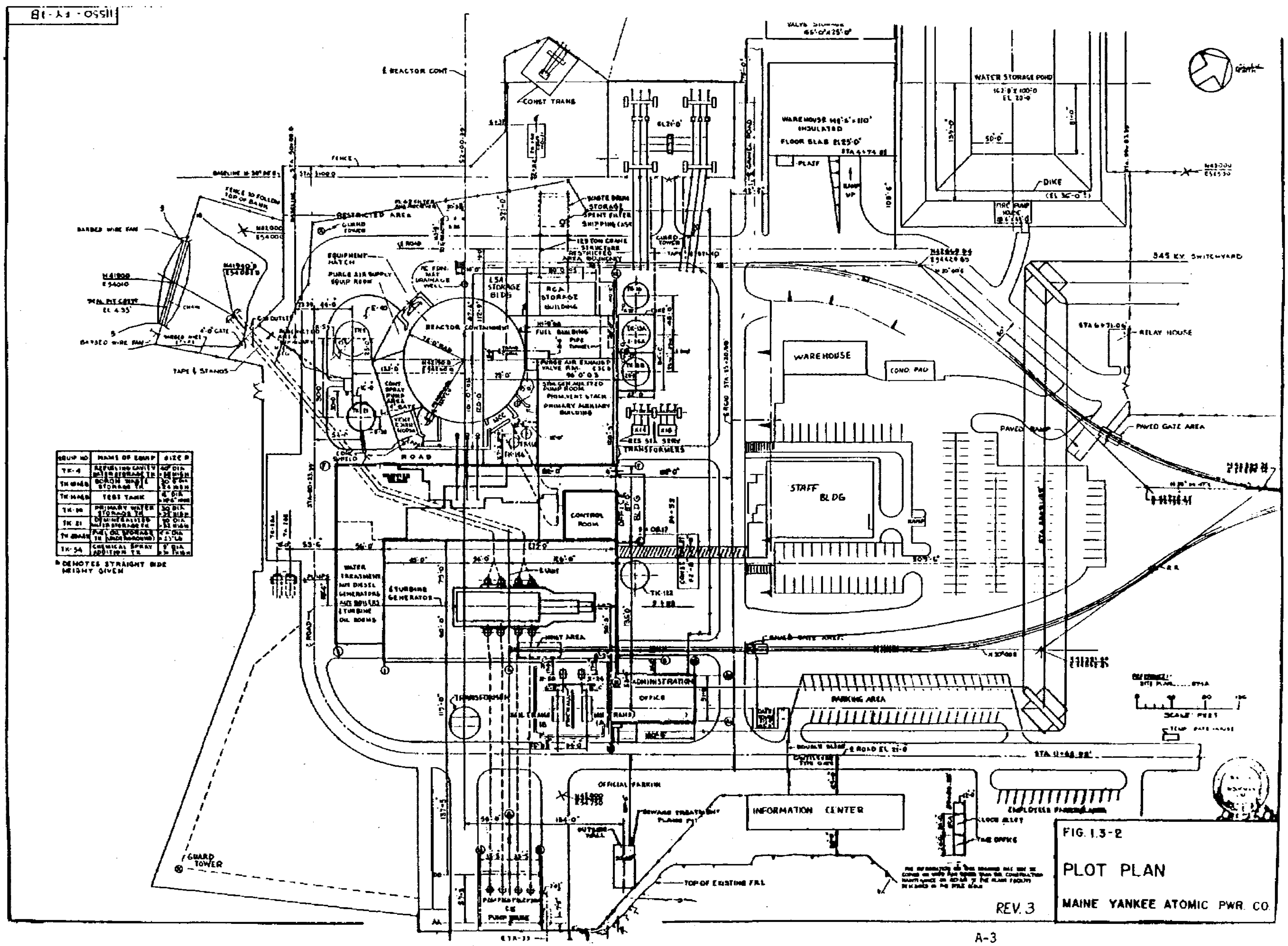


FIG. 1.3-2
PLOT PLAN
MAINE YANKEE ATOMIC PWR. CO.
REV. 3
11550 - FY - 1B
REACTOR CONTAINMENT
WAREHOUSE
STAFF BLDG
INFORMATION CENTER
CONTROL ROOM
ADMINISTRATION
OFFICE
PARKING AREA
TRANSFORMERS
TURBINE GENERATORS
WATER STORAGE POND
DIKE
RELAY HOUSE
345 KV SWITCHYARD
PAVED GATE AREA
PAVED RAMP
COND. PAD
LSA STORAGE BLDG
RCA STORAGE BUILDING
FUEL BUILDING
PRIMARY AUXILIARY BUILDING
RESTRICTED AREA
GUARD TOWER
CONST TRANS
EQUIPMENT HATCH
PURGE AIR SUPPLY EQUIP ROOM
OFFICIAL PARKING
EMPLOYEES PARKING AREA
TOP OF EXISTING FILL
TAPE & STAKES
BARBED WIRE FAN
SCALE: FEET

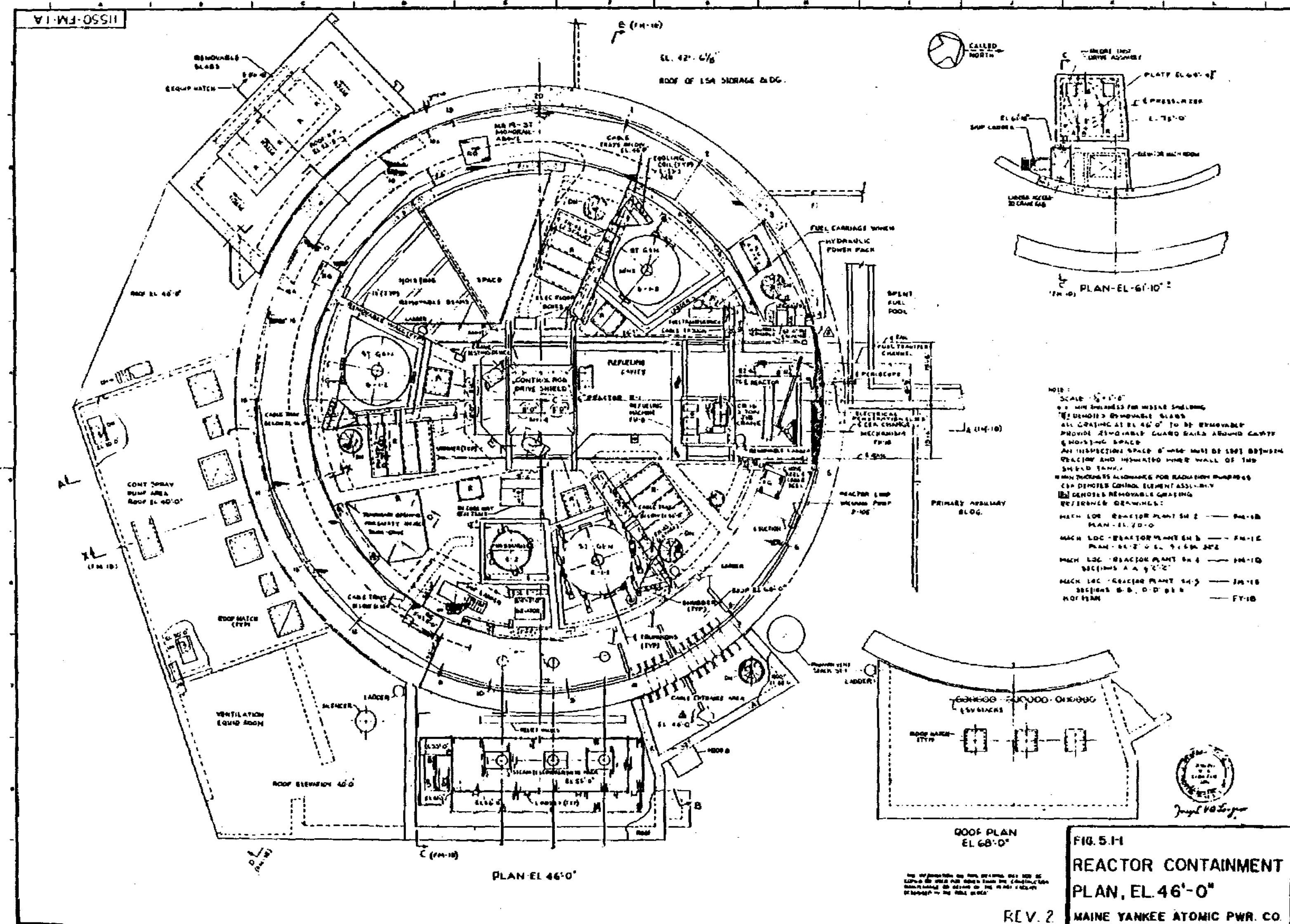
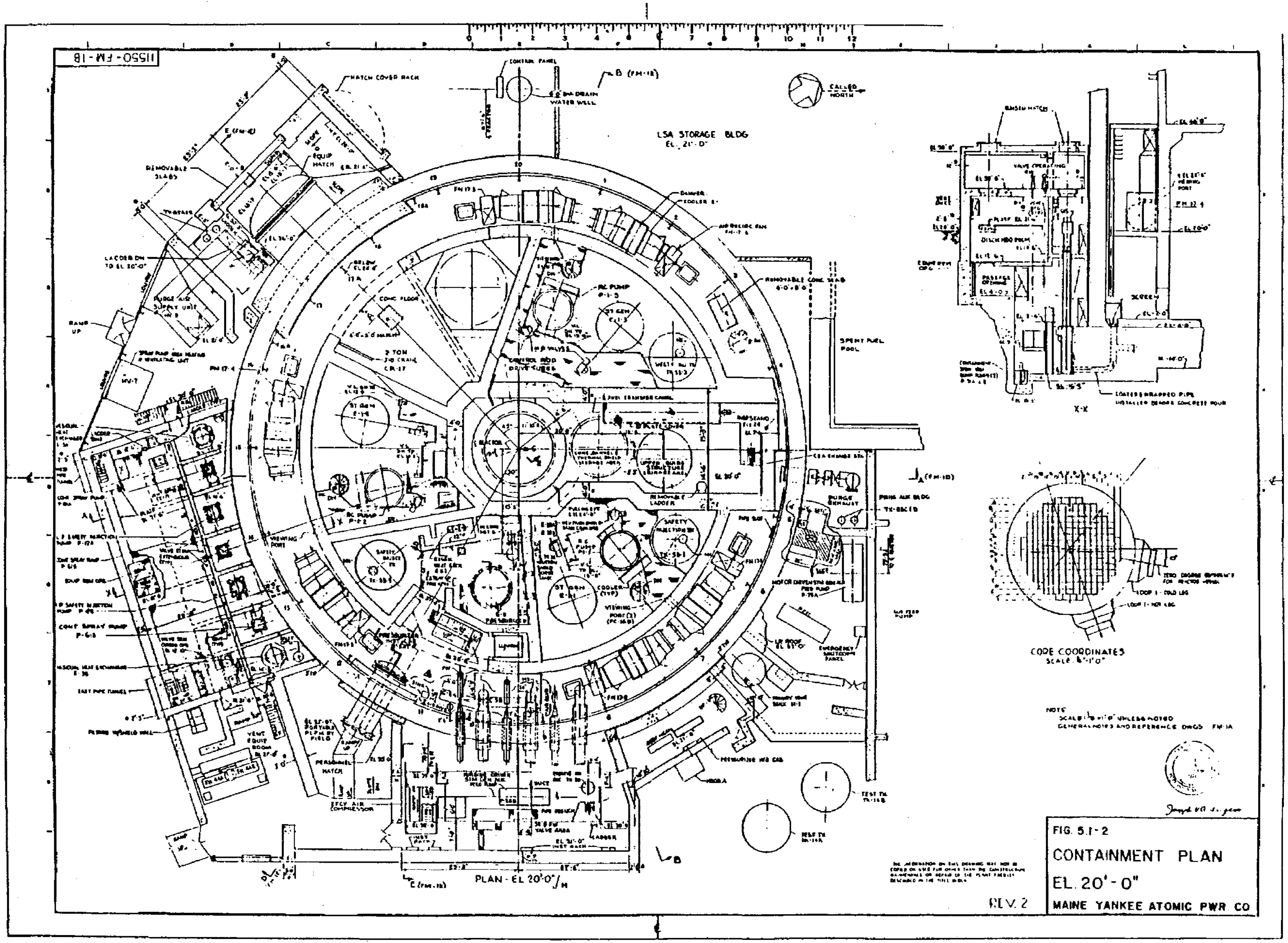
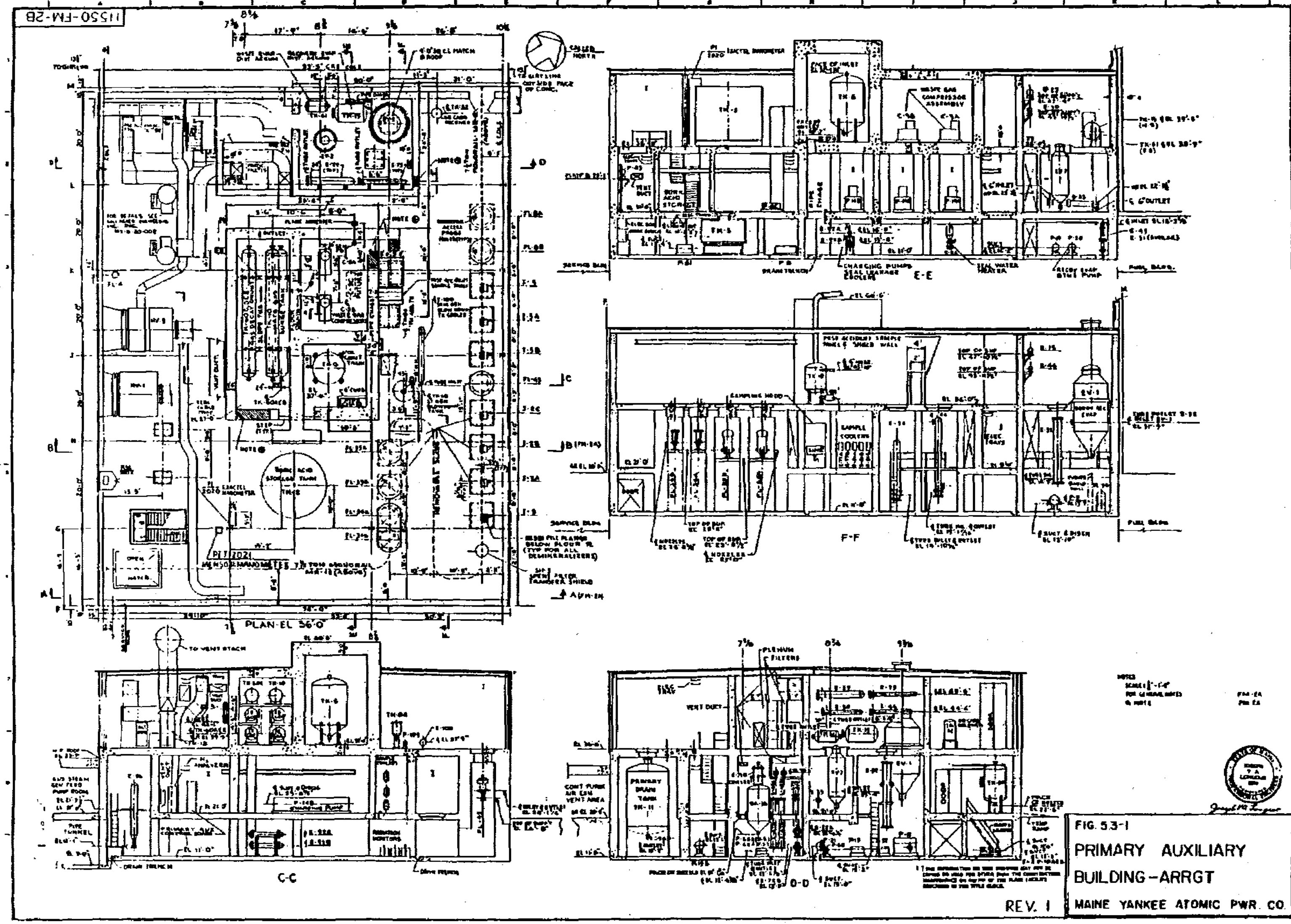


FIG. 5.1-1
REACTOR CONTAINMENT
PLAN, EL. 46'-0"
MAINE YANKEE ATOMIC PWR. CO.


11550-FM-1B
LSA STORAGE BLDG
EL. 21'-0"
CALLED NORTH
SPENT FUEL POOL
PLAN - EL 20'-0"
X-X
CORE COORDINATES
SCALE 1"=1'-0"
REV. 2
FIG. 5.1-2
CONTAINMENT PLAN
EL. 20'-0"
MAINE YANKEE ATOMIC PWR CO


11550-FM-2B
PLAN-EL 56'-0"
E-E
F-F
C-C
D-D
FIG. 53-1
PRIMARY AUXILIARY
BUILDING-ARRGT
MAINE YANKEE ATOMIC PWR. CO.
REV. 1

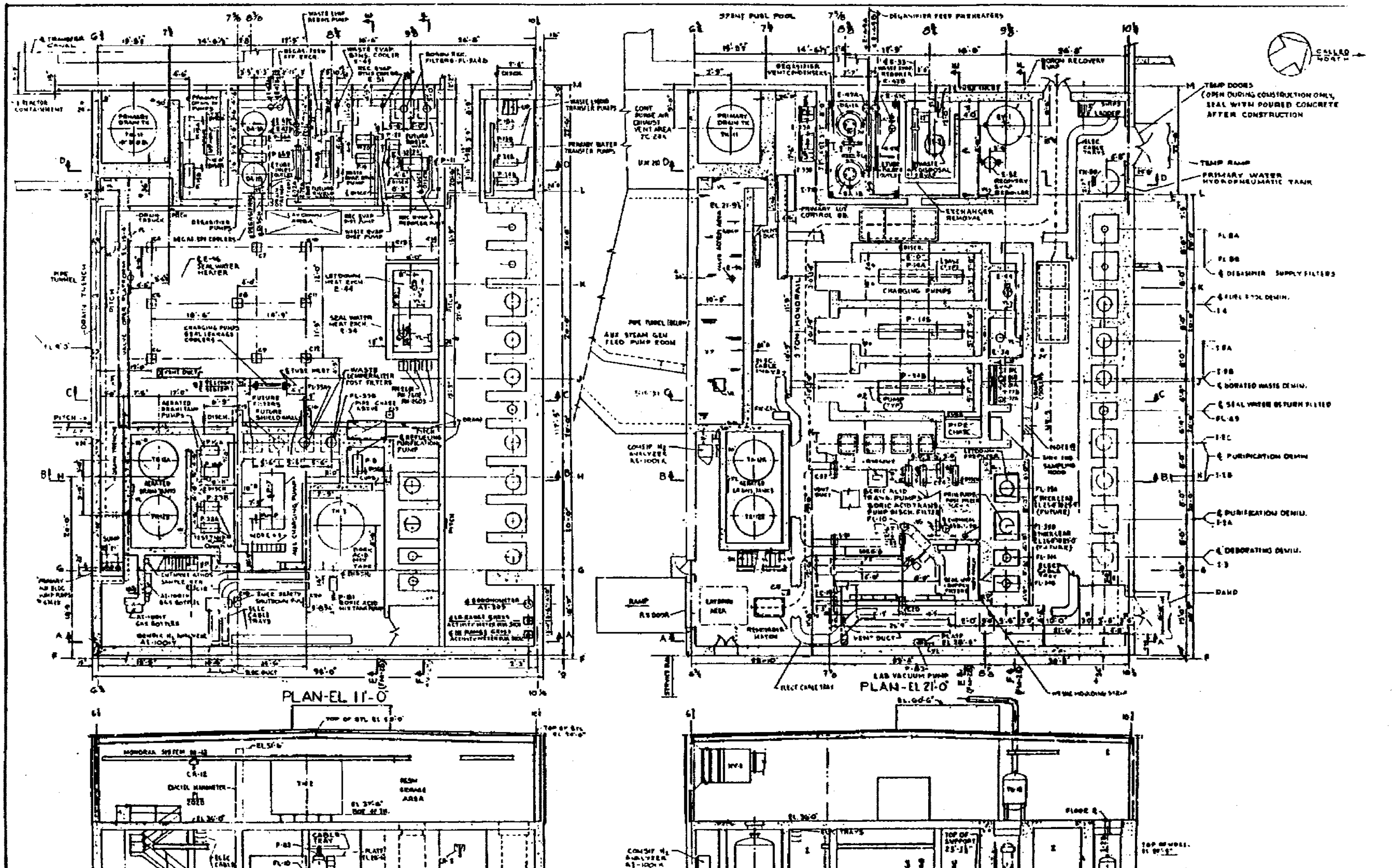


FIG. 5.3-2

PRIMARY AUXILIARY BUILDING ARRNGMT

MAINE YANKEE ATOMIC PWR CO

A-7

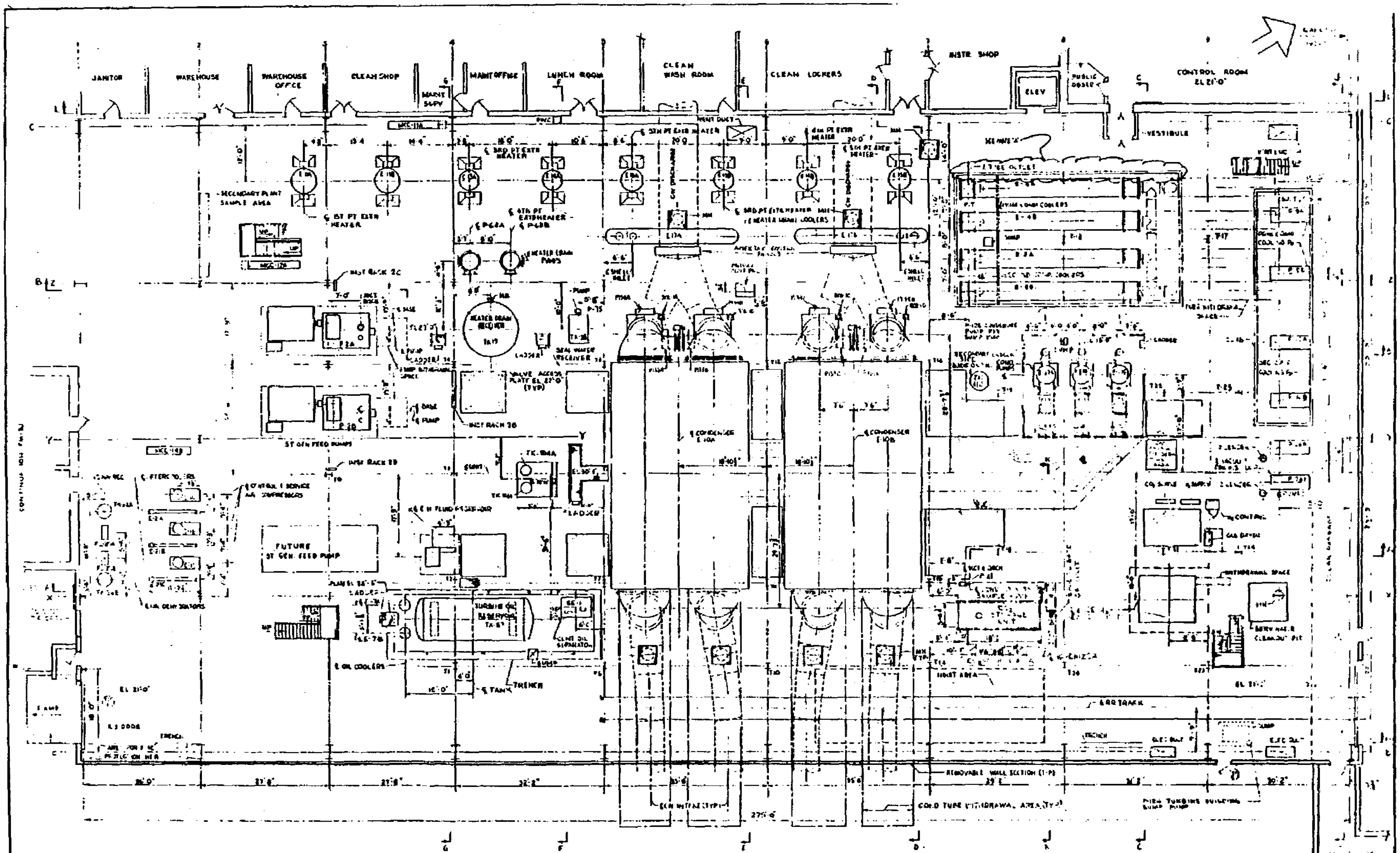
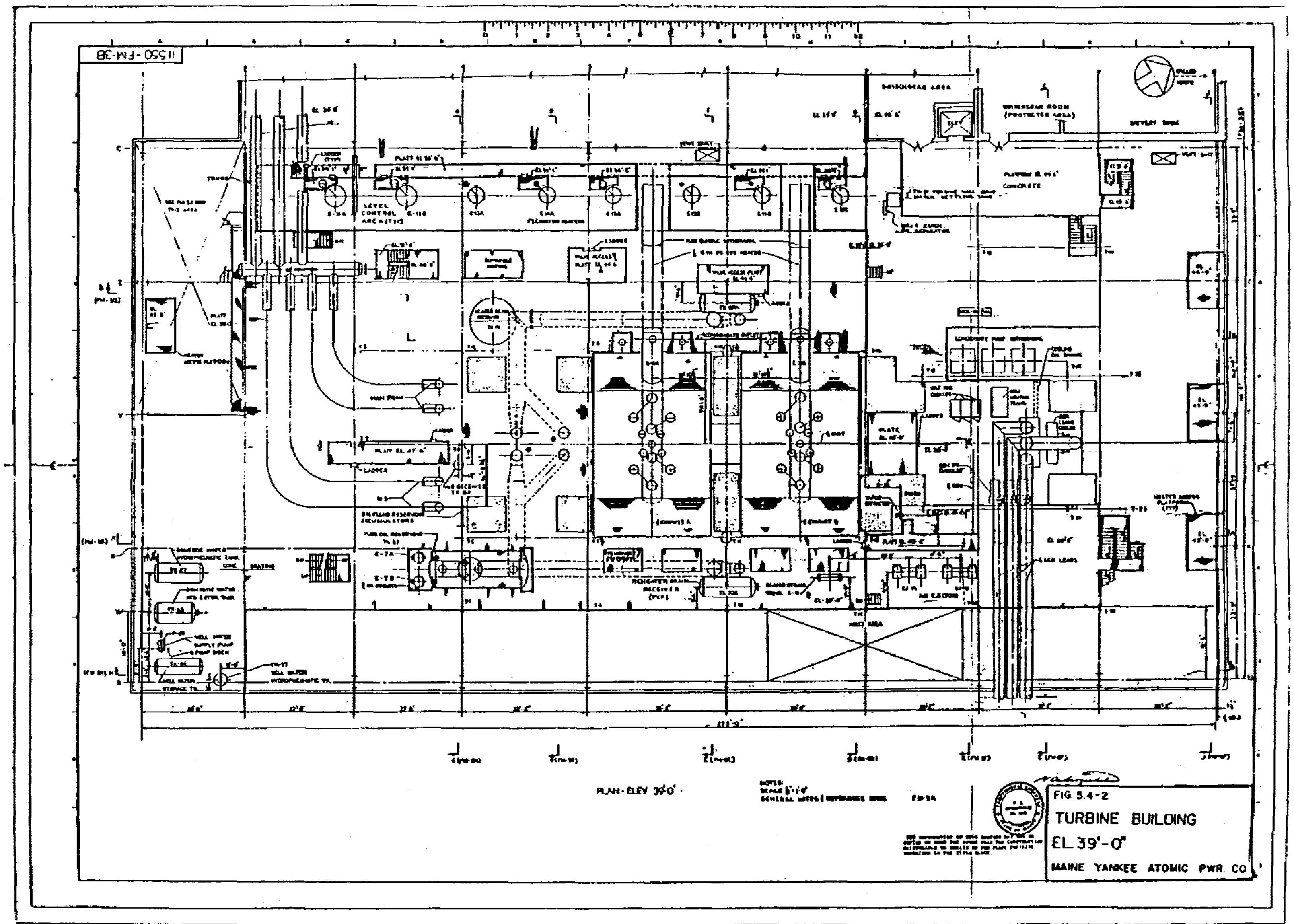


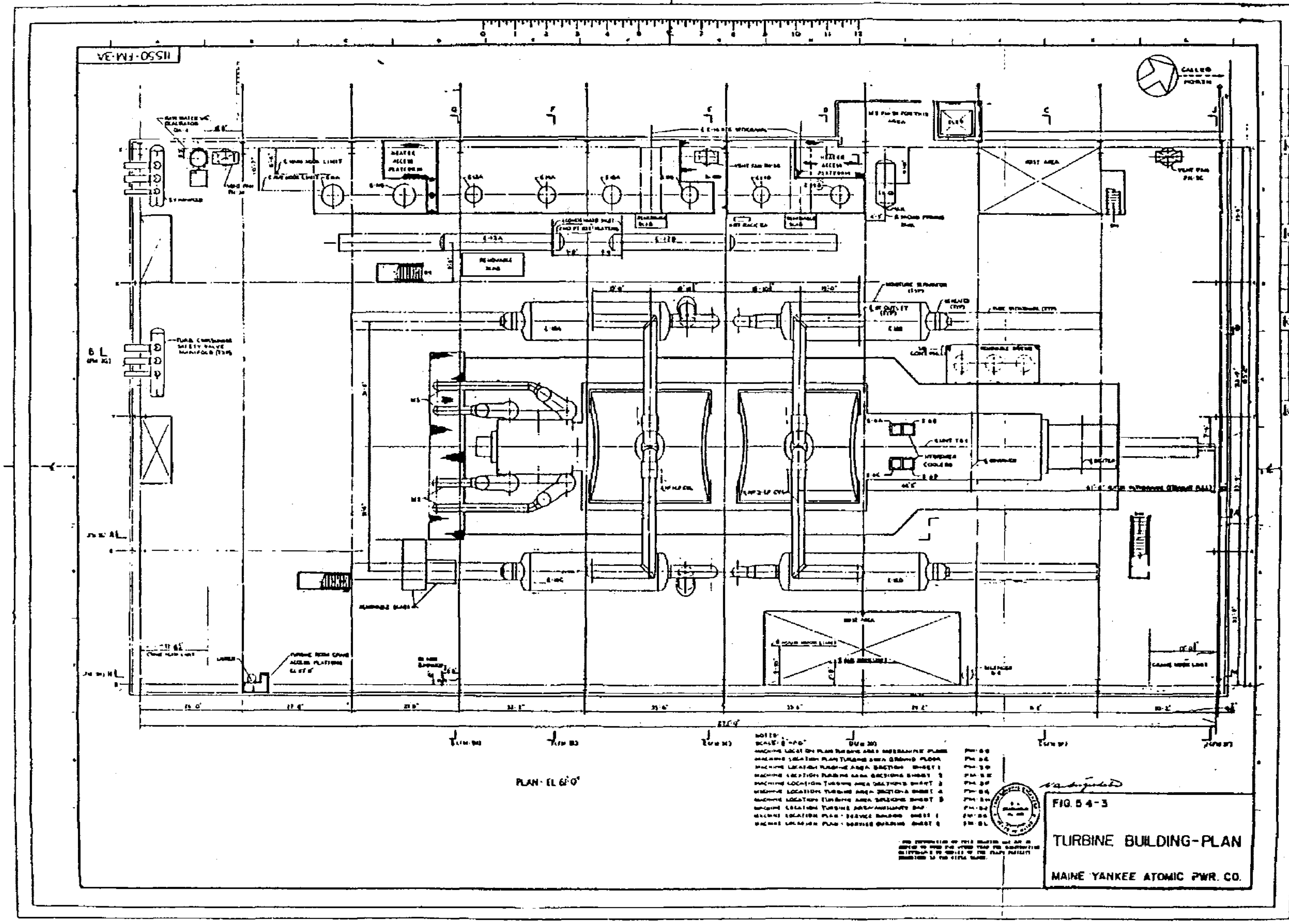
FIG. 5.4-1

TURBINE BUILDING ELEVATION 21 FEET

MAINE YANKEE ATOMIC PWR CO

Rev. 3


11550-FM-3B
LEVEL CONTROL ARCA (TYP)
PLAN - ELEV 39'-0"
FIG. 5.4-2
TURBINE BUILDING
EL 39'-0"
MAINE YANKEE ATOMIC PWR. CO.


11550-FM-3A
PLAN · EL 61'0"
FIG. 5 4-3
TURBINE BUILDING-PLAN
MAINE YANKEE ATOMIC PWR. CO.

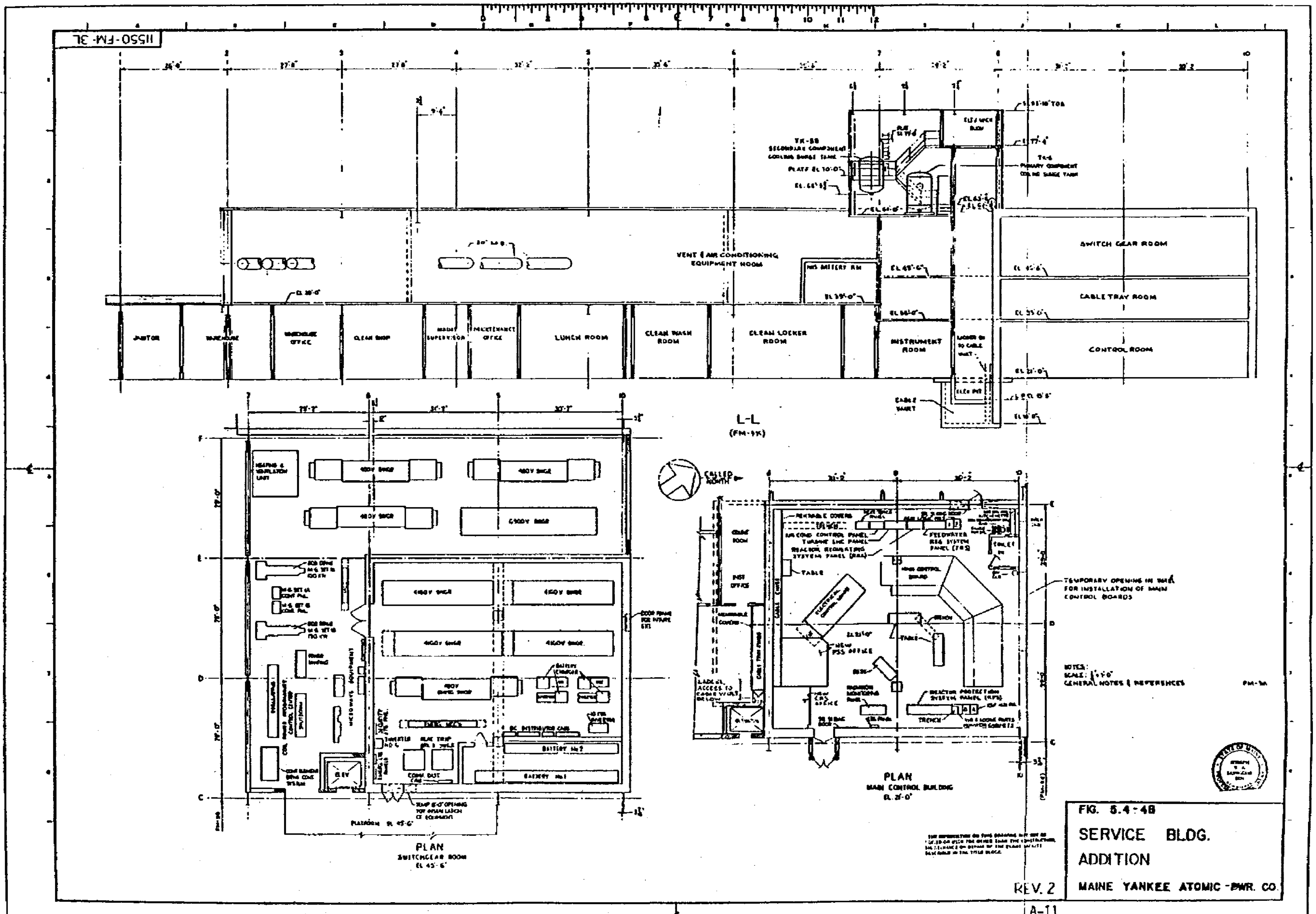


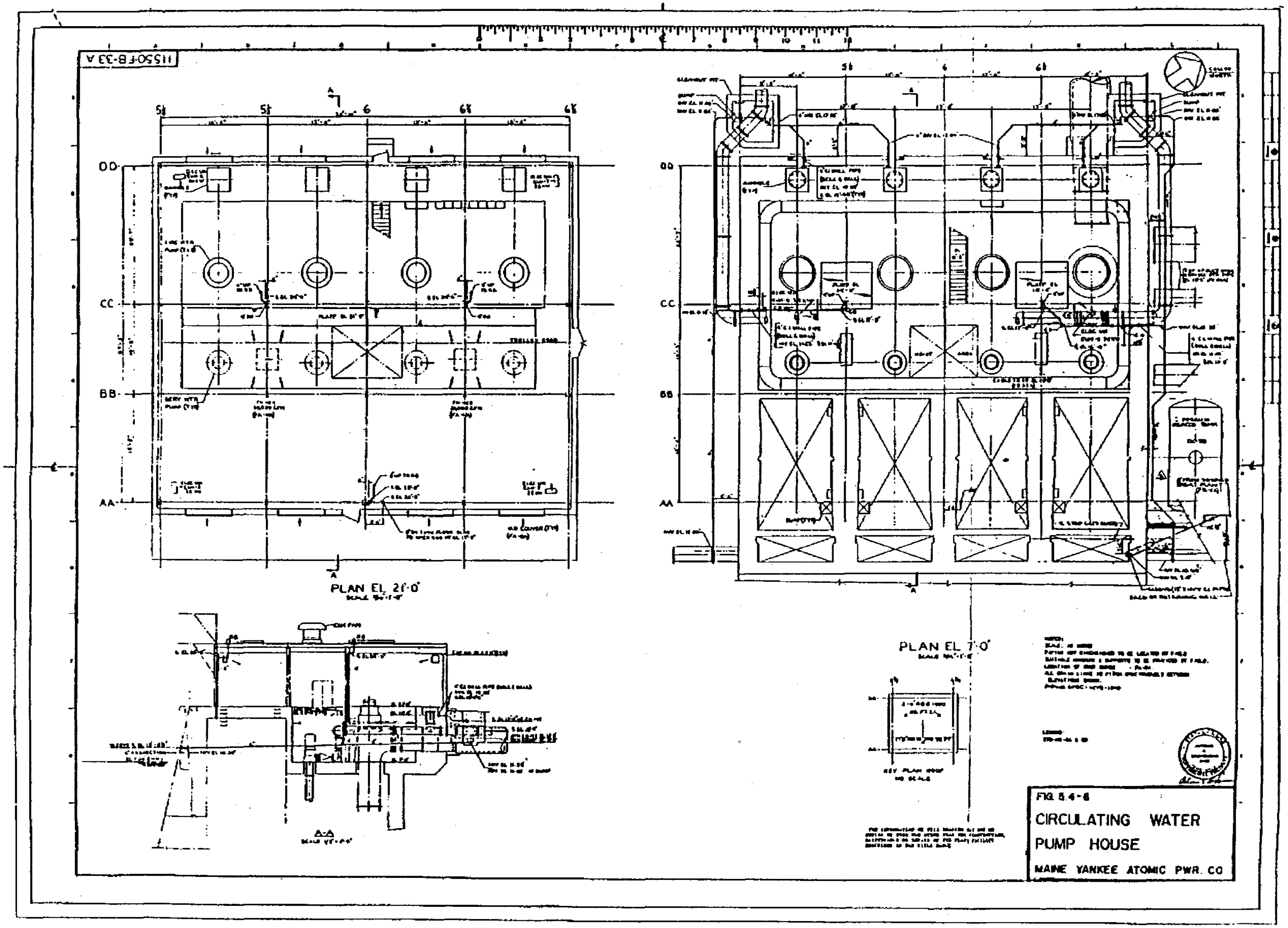
FIG. 5.4-4B

SERVICE BLDG. ADDITION

MAINE YANKEE ATOMIC-PWR. CO.

REV. 2

A-11


11550-FB-33A
PLAN EL 21'-0"
PLAN EL 7'-0"
A-A
KEY PLAN
FIG 5.4-6
CIRCULATING WATER
PUMP HOUSE
MAINE YANKEE ATOMIC PWR. CO

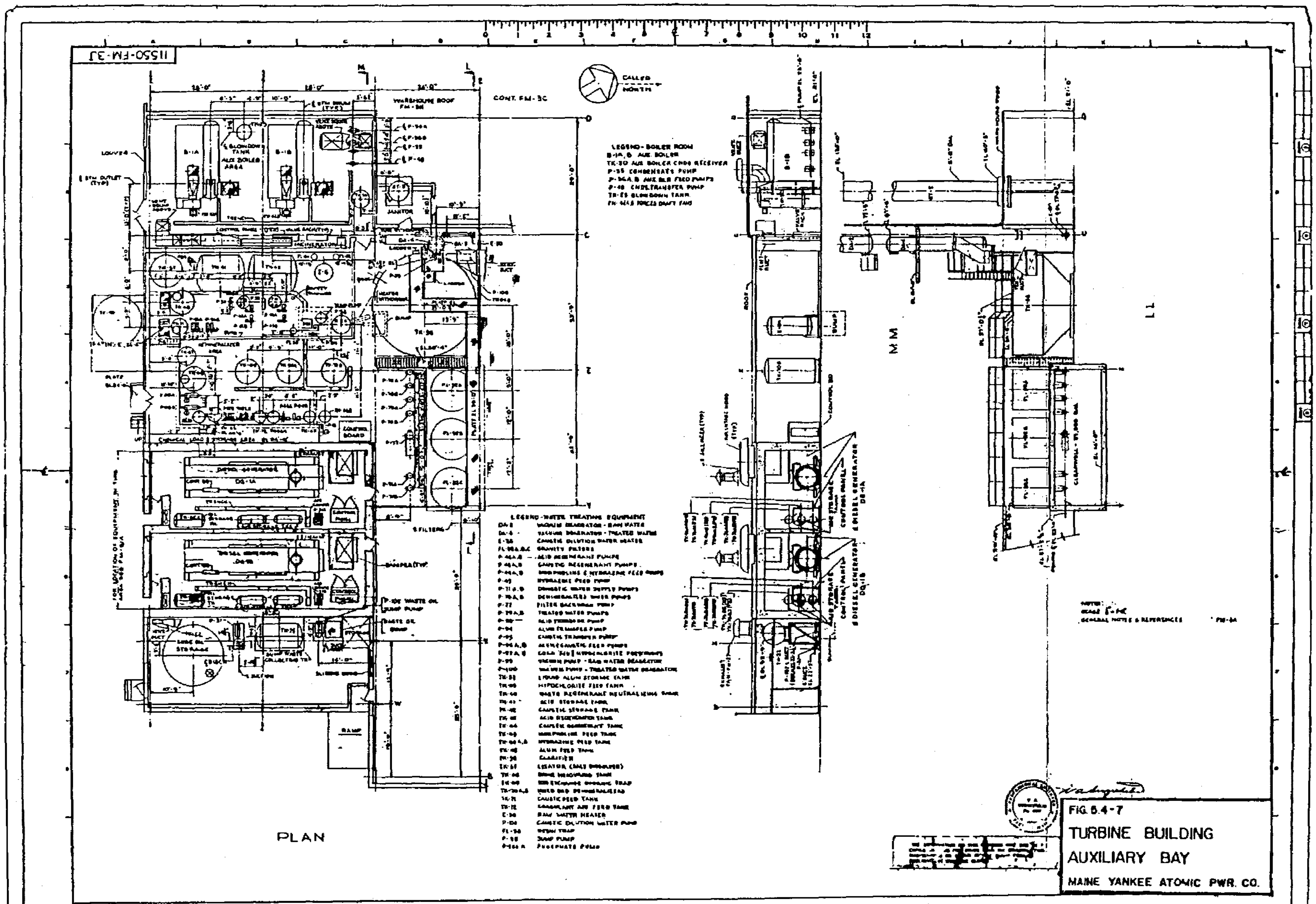


FIG. 5.4-7
TURBINE BUILDING
AUXILIARY BAY
MAINE YANKEE ATOMIC PWR. CO.

NRC FORM 335
(2-84)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE.

1. REPORT NUMBER *(Assigned by TIDC, add Vol. No., if any)*
NUREG/CR-4826
UCID-20948
Vol. 3

2. TITLE AND SUBTITLE
Seismic Margin Review of the Maine Yankee Atomic Power Station
Volume 3. Fragility Analysis

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
February	1987

5. AUTHOR(S)
M. K. Ravindra, G. S. Hardy, P. S. Hashimoto, and J. J. Griffin

6. DATE REPORT ISSUED

MONTH	YEAR
March	1987

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
EQE, Inc.
Newport Beach, CA 92660

Under Contract to: Lawrence Livermore National Laboratory
Livermore, CA 94550

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER
A0461

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Division of Engineering Safety
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a. TYPE OF REPORT
Technical

b. PERIOD COVERED *(Inclusive dates)*
June 1986-March 1987

12. SUPPLEMENTARY NOTES

13. ABSTRACT *(200 words or less)*

This Fragility Analysis is the third of three volumes for the Seismic Margin Review of the Maine Yankee Atomic Power Station. Volume 1 is the Summary Report of the first trial seismic margin review. Volume 2, Systems Analysis, document the results of the systems screening for the review. The three volumes demonstrate how the seismic margins review guidance (NUREG/CR-4482) of the NRC Seismic Design Margins Program can be applied.

The overall objectives of the trial review are to assess the seismic margins of a particular pressurized water reactor, and to test the adequacy of this review approach, quantification techniques, and guidelines for performing the review. Results from the trial review will be used to revise the seismic margin methodology and guidelines so that the NRC and industry can readily apply them to assess the inherent quantitative seismic capacity of nuclear power plants.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS
Seismic Margins
Seismic Analysis

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page)
Unclassified
(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

☆U.S. G.P.O. 1987-181-682:60027