Individual Plant Examination of External Events

for Severe Accident Vulnerabilities

Perry Nuclear Power Plant

June 1996

Cleveland Electric Illuminating Company

Abstract

An individual plant examination of external events (IPEEE) for severe accident vulnerabilities was performed for the Perry Nuclear Power Plant in response to USNRC Generic Letter 88-20, Supplement 4 dated June 28, 1991.

The IPEEE examined potential vulnerabilities in three broad areas for external events; internal fires, earthquakes and other external hazards such as high winds and tornadoes, external floods, and potential hazards from transportation and nearby facilities.

The methodology used for internal fires was the Electric Power Research Institute (EPRI) Fire Induced Vulnerability Evaluation (FIVE). The FIVE methodology provides a combination of deterministic and probabilistic techniques for examining the internal fire probability and characteristics. FIVE includes a two phase progressive screening with a third phase containing a walkdown/verification process. The discussion of the Sandia Fire Risk Scoping Study Issues were also included.

Large earthquakes were evaluated using the EPRI Seismic Margins methodology. This methodology uses safe shutdown success paths to identify the set of plant components to be evaluated. The components are screened for seismic ruggedness. Those that satisfy the screening criteria require no further evaluation. The remaining components are deterministically evaluated for the review level earthquake defined in NUREG-1407. The seismic capacity for which there is a high confidence of a low probability of failure (HCLPF) was generated for these components. The review level earthquake for Perry is 0.3g.

Other external hazards were evaluated against the 1975 Standard Review Plan (NUREG-75/087). A progressive screening approach was used involving both deterministic and probabilistic techniques.

Acknowledgments

The Perry Individual Plant Examination of External Events was performed by a team of personnel from Cleveland Electric Illuminating Company, Halliburton NUS, and Gilbert Commonwealth. The project has benefited from advice and review from Vectra Technologies, Dr. Paul Smith and Mr. Harry Johnson.

The assistance and review by other site organizations was a key factor in ensuring the accuracy of the final report.

Project Team

CEI

M.D. Makar
S.P. Meyer
S.C. Tomaszewski

G/C

C. Chen
H. Ho
D. Keiser
R. Schmehl
R. Svotelis

NUS

A. Afzali
J. Juliano
P. Guymer
D.J. McGuigan
G.W. Parry
G.P. Rozga
R.F Sacramo
F. Zikria

The Readiness Operation

Dr. P. Smith

Programmatic Solutions

H. Johnson

Vectra Technologies

C. Abou-Jaoude
J. Amason
G. Ashley
D. de la Cruz
S. Thompson

1 EXECUTIVE SUMMARY 1-1

1.1 Background and Objectives 1-1

1.2 Plant Familiarization 1-2

1.3 Overall Methodology 1-2

1.4 Summary of Major Findings 1-3
1.4.1 Seismic Analysis 1-3
1.4.2 Internal Fire Analysis 1-3
1.4.3 High Winds, Floods, and Others 1-3

1.5 References 1-4

2 EXAMINATION DESCRIPTION 2-1

2.1 Introduction 2-1

2.2 Conformance with Generic Letter and Supporting Material 2-1

2.3 General Methodology 2-1
2.3.1 EPRI Seismic Margin Assessment 2-1
2.3.2 Fire-Induced Vulnerability Evaluation 2-2
2.3.3 High Winds, External Flooding, and Other External Hazards Evaluation 2-2

2.4 Information Assembly 2-3
2.4.1 Plant Layout 2-3
2.4.2 Sources of Information 2-3

2.5 References 2-4

3 SEISMIC ANALYSIS 3-1

3.1 Seismic Margins Methodology -- EPRI Seismic Margin Assessment (Focused-Scope). 3-2
3.1.1 Review of Plant Information, Screening, and Walkdown 3-3
3.1.2 System Analysis 3-10
3.1.3 Analysis of Structure Response 3-17
3.1.4 Evaluation of Seismic Capacities of Components and Plant 3-18
3.1.5 Analysis of Containment Performance 3-25
3.1.6 Fire/Seismic Interaction 3-26

3.2 USI A-45, GI-131 and Other Seismic Safety Issues 3-27
3.2.1 USI A-45 "Shutdown Decay Heat Removal Requirements" 3-27
3.2.2 The Eastern U.S. Seismicity Issue (The Charleston Earthquake Issue) 3-27
3.2.3 USI A-17 "System Interactions in Nuclear Power Plants" 3-28
3.2.4 January 31, 1986 Leroy Earthquake Issues 3-28

3.3 References 3-29

4 INTERNAL FIRES ANALYSIS 4-1

4.1 Methodology Selection 4-1

4.2 Fire Hazard Analysis 4-1
4.2.1 Overview 4-1
4.2.2 Assumptions and Other Modeling Considerations for Screening 4-6
4.2.3 Analysis Results 4-9

4.3 Review of Plant Information and Walkdown 4-12
4.3.1 Plant Information Sources 4-12
4.3.2 Outstanding Modifications 4-13
4.3.3 Plant Walkdowns 4-13

4.4 Fire Growth and Propagation Modeling 4-14
4.4.1 Fire Scenario Identification and Evaluation 4-15
4.4.2 Fire Modeling 4-16
4.4.3 Characterization of Ignition Sources 4-17

4.5 Evaluation of Component Fragilities and Failure Modes 4-21
4.5.1 Cables Serving Safe Shutdown Equipment 4-21
4.5.2 Electrical Equipment 4-21
4.5.3 Intervening Combustibles 4-22
4.5.4 Characterization of Secondary Fire Sources 4-22

4.6 Fire Detection and Suppression 4-22
4.6.1 Automatic Suppression 4-22
4.6.2 Manual Suppression 4-23

4.7 Analysis of Plant System Sequences and Plant Response 4-24
4.7.1 Detailed Analysis for Unit 1, Division 2 Switchgear Room 4-24
4.7.2 Control Room Analysis 4-33
4.7.3 Results 4-43

4.8 Containment Performance 4-45
4.8.1 Containment Bypass 4-45
4.8.2 Containment Isolation 4-46
4.8.3 Containment Pressure Suppression and Heat Removal 4-46
4.8.4 Containment Performance Summary 4-46

4.9 Treatment of Fire Risk Scoping Study Issues 4-47
4.9.1 Background 4-47
4.9.2 Seismic/Fire Interactions 4-48
4.9.3 Fire Barrier Qualifications 4-50
4.9.4 Manual Fire Fighting Effectiveness 4-52
4.9.5 Total Environment Equipment Survival 4-57
4.9.6 Control Systems Interactions 4-58
4.9.7 Conclusions 4-58

4.10 USI A-45 and Other Safety Issues 4-58

4.11 References 4-59

5 HIGH WINDS, FLOODS, AND OTHERS 5-1

5.1 Generic Plant Description 5-1
5.1.1 Site Description 5-1
5.1.2 Identification of Structures, Systems and Components Potentially Susceptible to External Events 5-2

5.2 Screening or External hazards 5-2
5.2.1 Description of Approach 5-2
5.2.2 Results of Screening Analysis 5-4

5.3 High Winds 5-4
5.3.1 Straight Winds 5-4
5.3.2 Tornadoes 5-6
5.3.3 Walkdown 5-6

5.4 External Floods 5-7
5.4.1 General Plant Description 5-7
5.4.2 Sources of External Flooding 5-7
5.4.3 Flooding Caused by Probable Maximum Precipitation (PMP) 5-8
5.4.4 Flooding Caused by Fiberglass Pipe Rupture 5-11
5.4.5 Flooding Caused by Miscellaneous Events 5-13
5.4.6 Conclusion and Results 5-14

5.5 Transportation and Nearby Facility Accidents 5-15
5.5.1 Aircraft Impact 5-15
5.5.2 Transportation Accidents 5-22
5.5.3 Hazardous Materials 5-23
5.5.4 Fixed Facility Accidents 5-33

5.6 Conclusions 5-40

5.7 References 5-41

6 LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM 6-1

6.1 IPEEE Program Organization 6-1

6.2 Composition of Independent Review Team 6-1

6.3 Areas of Review and Major Concerns 6-2
6.3.1 Seismic Margins Assessment 6-2
6.3.2 Internal Fires 6-2
6.3.3 High Winds, External Floods, and Other External Hazards 6-3

6.4 Resolution of Comments 6-3

7 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES 7-1

7.1 Unique Safety Features 7-1
7.1.1 Seismic Analysis 7-1

7.1.2 Internal Fire Analysis 7-1
7.1.3 High Winds, Floods, and Others 7-1

7.2 Plant Improvements 7-2
7.2.1 Seismic Analysis 7-2
7.2.2 Internal Fire Analysis 7-3
7.2.3 High Winds, Floods, and Others 7-3

8 SUMMARY & CONCLUSIONS (including proposed resolution of USIs, GIs and other issues)8-1

8.1 Seismic Analyses 8-1

8.2 Internal Fire Analyses 8-1

8.3 High Winds, Floods, and Other External Hazards 8-1

8.4 Resolution of USIs, GIs and Other Issues 8-2
8.4.1 USI A-17 "System Interactions in Nuclear Power Plants" 8-2
8.4.2 USI A-45 "Shutdown Decay Heat Removal Requirements" 8-2
8.4.3 The Eastern U.S. Seismicity Issue (The Charleston Earthquake Issue) 8-2
8.4.4 January 31, 1986 Leroy Earthquake Issues 8-2

List of Tables

Table 2-1 - Summary of Design Features - Perry 2-5
Table 3-1 - Success Path A Systems 3-30
Table 3-2 - Success Path B Systems 3-31
Table 3-3 - Pre-Screening Basis for Structures 3-32
Table 3-4 - Pre-Screening Basis for Equipment 3-33
Table 3-5 - PNPP Seismic Input Scaling Factors 3-34
Table 3-6 - PNPP Dominant Natural Frequencies of Structures 3-35
Table 3-7 - PNPP Percent Critical Damping Values Used for the SMA 3-36
Table 3-8 - PNPP Calculated HCLPF Values 3-37
Table 4-1 - PNPP Fire Areas and Associated Compartments 4-60
Table 4-2 - Fire Induced System Loss and Initiating Event Matrix 4-62
Table 4-3 - Fire Area/Compartment Screening Results through Phase II Step 2 4-74
Table 4-4 - F2 Calculations for Fire Compartment 1CC3a Fire Scenarios 4-77
Table 4-5 - Grouping of Control Room Cabinets 4-80
Table 4-6 - Cabinet Equipment Status and Fire Frequency Evaluation 4-81
Table 4-7 - Screening of Internal Cabinet Fires 4-86
Table 4-8 - Control Room Fire Damage States 4-88
Table 4-9 - Results of Control Room Analysis 4-89
Table 4-10 - Summary of Overall Fire Analysis Results for Areas which Did Not Initially Screen4-91
Table 5-1 - Screening of External Events for PNPP 5-43

List of Figures

Figure 3-1 - Success Path Logic Diagram (Success Path A) 3-38
Figure 3-2 - Success Path Logic Diagram (Success Path B) 3-39
Figure 4-1 - Layout of Unit 1, Division 2 Switchgear Room 4-92
Figure 4-2 - Layout of Control Room 4-93
Figure 4-3 - Example of Control Room Cabinet Fire Growth Event Tree 4-94
Figure 5-1- Airways and Airports in the Vicinity of PNPP 5-45
Figure 5-2 - Nearby Industrial Facilities and Transportation Routes 5-46
Figure 5-3 - Gas Pipelines in the Vicinity of PNPP 5-47
Figure 6-1 - IPEEE Project Organization Chart 6-4
Figure 6-2 - Peer Review Organization 6-5

Table of Acronyms

ADS	Automatic Depressurization System
AFSS	Automatic Fire Suppression Systems
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CCDP	Conditional Core Damage Probability
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin Approach
CEI	Cleveland Electric Illuminating Company
CFR	Code of Federal Regulations
CIEL	Containment Integrity Equipment List
CRD	Control Rod Drive
CST	Condensate Storage Tank
DHR	Decay Heat Removal
DOT	Department of Transportation
ECCS	Emergency Closed Cooling System
EOC	Emergency Operations Center
EOF	Emergency Operations Facility
EPRI	Electric Power Research Institute
ERG	Emergency Response Guidebook
ESW	Emergency Service Water System

FAA	Federal Aviation Administration
FCIA	Fire Compartment Interaction Analysis
FDS	Fire Damage Scenario
FEDB	Fire Events Data Base
FIVE	Fire-Induced Vulnerability Evaluation
F.M.	Factory Mutual
FPCC	Fuel Pool Cooling and Clean-Up
FRP	Fiberglass Reinforced Plastic
FRS	Floor Response Spectra
FRSS	Fire Risk Scoping Study
G/C	Gilbert/Commonwealth, Inc.
GDC	General Design Criteria
GE	General Electric Corporation
GERS	Generic Equipment Ruggedness Spectra
GI	Generic Issue
GIP	Generic Implementation Procedure
GRS	Ground Response Spectrum
HCLPF	High Confidence of a Low Probability of Failure
HCU	Hydraulic Control Unit
HGL	Hot Gas Layer
HPCS	High Pressure Core Spray System
HRR	Heat Release Rate
HVAC	Heating, Ventilation, Air Conditioning
IOI	Integrated Operating Instruction

IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
LEPC	Lake County Emergency Planning Committee
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray System
MCC	Motor Control Center
MG	Motor Generator
MMI	Modified Mercalli Intensity
MZIRWL	Minimum Zero Injection RPV Water Level
NCC	Nuclear Closed Cooling
NOAA	National Oceanographic and Atmospheric Administration
NOL	Normal Operating Loads
NSAC	Nuclear Safety Analysis Center
NSRC	Norfolk Southern Railway Company
NTSB	National Transportation Safety Board
NUS	Halliburton NUS Corporation
NWS	National Weather Service
OBE	Operating Basis Earthquake
ONI	Off-Normal Instruction
PAD	Protective Action Distance

PAP	Plant Administrative Procedure
PCS	Power Conversion System
PGA	Peak Ground Acceleration
PMF	Probable Maximum Flooding
PMM	Perchloro Methyl Mercaptan (trichloromethane sulfenyl chloride)
PMP	Probable maximum Precipitation
PNPP	Perry Nuclear Power Plant
PRA	Probabilistic Risk Assessment
PTI	Periodic Test Instruction
RCIC	Reactor Core Isolation Cooling System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RLE	Review Level Earthquake
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRA	Radiologically Restricted Area
RRCS	Redundant Reactivity Control System
RRS	Required Response Spectrum
RSP	Remote Shutdown Panel
SBLOCA	Small Break LOCA
SCBA	Self Contained Breathing Apparatus
SEWS	Screening Evaluation Work Sheet
SI	System Interaction

SMA	Seismic Margin Assessment
SMM	Seismic Margins Methodology
SPLD	Shutdown Path Logic Diagram
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SRT	Seismic Review Team
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSRAP	Senior Seismic Review and Advisory Panel
STEL	Short Term Exposure Limit
TAF	Top of Active Fuel
TRS	Test Response Spectrum
TSEE	Temperature Sensitive Electrical Equipment
TWA	Time Weighted Average
UHS	Uniform Hazard Spectra
U.L.	Underwriters Laboratories
USAR	Updated Safety Analysis Report
USGS	U.S. Geological Survey
USI	Unresolved Safety Issue
USNRC	U.S. Nuclear Regulatory Commission
ZPA	Zero Period Acceleration

1 EXECUTIVE SUMMARY

This document presents the results of the Individual Plant Examination of External Events (IPEEE) performed in response to USNRC Generic Letter 88-20, Supplement 4[1-1] which requested each licensee to perform a plant examination of each of their nuclear power plants to search for vulnerabilities to severe accidents arising from external events and to propose cost effective safety improvements that reduce or eliminate the important vulnerabilities. The content and format of this report is in accordance with the USNRC guidance document NUREG-1407[1-2] and provides sufficient information to show how the results were obtained.

This summary includes a discussion on the background and objectives, the plant familiarization activities, the overall methodologies, and a summary of the study findings.

1.1 Background and Objectives

The Commission issued a policy statement in 1985 to the effect that, based on available information, existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants. However, it was decided that a systematic evaluation of each plant would be beneficial in that it would provide information on any plant-specific vulnerabilities to accidents and, through their resolution, enhance safety. Generic Letter 88-20 was issued and an Individual Plant Examination (IPE) using probabilistic risk assessment methodology was performed for Perry, submitted to the USNRC on July 15, 1992 and approved by the USNRC in a letter dated August 18, 1994. The IPE addressed internal events and internal flooding. Supplement 4 to the generic letter was issued to extend the analysis to external events. In addition to identifying each vulnerability, it was decided that similar to the IPE, the IPEEE should be performed by a team in which utility personnel played a major part as recommended in the generic letter. The general purpose of the IPEEE is also similar to the internal event IPE. Thus the IPEEE would ensure that utility personnel also achieved the following:

- Develop an appreciation of severe accident behavior.
- Understand the most likely severe accident sequences that could occur at the plant under full power operating conditions.
- Gain a qualitative understanding of the overall likelihood of core damage and radioactive material releases.
- If necessary, reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

It should be emphasized that the involvement of CEI personnel in all aspects of the analysis and the development of appropriate support documentation has ensured that the IPEEE addresses potential plant vulnerabilities to severe accidents as accurately as possible and, therefore, enables the above objectives to be met.

1.2 Plant Familiarization

A separate project team was established for each of the three portions of the IPEEE with overall coordination provided by the CEI Project Lead. Each team consisted of a number of engineers from CEI and one or more of several consulting engineering firms.

The seismic team utilized three CEI individuals with support from Gilbert/Commonwealth, Inc., NUS Corporation, The Readiness Operation and Programmatic Solutions. The internal fire analysis team was led by a CEI fire protection engineer with assistance from Vectra Technologies, Inc. and NUS Corporation. The high winds, floods, and other external hazards analyses were performed by a team consisting of CEI and NUS Corporation engineers.

The IPEEE was performed both at the site and in the offices of the consultants. Walkdowns by both CEI and consultant personnel were made as needed to obtain plant specific information. Drawings were available to all individuals involved. All work products were reviewed by the CEI personnel directly involved in the tasks to verify that the information used was accurate.

The Perry Nuclear Power Plant (PNPP) consists of a single operating unit located in Lake County Ohio approximately 7 miles northeast of Painesville and 35 miles northeast of Cleveland. The plant site is located along the southeastern shoreline of Lake Erie on an ancient lake plain approximately 50 feet above the lake low water elevation. The site and its environs consist mainly of woodland and former nursery lands. The total area of the site is approximately 1,100 acres and is relatively flat. The land has a gentle slope toward the lake and is crossed by small streams which drain into the lake. The main plant building are located about 800 feet from the toe of a 45 foot high steep bluff that forms the shoreline.

PNPP is a 3,579 MW_{th} General Electric BWR/6 with a Mark III containment. The balance of plant systems were engineered by Gilbert Commonwealth, Inc. The unit started commercial operation in November 1987.

1.3 Overall Methodology

The IPEEE for the Perry Nuclear Power Plant was split into three separate sets of analyses:

- Seismic Events
- Internal Fires
- High Winds, Floods, and Other External Hazards

A seismic margins analysis (SMA) using the EPRI seismic margins methodology (SMM) outlined in the EPRI document NP-6041-SL[1-3] for a focused scope plant was performed to meet the requirements in Generic Letter 88-20, Supplement 4. Two safe shutdown paths were selected and evaluated to demonstrate that they are capable of performing their safe shutdown functions following a 0.3g review level earthquake.

The internal fires evaluation was performed using the EPRI Fire-Induced Vulnerability Evaluation (FIVE) as described in EPRI document TR-100370.[1-4] FIVE is a progressive screening approach based on quantifying the frequency of fire ignition in specific plant areas, the availability of automatic suppression systems, the probability of having sufficient combustibles and heat release to cause damage to shutdown systems and the probability of manual suppression effectiveness.

The approach used for high winds, floods and other external hazards follows the method described in NUREG/CR-4839[1-5] by performing an initial screening analysis, followed by bounding or detailed analyses as necessary.

1.4 Summary of Major Findings

1.4.1 Seismic Analysis

The seismic margin assessment results showed that PNPP is capable of attaining shutdown conditions and maintaining them for 72 hours following a review level earthquake (RLE) of 0.3g. During the walkdown, several pieces of equipment were found which did not meet the caveats in the Generic Implementation Procedure (GIP)[1-8] developed by the Seismic Qualification Utility Group (SQUG) and, therefore, could not be immediately determined to have a HCLPF of greater than or equal to 0.3g. Those components not meeting the caveats are discussed further in Section 3.1.3. Analyses subsequent to the walkdown determined that the minimum HCLPF met or exceeded the RLE.

1.4.2 Internal Fire Analysis

Based on the internal fires evaluation performed for the IPEEE, PNPP has no significant fire hazards of concern. Using the EPRI FIVE methodology the majority of the fire zones were screened out using either qualitative screening or determination that their respective core damage frequencies were less than 10^{-6}/yr. More detailed analyses were performed on the remaining fire zones. As a result of the more detailed analyses, most of these fire zones were also shown to have core damage frequencies less than 10^{-6}/yr.

1.4.3 High Winds, Floods, and Others

The review of the impact on PNPP of high winds, floods and other external hazards resulted in the conclusion that there are no significant events of concern. A comprehensive screening analysis of those other external hazards identified in the PRA Procedures Guide[1-6] resulted in supporting the NUREG-1407 conclusion that only high winds, external floods and transportation and nearby facility accidents need be reviewed in detail. The plant as designed meets the intent of the criteria of the Standard Review Plan of 1975 and thus it is not surprising that these events also do not pose a significant threat to the plant.

1.5 References

1-1 USNRC Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

1-2 NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Final Report, June 1991.

1-3 EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Revision 1, August 1991.

1-4 EPRI TR-100370, "Fire-Induced Vulnerability Evaluation (FIVE)," April 1992.

1-5 NUREG/CR-4839, Ravindra, M.K. and Bannon, H., "Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development," July 1992.

1-6 NUREG/CR-2300, "PRA Procedure Guide," December 1983.

1-7 NUREG-0800, "Standard Review Plan," USNRC, Revision 2, July 1981.

1-8 "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Seismic Qualification Utility Group, February 1992.

2 EXAMINATION DESCRIPTION

2.1 Introduction

The Perry IPEEE project is aimed at meeting two objectives: satisfying the USNRC requirements for an individual plant examination of external events and providing confidence that external factors do not constitute undue risk to the plant. In this section, it is demonstrated that the project conforms with the USNRC requirements. The general methodology and information used in the course of the study are also discussed.

2.2 Conformance with Generic Letter and Supporting Material

Generic Letter 88-20, Supplement 4, identifies a number of requirements in the areas of the examination process, methodology, treatment of outliers and reporting. The IPEEE conforms with the requirements laid out in the generic letter and uses the current supporting material appropriate to the analysis of Perry. However, neither the reactor pressure vessel internals nor soil-related failures were evaluated as part of the seismic portion of the IPEEE. This deviation from the guidance of NUREG-1407 is based on Generic Letter 88-20, Supplement 5.[2-9]

It is the belief of the USNRC as stated in the generic letter that the quality and comprehensiveness of the results derived from the IPEEE depend on the rigor with which the utility applies the method of examination and on their commitment to the intent of the IPEEE. It can be seen from the project organization shown in Figure 2.1 that the CEI personnel were involved in the performance of all tasks. In addition to CEI personnel, assistance from a number of consultants was obtained to produce a comprehensive product.

In-house review of the IPEEE analyses was performed by Perry staff conversant with systems design, structural integrity and phenomenology of the items being examined. In addition to the in-house review, a further review of the examination process and results was performed by an independent consultant.

The combination of experienced consultants, Perry personnel and an independent review in the performance of the analysis has ensured that the quality and comprehensiveness of the Perry IPEEE results meet the standards imposed by the generic letter.

2.3 General Methodology

2.3.1 EPRI Seismic Margin Assessment

The Perry Nuclear Power Plant (PNPP) Seismic Margin Assessment (SMA) was achieved using the guidelines contained in Generic Letter 88-20, Supplement 4[2-1] and NUREG-1407.[2-2] NUREG-1407, Table 3.1 binned PNPP into those plants which could satisfy the requirements of the generic letter by performing a "focused scope" SMA for a review level earthquake (RLE) of 0.3g.

The actual evaluation was performed using the Electric Power Research Institute (EPRI) seismic margins methodology for a focused scope plant described in EPRI NP-6041-SL[2-3] as recommended in

NUREG-1407. The basis of this approach is selection of a minimum set of equipment, designated the safe shutdown equipment list (SSEL), that is required to safely perform a reactor shutdown following a RLE. An assessment is made of the items contained on the SSEL, and the structures containing those items, as to their overall capability to survive an RLE event. Those items capable of withstanding the RLE event can be screened from further consideration. Any items that cannot be screened are identified and must be addressed to determine their final outcome.

2.3.2 Fire-Induced Vulnerability Evaluation

The Fire-Induced Vulnerability Evaluation (FIVE) methodology developed by EPRI is a screening technique based on conservative assumptions using industrial and plant specific data bases for evaluating fire sequences. The overall objective is to determine the availability of plant equipment necessary to achieve and maintain safe and stable shutdown of a nuclear power plant. FIVE consists of three phases.

- Phase I: Fire Area Screen (qualitative analysis)
- Phase II: Critical Fire Compartment Screen (quantitative analysis)
- Phase III: Plant Walkdown/Verification and Documentation

In addition, several potentially risk significant items that were identified in the Fire Risk Scoping Study, NUREG/CR-5088,[2-8] are also included in the evaluation.

2.3.3 High Winds, External Flooding, and Other External Hazards Evaluation

The approach used follows the method described in NUREG/CR-4839[2-4] by performing an initial screening analysis followed by bounding or detailed analyses as necessary.

The objective of the screening analysis is to provide confirmation of the NUREG-1407 conclusion that there are no hazards unique to the site that require evaluation, other than those posed by high winds, external floods, and transportation and nearby facility accidents.

The PRA Procedures Guide[2-5] provides an exhaustive list of potential hazards which is the starting point for the analysis. The screening is performed by reviewing information on the site region and plant design to identify external events that are applicable using the screening criteria described in Section 5 of this report. The data in the safety analysis report on the geologic, seismologic, hydrologic, and meteorological characteristics of the site region as well as present and projected industrial activities in the vicinity of the plant are reviewed for this purpose.

The evaluation of high wind loading for PNPP was performed by comparing the PNPP design as described in the Updated Safety Analysis Report (USAR), Section 3.3.1[2-6] with the USNRC Standard Review Plan,[2-7] Section 3.3.1, "Wind Loading."

External floods were evaluated by examining the four potential flooding sources at Perry; Lake Erie, intense local precipitation, and two small streams which border the site. In addition, the impact of

flooding due to Service Water system or Circulating Water system fiberglass pipe rupture was also investigated.

Accidents involving hazardous chemicals, both toxic and having explosive potential, were evaluated for shipping traffic on Lake Erie, road traffic, rail traffic and fixed nearby facilities and gas pipelines. The risk from aircraft was also evaluated.

2.4 Information Assembly

The following paragraphs discuss some of the sources of information which were used in the IPEEE analyses. The combination of plant design documents, operating instructions, walkdowns, input from outside organizations, and the collective experience of the IPEEE team form the basis of the IPEEE.

2.4.1 Plant Layout

The Perry Nuclear Power Plant (PNPP) began as a two unit site. Only Unit 1 has been completed and is operating. Unit 2 was approximately 40 percent complete when construction activities stopped. Unit 2 has since been formally abandoned. The site is therefore a single unit site as far as the IPEEE analyses are concerned.

The unit is a General Electric BWR/6 with a Mark III containment. The balance of plant systems were engineered by Gilbert Commonwealth. The unit started commercial operation in November 1987.

2.4.2 Sources of Information

The safe shutdown paths developed for the seismic portion of the IPEEE were based on the Perry PRA. Detailed information was taken from plant drawings and operating instructions. Seismic walkdown preparation depended on plant drawings and spot checks in the field. The seismic walkdowns provided the information used to verify the seismic ruggedness of the safe shut equipment.

Information used in the FIVE methodology came from fire protection design documents. The Perry PRA was also used to determine conditional core damage frequencies for screening out some fire areas. Detailed fire analyses of the remaining fire areas depended on information gathered during fire walkdowns.

Outside organizations, site drawings and local maps provided much of the documentation used for the other external hazards. National government agencies were contacted and provided information on transportation related items such as air traffic and Lake Erie shipping. Local government agencies provided information on the storage of hazardous chemicals by nearby industries. Information on the transport of hazardous materials by rail traffic were obtained from the operators of the rail roads in the vicinity of the plant. When needed, specific details on hazardous materials were obtained directly from the nearby facilities storing and using those materials. Site walkdowns were also made to support the high winds and external flooding analyses.

2.5 References

2-1 USNRC Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

2-2 NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Final Report, June 1991.

2-3 EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Revision 1, August 1991.

2-4 Ravindra, M.K. and Bannon, H., "Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development," NUREG/CR-4839, July 1992.

2-5 NUREG/CR-2300, "PRA Procedures Guide," January 1983.

2-6 Perry Nuclear Power Plant Updated Safety Analysis Report.

2-7 NUREG-0800, "Standard Review Plan," USNRC, Revision 2, July 1981.

2-8 NUREG/CR-5088, "Fire Risk Scoping Study."

2-9 USNRC Generic Letter 88-20, Supplement 5, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," September 1995.

Table 2-1 - Summary of Design Features - Perry

3,579 MW_{th} GE Boiling Water Reactor with a 238" diameter Reactor Pressure Vessel (RPV)

High Pressure RPV Injection Systems

High Pressure Core Spray (HPCS) - motor driven pump backed by Div 3 diesel generator

Reactor Core Isolation Cooling (RCIC) - turbine driven pump

Condensate/Feedwater - combination of turbine and motor driven pumps; not diesel backed

Low Pressure RPV Injection Systems

Residual Heat Removal (RHR) in Low Pressure Coolant Injection (LPCI) mode of operation - three independent trains; RHR A motor driven pump backed by Div 1 diesel generator and RHR B and C motor driven pumps backed by Div 2 diesel generator

Low Pressure Core Spray (LPCS) - motor driven pump backed by Div 1 diesel generator

Condensate Transfer Alternate Injection - non-diesel backed motor driven pumps

Fast Firewater Alternate Injection - diesel driven fire pump

RPV Depressurization Systems

Automatic Depressurization System (ADS) - 8 relief valves supplied with safety related instrument air

Manual RPV Depressurization using either the 8 ADS valves and/or the 11 non-ADS relief valves supplied with non-safety related instrument air

Decay Heat Removal

Power Conversion System (PCS) - Decay heat removed via the main steam lines and the condenser

Residual Heat Removal (RHR) in either Containment Spray or Suppression Pool Cooling modes of operation - RHR A and B have heat exchangers for removing decay heat

Containment Venting - Containment may be vented using the RHR A or B with the containment spray headers or via the Fuel Pool Cooling and Clean-Up (FPCC) system

Table 2-1 - Summary of Design Features - Perry (Continued)

Electrical Design

Offsite power

2 Standby Diesel Generators supplying power to Division 1 and Division 2, respectively

1 Diesel Generator supplying power to Division 3 for HPCS

3 d.c. buses supplied by batteries and backed by Division 1, 2 and 3 a.c. power and inverters

Containment Structure

Mark III freestanding steel containment with pressure suppression

3 SEISMIC ANALYSIS

Methodology Selection

The Perry Nuclear Power Plant (PNPP) Seismic Margin Assessment (SMA) was achieved using the guidelines contained in Generic Letter 88-20, Supplement 4[3-1] and NUREG-1407.[3-2] The actual evaluation was performed using the Electric Power Research Institute (EPRI) seismic margins methodology recommended in NUREG-1407, that being the EPRI NP-6041-SL[3-3] for a focused scope plant with a review level earthquake (RLE) of 0.3g as binned by NUREG-1407, Table 3.1. The basis of this approach is selection of a minimum set of equipment, designated the safe shutdown equipment list (SSEL), that is required to safely perform a reactor shutdown following a RLE. An assessment is made of the items contained on the SSEL, and the structures containing those items, as to their overall capability to survive an RLE event. Those items capable of withstanding the RLE event can be screened from further consideration. Any items that cannot be screened are identified and must be addressed to determine their final outcome.

For PNPP, the NUREG/CR-0098[3-4] median-centered rock spectrum anchored at 0.3g peak ground acceleration is specified as the horizontal ground RLE for rock-supported buildings. The vertical ground RLE is equal to 2/3 times the horizontal ground RLE. Likewise, for soil-supported structures (Diesel Generator Building), the NUREG/CR-0098 median soil spectra anchored at 0.3g was used. These spectra were specified at grade level in the free field. The PNPP design Safe Shutdown Earthquake (SSE) is based on Regulatory Guide 1.60[3-5] response spectra anchored at 0.15g peak ground acceleration. The 0.3g RLE with respect to the 0.15g SSE provides a sufficient challenge to the PNPP seismic design basis as intended by the SMA evaluation. It is not the intent of this program to determine the absolute largest earthquake the site could withstand, only to demonstrate with a high degree of confidence that the site could withstand an RLE. This challenge will help in identifying any plant vulnerabilities and any weak links that might limit the plant shutdown capacity.

Safe shutdown success paths were developed to identify the systems that must function to successfully shutdown and cool the reactor following the occurrence of a RLE event. A safe shutdown success path is a combination of systems that is used to accomplish all of the required safe shutdown functions. A success path can be depicted in a Shutdown Path Logic Diagram (SPLD). The SPLD for Perry identified 39 systems and 933 components that required evaluation for seismic adequacy.

3.1 Seismic Margins Methodology -- EPRI Seismic Margin Assessment (Focused-Scope)

As discussed previously, NUREG-1407, for focused-scope plants, and the EPRI Seismic Margin Methodology (SMM) was used to perform the seismic analysis.

- A Seismic Review Team (SRT) was assembled and trained which allowed them to perform the required analyses and walkdowns.
- The selection of the safe shutdown paths and safe shutdown equipment list, SSEL, was assembled to determine the minimum set of equipment required to safely shutdown the reactor.
- A review of known low seismic ruggedness relays was conducted.
- Walkdown information was assembled to enable thorough walkdown evaluation.
- The seismic capability walkdowns were performed.
- The SSEL, and associated structures housing those components, were evaluated for their seismic capacity.
- Documentation of the evaluation was completed.

The seismic margins analysis guidelines help focus on the identified success paths rather than a re-analysis of the entire plant. The success paths can achieve and maintain a safe shutdown condition for a minimum of 72 hours following an RLE event. Structures, systems, and components in the shutdown paths are reviewed and evaluated. The safe shutdown paths and components list were compiled along EPRI developed guidelines and formed the basis for the plant walkdowns. Neither reactor pressure vessel internals, nor soil-related failures, were evaluated in this evaluation. This deviation from the guidance of NUREG-1407 is based on Generic Letter 88-20, Supplement 5.[3-9]

The safe shutdown path walkdown methodology used at PNPP has been adapted from industry work regarding Unresolved Safety Issue (USI) A-46.[3-6] Technical research by the Seismic Qualification Utility Group (SQUG) and the USNRC resulted in an approach for A-46 resolution called the Generic Implementation Procedure (GIP).[3-7]

Since PNPP was constructed after the implementation of the IEEE 344-1975[3-8] standard for qualification of Class I electrical equipment, an A-46 review was not required to be performed. However, the GIP provides the detailed technical approach, generic procedures, and documentation guidance which can be used to verify the seismic adequacy of mechanical and electrical equipment used in the safe shutdown paths of the plant.

The issue of spurious activation of relays, known as relay chatter, was addressed using the guidance of NUREG-1407. The extent of the relay evaluation consisted of a review of the relays in the safe shutdown paths and comparing those found to a compilation of known low-seismic ruggedness relays.

All aspects of the SMA were performed under the direction and control of utility personnel. Consequently, the PNPP seismic margins evaluation was performed as a joint effort between Cleveland Electric Illuminating (CEI), Gilbert/Commonwealth (G/C) and NUS. The consultants listed were used for their expertise in the seismic field and provided valuable guidance in key areas.

3.1.1 Review of Plant Information, Screening, and Walkdown

This section provides general plant information, the seismic input, and walkdown method used in the evaluation.

3.1.1.1 General Plant Information

PNPP is located in Lake County, Ohio approximately 7 miles northeast of Painesville and 35 miles northeast of Cleveland, the nearest principal city. The eastern two thirds of the site is within the boundaries of North Perry Village while the western third is within Perry Township.

The plant site is located along the southeastern shoreline of Lake Erie on an ancient lake plain approximately 50 feet above lake low water elevation. The site and its environs consist mainly of woodland and former nursery lands. The total area of the plant site is approximately 1,100 acres and relatively flat. The land has a gentle slope toward the lake and is crossed by small streams which drain onto the lake. The main plant buildings are located about 800 feet from the toe of a 45 foot high steep bluff that forms the shoreline. Upper Devonian shale bedrock underlies the site about 55 feet below the existing ground surface. Bedrock onshore is overlain by approximately 30 feet of very dense till which in turn is overlain by about 25 feet of poorly compacted lacustrine deposits. Thin layers of glacial till and beach deposits overlie bedrock at the shore. Pleistocene glaciation induced localized shallow faults and folds in the shale beneath the site.

Subsurface exploration, substantiated by laboratory testing of soil and rock, along with excavation experience, confirmed that stratigrahically, the subsurface materials and their respective physical properties were similar throughout the plant site. The Upper Devonian shale strata beneath the site dips less than 5° southeast, but the erosional bedrock surface slopes north toward Lake Erie. Groundwater levels ranged between three and five feet below ground surface. The depth to groundwater gradually increased toward Lake Erie.

The bearing characteristics of the lacustrine deposits and upper till with a combined thickness of 35 feet are generally unsuitable for the support of Seismic Category I structures. Support for most Seismic Category I buildings is provided by lower till and Chagrin shale. Seismic Category I structures, such as piping, duct banks, buried fuel oil storage tanks, and the diesel generator building is founded on compacted Class A backfill, an imported well-graded sand and gravel fill containing less than 5 percent fines (that is, fraction passing the No. 200 sieve). The lower till exhibits a very low compressibility under static loads up to 6 tons/ft^2 (tsf). The shale is capable of supporting loads to at least 25 tsf without detrimental settlement. Subsurface investigation of the cooling water tunnel alignments indicated that Chagrin shale beneath Lake Erie is relatively uniform and generally competent and free from detrimental soft zones.

Shallow deformation exposed by foundation excavation into the Chagrin shale, although unanticipated, was similar in style and origin to that identified during preconstruction site-location geologic

investigation. A similar result was obtained during investigation of surrounding features following the January 31, 1986 Leroy earthquake. In this instance the shallow structures are confined to the Cleveland shale directly overlying the Chagrin shale.

An abnormal, small-displacement, thrust fault intersecting the cooling water tunnels, was revealed during tunneling. Studies show that its last movement, an adjustment to glacio-isostatic rebound, occurred in Pleistocene time. None of the site faulting evaluated during initial site investigations, or faults mapped during post Leroy earthquake studies, are capable as defined in Appendix A to 10 CFR 100.

Ohio and adjacent areas are characterized by small infrequent earthquakes with an occasional moderate earthquake. Three moderate earthquakes, one of Modified Mercalli Intensity VII-VIII (MMI) centered in the Anna, Ohio area, 185 miles southwest of the site, one of Intensity VIII (MMI) in the Attica, New York area, 160 miles northeast of the site, and the Intensity V-VI (MMI) event centered near Leroy, Ohio, 10 miles south of the site, represent the largest earthquakes to have occurred within 200 miles of the site.

The principal buildings and structures housing safety related equipment include the Containment Structure, the Auxiliary Building, the Intermediate Building, the Control Complex, the Diesel Generator Building, and the Emergency Service Water Pumphouse.

These buildings and structures are founded upon suitable material for their intended application. Structures essential to the safe operation and shutdown of the plant are designed to withstand more extreme loading conditions than normally considered in non nuclear design applications. The buildings and internal structures so designated are designed to provide protection as required from tornadoes, earthquakes, and failure of equipment producing flooding, missiles, jet impingement loads, and pipe whip.

The Containment Structure is a Seismic Category I structure which encloses the reactor vessel, the drywell, suppression pool, upper fuel pool and some of the engineered safety features and support systems. The functional design basis of the Containment, including its penetrations and isolation valves, is to contain with adequate design margin the energy released from a design basis loss-of-coolant accident and to provide a leaktight barrier against the uncontrolled release of radioactive material to the environment. This structure is founded on bedrock.

The Auxiliary and Intermediate Buildings are Seismic Category I multistoried reinforced concrete structure that contain safety systems, fuel storage, and necessary auxiliary support systems. Redundant safety trains in the Auxiliary Building and all other areas of the plant are separated and protected so that a loss of function of one train will not prevent the other train from performing its safety function. These structures are founded on bedrock or utilize caissons to the bedrock.

The Control Complex is a Seismic Category I multistoried, steel and reinforced concrete structure in which many of the control and electrical systems, including required support systems directly related to safety or necessary for plant operations, are located. This structure is founded on bedrock.

The Diesel Generator Building is a Seismic Category I reinforced concrete structure. The building contains the three diesel generators, three fuel oil day tanks, building ventilation, filters, silencers, and

controls. Each diesel generator and its associated equipment are housed in a separate room within the building. This structure is founded on well compacted Class A fill material.

The Emergency Service Water Pumphouse is a Seismic Category I reinforced concrete structure and contains the emergency service water pumps and associated equipment. Additionally, it houses the diesel powered fire pump. This building is founded on bedrock.

3.1.1.2 Seismic Input to Structures and Equipment

3.1.1.2.1 The Safe Shutdown and Operating Basis Earthquakes

On the basis of the seismology of the site area, the (SSE) is designated as an Intensity VII (MMI) earthquake occurring close to the site. The resulting maximum horizontal ground acceleration at foundation level within the competent bedrock at the site is estimated to be in the range of 0.07g to 0.13g. In order provide an additional margin of conservatism, a value of 0.15g is assigned as the maximum horizontal ground acceleration. This value is deemed adequately conservative under Appendix A of 10 CFR 100. Safety related structures, systems and components are designed to ensure safe plant shutdown for two horizontal and one vertical excitations simultaneously.

The Operating Basis Earthquake (OBE) is designated as one with half the acceleration of the SSE and equivalent to an Intensity VI (MMI) earthquake near the site. The corresponding horizontal acceleration at foundation level would be less than 0.075g.

The design response spectra used for Seismic Category I structures, systems and components were developed in accordance with Regulatory Guide 1.60. The horizontal and vertical design spectra were normalized to 0.15g for SSE and 0.075g for OBE.

3.1.1.2.2 The Review Level Earthquake

Since PNPP is designated as a focused-scope plant, the RLE was set at 0.3g, as required by NUREG-1407. For the purposes of conducting the seismic capability walkdowns and evaluations, the existing in-structure SSE floor response spectra were conservatively scaled up to the level of the RLE.

3.1.1.3 Seismic Walkdowns

This section describes the approach to the seismic capability walkdown, the screening criteria used and the final details of the walkdown.

3.1.1.3.1 Approach

A significant amount of plant information was reviewed and used in the SMA. This includes the PNPP USAR, PRA, and numerous other documents such as drawings, procedures, seismic analyses, and seismic qualification test reports. The seismic capability analyses of components and structures, including walkdown notes are documented in PNPP Screening Evaluation Work Sheets (SEWS). The SEWS are similar to those developed by SQUG.

The key element in the EPRI SMA is the performance of a plant walkdown for both the A and B success paths. The approach used to perform the systems and component selection walkdown, and the seismic capability walkdown follows the recommendations of EPRI NP-6041-SL. This includes the following activities:

- Selection of the assessment team
- Pre-walkdown preparation
- System and component selection for walkdown
- Seismic capability walkdown

The seismic assessment team, all part of the Seismic Review Team (SRT), was comprised of six individuals. The four team members responsible for the actual seismic evaluation all met the qualification requirements of EPRI NP-6041-SL. The qualifications included the training and certification to the SQUG walkdown methodology. Four SRT members are G/C personnel, while the remaining two are CEI personnel at the PNPP site. One systems engineer, who developed the A and B safe shutdown paths, provided systems and operations input to the two walkdown teams. Additionally, one walkdown coordinator assisted the teams in developing the daily schedule of items to be evaluated. The following is a listing of the SRT members, their affiliation and area of expertise.

Mr. Rich Schmehl	Gilbert/Commonwealth	Seismic Engineer
Mr. Ry Svotelis	Gilbert/Commonwealth	Seismic Engineer
Mr. Hoat Ho	Gilbert/Commonwealth	Seismic Engineer
Mr. Stan Tomaszewski	Cleveland Electric Illuminating	Seismic Engineer
Mr. Steve Meyer	Cleveland Electric Illuminating	Shutdown Path Systems Engineer
Mr. Don Keiser	Gilbert/Commonwealth	Walkdown Coordinator

The walkdown team members were technically supported by Dr. Chang Chen, the G/C Task Manager, Dr. Paul Smith of The Readiness Operation (TRO) and Mr. Harry Johnson of Programmatic Solutions. Either Dr. Paul Smith or Mr. Harry Johnson were available on site at all times to act as an observer supporting both walkdown teams. Additionally, Dr. Paul Smith participated as a walkdown team member part time.

Prior to the walkdown, preparatory work was performed that consisted of gathering and reviewing information about the plant design and operation including the USAR, specifications, drawings, qualification calculations, and existing internal events PRA. A detailed walkdown procedure was prepared, including the detailed technical approach of each task and the interfaces between the team members and PNPP support personnel. The front-line and support systems for the A and B success paths and the equipment and components in these systems that must function in order for the system to accomplish its safe shutdown function were identified by the system engineer. A system and element selection walkdown was then conducted by a seismic engineer and the system engineer to review a majority of components for any obvious seismic problems and to locate and arrange access for equipment for the subsequent seismic walkdown. Two independent teams were used during the seismic capability walkdowns.

Specific documentation assembled and evaluated prior to and during the walkdowns included:

- PNPP Safe Shutdown Paths and Equipment Lists.
- Plant arrangement drawings.
- The sections of the PNPP USAR relating to the seismic design and licensing basis of the plant.
- The ground response spectra for the SSE and RLE.
- Floor response spectra and subsequent scaling factors for the RLE.
- A sample of construction details of equipment anchorage including drawings and specifications.
- A sample of procurement and seismic qualification testing specification for equipment.
- Examples of calculations for seismic and anchorage qualifications.
- Design basis documents for the PNPP structures.

The data obtained was not intended to be the sole basis for the screening of components, but assisted the SRT in their review.

The structures at PNPP were screened generically. The drawings and analysis models were reviewed for details that might indicate seismic vulnerabilities in accordance with the requirements of a focused-scope SMA. The drawing and structural analysis reviews confirmed that consistent good practice in design detail and analysis was utilized at PNPP. Therefore, a small sample of the details of connections, reinforcement bar placement, construction joints, etc. was justified in making the judgments on screening.

In addition to the 39 systems and 933 components, the following structures housing the equipment were included in the success paths:

- Primary Containment Structure
- Shield Building
- Auxiliary Building
- Fuel Handling/Intermediate Building
- Control Complex
- Diesel Generator Building
- Emergency Service Water Pumphouse

The SRT then conducted the seismic capability walkdown and reviewed the equipment and components from both the primary and alternate success paths for seismic adequacy (both structural integrity and anchorage) and system interaction (SI). The SRT review consisted of detailed walkdown of the representative equipment, and walkby of similar equipment.

3.1.1.3.2 Screening Criteria

Initially, the SRT pre-screened a number of structures, components, and equipment using the screening criteria guidelines contained in EPRI NP-6041-SL. Tables 3.3 and 3.4 provide a general listing of the pre-screened structures and equipment, along with the basis for the pre-screening. The basis for these tables are the screening guidelines given in Tables 2-3 and 2-4, respectively, of EPRI NP-6041-SL for the spectral accelerations less than 0.8g.

During the walkdown, the basis for pre-screening was verified for the success path structures and equipment selected for review. The issues and considerations discussed in Appendix A of EPRI NP-6041-SL and the judgment of the SRT were used as the basis for verifying that the screened out elements are seismically adequate.

Issues which influence the screening discussed previously are: redundancy provided by multi-train systems, similarity in design and location of redundant trains, treatment of single failures, access to components during walkdowns, and system interactions potential including fire and internal flood sources.

The screening approach, described in additional depth below, is appropriate for modern plants of PNPP's vintage. The document review and the system and element selection walkdown verified that uniform practices in accordance with the plant design basis for construction, design and installation were followed.

During the PNPP walkdown, it was confirmed, as expected, that most items in a given equipment class were either identical or very similar. The plant documentation review and walkdown confirmed that the vast majority of equipment was manufactured, and installed as specified. The screening procedures

employed at PNPP for generic categories of equipment and structures contained caveats or inclusion rules from the GIP that were verified during the walkdown. Since the equipment at PNPP was purchased and installed to similar codes and standards the SRT screened generic classes of equipment on the basis of their relative ruggedness and said caveats. The screening sampling size for identical or very similar equipment, in a given equipment class, was one or greater and for similar equipment in a given class with identical or very similar anchorage was two or greater for each walkdown team. The increased sample for anchorage is based on experience that anchorage installations are not always consistent. This approach is consistent with the guidance provided in Appendix D of EPRI NP-6041-SL. Those specific components which were inspected based on the above approach were considered to be "walked-down." A 100 percent "walk-by" of all remaining equipment on the SSEL was employed to check for unique equipment details and for seismic interactions.

The walkdown and inspection of distribution systems that were installed in bulk, such as piping, cable trays, HVAC ductwork, electrical conduit, and instrumentation lines were performed on an area basis. To accomplish this, the areas where these systems are located were generically inspected rather than selecting specific trains or runs of these distribution systems. All distribution system elements in the area were walked by. In addition to area walk by of distribution systems, detailed walk downs of a portion of each of the types of distribution systems were performed. Using this approach, distribution system components included in the success paths, as well as elements not included, in the area were evaluated. Upon completion of the walkdown, the distribution systems were reviewed so as to verify that all the inclusion rules were met. It was confirmed that the design and installation practices at PNPP are consistent, therefore, the screening judgment was valid based upon a review of general specifications and drawings for a single run of each generic class of distribution system. As expected the review of the general specifications and drawings did not indicate significant differences in design and installation practice. Therefore, based on the above discussion, this was determined to be an effective approach and all systems were found to be ruggedly supported.

3.1.1.3.3 Walkdown Preparation

To ensure an efficient use of time and to ensure that the scope of the walkdowns focused on the seismic assessment, preparatory work prior to the walkdown was considered vital. This work consisted of reviewing plant seismic design documentation and qualification test reports. The information reviewed included the following items:

- Selected sections, including seismic design basis sections, of the PNPP USAR.
- Plant general arrangement drawings, civil/structural drawings, specifications, and representative anchorage details for equipment in the SSEL.
- Sample seismic equipment qualification reports and example equipment anchorage packages.
- In-structure response spectra for elevations where success path equipment and components are located.
- Topical and vendor drawings, reports, and calculations.
- Preliminary list of the success path equipment selected by the system engineer.

Anchorage details and drawings indicated that the equipment anchorage at PNPP is relatively rugged. The constructed anchorage details in the plant was verified during the walkdown and were found to meet or exceed the standard design details.

3.1.1.3.4 Walkdown Process

The SRT performed the seismic capability walkdowns throughout a two week period during Refuel Outage 4 (RFO-4). The walkdown consisted of reviewing the success path equipment for structural integrity, anchorage adequacy, and system interaction. Attention was directed to anchorage and system interaction since these would be the major modes of failure for most equipment.

The walkdowns were organized so that easily assessable areas were reviewed first. These areas included the Emergency Service Water Pumphouse, Diesel Generator Building and Control Complex. Once these areas were completed, the contaminated areas, those requiring protective clothing, were walked down. This included Containment, Drywell, Main Steam Tunnel, Auxiliary Building, and the Fuel Handling/Intermediate Building. During the walkdowns, the SRT requested and reviewed additional information, such as calculations, drawings, and vendor qualification test reports, to address concerns raised during the inspections or to confirm assumptions. Additionally, cabinets were opened when the SRT had specific interest, such as anchorage details and internal device mountings, that allowed for a visual inspection of the internals and anchorage.

3.1.2 System Analysis

3.1.2.1 Principal Safety Functions

The basic requirements for prevention of core damage and prevention or mitigation of the release of fission products can be divided into the following four functions.

- Reactivity Control
- Reactor Coolant System Overpressure Protection
- Reactor Pressure Vessel (RPV) Level Control
- Containment Overpressure Protection and Fission Product Control (includes decay heat removal)

The Perry PRA was used as the basis for defining these functions and identifying the systems needed for accomplishing them.

Successful reactivity control is defined as the rapid insertion of negative reactivity into the core such that the reactor is brought subcritical. This is analyzed as the rapid insertion of a sufficient number of control rods into the core to provide a shutdown rod pattern.

Reactivity control is required to rapidly make the core subcritical to reduce the power generation to the fission product decay heat level. If a seismic margin earthquake were to occur when the plant is operating,

it is likely that the turbine would first trip due to either load rejection or excessive vibrations in the turbine generator unit. The turbine trip would send reactor scram signals to the reactor protection system. Reactivity control is initiated by the reactor protection system and accomplished by the rapid insertion of control rods into the core, commonly referred to as a reactor scram. Reactivity control can also be accomplished by injecting a highly concentrated boron solution into the reactor coolant system with high pressure pumps. This emergency boration requires operator action to initiate as well as several other anticipated transient without scram (ATWS) mitigation operator actions. Rapid emergency boration is not considered a viable method of rapid reactivity control during or after a seismic margin event because of the additional stress imposed on the operators from the RLE and the time it takes for shutdown to occur.

Successful reactor coolant system overpressure protection is defined as the successful operation of plant systems such that the integrity of the reactor coolant system is not endangered. The RPV is designed to ASME code which permits a maximum pressure transient of 110 percent of design pressure. RPV design pressure is 1,250 psig. 110 percent of design pressure is 1,375 psig. The assumed pressure limit for the PRA analysis is, therefore, 1,375 psig.

Overpressure protection can be provided by the power conversion system (the main steam lines, condenser and circulating water) or by cycling multiple safety relief valves (SRVs). Following an earthquake and loss of offsite power the power conversion system is assumed to be unavailable. This leaves the SRVs as the only mechanism for controlling RPV overpressure. No support systems to the SRVs are necessary for RPV overpressure protection. The SRVs are designed to open against springs (Bellville washers) if needed.

Successful RPV level control is defined as the successful operation of plant systems such that RPV water level is maintained above the Minimum Zero Injection RPV Water Level (MZIRWL) or that a high capacity injection system is recovered at the point of reaching the MZIRWL. In such a condition, the heat transfer from the fuel rods to the water is sufficient at the relatively low heat flux associated with decay heat to result in acceptable fuel rod temperatures. Maintaining adequate water inventory in the RPV satisfies the RPV level control function, but the decay heat must still be transferred from the RPV water to a heat sink.

Successful containment overpressure protection is defined as maintenance of containment pressure below the Containment Pressure Limit of 40 psig. The results of the containment strength analysis performed for the Individual Plant Examination (IPE) showed that maintaining pressures below 50 psig would prevent containment failure. Therefore, 50 psig has been used to define successful containment overpressure protection.

For the purposes of this analysis, decay heat is removed from the RPV by steam flow through the SRVs and into the suppression pool. Eventually the suppression pool heats up and causes the containment pressure to rise. The heat is ultimately removed from the containment by the RHR system operating in suppression pool cooling mode or by venting the containment to the atmosphere. Both of these mechanisms were included as part of this analysis.

3.1.2.2 Safe Shutdown Success Paths

One of the principal objectives of the SMA was to develop an understanding of how the systems and plant operators would respond to a review level earthquake.

The RLE is assumed to cause a loss of offsite power (LOOP) and, potentially, a small loss of coolant accident (LOCA). Under these conditions, the EPRI methodology stipulates the plant be evaluated for performing the safety functions necessary to achieve and maintain hot or cold shutdown for 72 hours following the occurrence of a seismic event.

The redundancy and diversity of a nuclear power plant design provides several success paths that can perform the safety functions. For the purposes of the SMA, two separate success paths were chosen which would be available following a loss of offsite power and which could achieve and maintain hot or cold shutdown given an RLE in conjunction with a seismically induced small LOCA.

Success path logic diagrams (SPLD), Figure 3.1 and Figure 3.2, were constructed to identify which systems are needed to accomplish the safety functions. The success criteria from the Perry PRA was also used as the success criteria for this analysis. Once the success paths were established with the success path logic diagrams, the front-line systems and their support systems were determined using the Perry PRA. The systems included in Success Path A are listed in Table 3.1 by safety function. The systems for Success Path B are listed in Table 3.2.

The success paths for the Perry SMA are both redundant and diverse. Although the Reactivity Control and Reactor Coolant System Overpressure Protection safety functions use the same set of systems and components for each success path, they are divisionally separated, and highly redundant, i.e., there are nineteen SRVs to prevent RCS overpressure.

Success Path A can provide water to the RPV under high reactor pressure conditions. The Containment Overpressure Protection safety function is assumed to be via containment venting through the fuel pool cooling and clean-up lines. If the Division 2 diesel generator is unavailable, the Division 3 diesel generator, by procedure, can be cross-tied to the Division 2 bus to provide motive power for valve operation. Other than electrical buses, no other Division 1 or 2 components are used for RPV Level Control or Containment Overpressure Protection.

Success Path B is the low pressure success path. It is both diverse and redundant to Success Path A. For Success Path B, the Automatic Depressurization System is used to depressurize the RPV and allow the low pressure ECCS systems to inject. The Containment Overpressure Protection safety function is accomplished either through containment venting or using the Suppression Pool Cooling Mode of RHR.

3.1.2.3 System Descriptions

<u>Safety Relief Valves and Automatic Depressurization System, B21</u>

The nineteen safety relief valves (SRVs) provide a relief path for steam in the event of a reactor overpressure condition. Each SRV is a spring loaded valve with an externally attached pneumatic operating cylinder. Each SRV can operate in either the safety mode or the relief mode. The safety mode of all nineteen valves supports the RPV overpressure function. The system is single failure proof and common to the RPV overpressure protection function for both success paths.

Eight of the nineteen valves are designated as part of the Automatic Depressurization System (ADS). These function in the relief mode as needed to depressurize the RPV in support of the RPV Level Control function for Success Path B.

Control Rod Drive System, C11

For the IPEEE the control rod drive (CRD) system consists of the hydraulic control units (HCU), with the accumulators, associated piping, the control rod drives and controls associated with scramming the reactor. The system is single failure proof and common to the reactivity control function for both success paths.

Redundant Reactivity Control System, C22

The redundant reactivity control system (RRCS) is designed to mitigate the potential consequences of an ATWS. It provides control signals to the control rod drive system, the feedwater water control system and the reactor recirculation control system. The RRCS attempts to shut down the reactor or reduce reactor power by increasing the amount of negative reactivity in the core.

RRCS consists of two divisions each with two channels. Either division can initiate protective actions. Power is supplied from Class 1E 125 VDC buses to the alternate rod insertion (ARI) valves and RRCS logic. Because only the control rods were credited with shutting down the reactor for the IPEEE, only that portion of the RRCS associated with ARI was included in the SMA. The RRCS is common to the reactivity function for both success paths.

Reactor Protection System, C71

The reactor protection system (RPS) is designed to provide protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and reactor coolant pressure boundary. The RPS consists of two independent, functionally identical trip systems. Each trip system is divided into two independent, functionally identical trip channels. The RPS logic is arranged such that at least one of the two channels in each trip system must de-energize to cause a scram. Because de-energization causes a scram, the power supplies for RPS were not included as part of the IPEEE. The RPS is common to the reactivity function for both success paths.

Residual Heat Removal System, E12

The residual heat removal (RHR) system is a three train system that automatically restores and maintains the RPV water level following an accident. Two of the trains also remove decay heat from the reactor coolant system, the suppression pool or the containment as required. RHR is a low pressure system that operates in conjunction with ADS. Trains A and B each have two heat exchangers in series downstream of the pumps. Train C has no heat exchangers. Each of the three trains takes suction from the suppression pool. Train A receives a.c. and d.c. power from Class 1E Division 1 electrical buses. Trains B and C receive a.c. and d.c. power from Class 1E Division 2 electrical buses.

Each of the trains support the emergency core cooling safety function of Success Path B via the low pressure coolant injection (LPCI) mode of operation of RHR. Cooling water to the heat exchangers in Trains A and B is not required for LPCI. Trains A and B also support the containment overpressure

protection safety function for Success Path B through the suppression pool cooling and containment spray modes of operation of RHR. Cooling water flow through the heat exchangers, supplied by the emergency service water system, is required for these modes of operation.

In addition to LPCI, suppression pool cooling and containment spray modes of operation, the lines for RHR A and B may also be used to vent the containment in support of Success Path B. The flow path is from the containment spray headers, through the containment spray lines, cross-tied to the lines to the spent fuel pool inside the fuel handling building and out in the atmosphere.

Low Pressure Core Spray System, E21

The low pressure core spray (LPCS) system automatically provides coolant to the RPV during accidents when the reactor pressure is low. ADS can be used in conjunction with LPCS to attain a low enough reactor coolant system pressure to permit injection. LPCS consists of a single train with motor-operated valves and a motor driven pump. Suction is taken from the suppression pool. LPCS receives a.c. and d.c. power from the Class 1E Division 1 electrical buses. LPCS is one of several injection systems/trains in Success Path B for the RPV level control function.

High Pressure Core Spray System, E22

The high pressure core spray (HPCS) system automatically provides coolant to the reactor pressure vessel during accidents when the reactor pressure remains high. HPCS consists of a single train with motor-operated valves and a motor driven pump. Suction is taken from two possible water sources, the condensate storage tank (CST) or the suppression pool. HPCS is normally aligned with the CST. When a low level in the CST or a high level in the suppression pool is sensed, HPCS suction is automatically aligned to the suppression pool. As a non-seismic tank, the CST was not included for the SMA. HPCS receives a.c. and d.c. power from the Class 1E Division 3 electrical buses. HPCS fulfills the RPV level control function for Success Path A.

Fuel Pool Cooling and Clean-Up System, G41

The fuel pool cooling and clean-up (FPCC) system normally removes decay heat generated by spent fuel stored in the fuel storage pools and maintains the purity, clarity and water level in the Upper Pools in the containment. This system may also be used to vent the containment if the containment pressure cannot be maintained below the Containment Pressure Limit provided in the Plant Emergency Instructions.

The FPCC containment venting flowpath is through the eleven skimmers in the Fuel Transfer and Storage Pool, the Reactor Well and the Separator Storage Pool in the containment, to the FPCC surge tank and into the fuel handling building atmosphere through the five skimmers in the Spent Fuel Storage Pool.

Containment venting via FPCC is the primary mechanism for meeting the containment overpressure protection safety function for Success Path A. All valves with the exception of the inboard containment isolation valve can be opened locally by operators using handwheels. The inboard containment isolation valve is normally powered from a Division 2 Class 1E bus. In the event that Division 2 power

is unavailable, the Division 3 Class 1E bus can be cross-tied to the Division 2 bus for valve manipulation. Division 3 provides the electric power for Success Path A.

ECCS Pump Room Cooling System, M39

The ECCS pump room cooling system dissipates heat generated by the ECCS pump motors, the RCIC turbine, and associated piping in the ECCS pump rooms and the RCIC pump room. Cooling is accomplished by circulating room air through cooling coils. Each pump room is provided with its own cooling unit which automatically starts when its associated train commences operation.

Each of the six independent cooling units take suction from room air in which they are located, passes the air through a roughing filter and a cooling coil before discharging the cooled air in the vicinity of the pump. Class 1E a.c. power is divisionally supplied.

The room cooler for the HPCS pump room supports Success Path A. The balance of the room coolers, with the exception of the RCIC room cooler, support Success Path B. RCIC is not included in either Success Path A or B.

Diesel Generator Building Ventilation System, M43

The diesel generator (D/G) building ventilation system functions to provide ventilating air to the D/G rooms. This air dissipates heat generated by the D/Gs and miscellaneous equipment during operation. Each of the D/G rooms is served by independent trains consisting of two 100 percent capacity fans, four 50 percent capacity motor-operated louvers, outside air and recirculation dampers and ductwork.

Trains A and B provide support for the Division 1 and 2 D/Gs used in Success Path B. Train C supports the Division 3 D/G used in Success Path A.

Emergency Closed Cooling System, P42

The emergency closed cooling (ECC) system is a closed loop system that provides cooling to several of the systems used in Success Path B. ECC is divided into two independent trains, each consisting of a pump, motor-operated valves and a heat exchanger. Train A receives a.c. and d.c. power from the Class 1E Division 1 buses. Cooling for the train A heat exchanger comes from train A of emergency service water (ESW). Train B of ECC is associated with Division 2 Class 1E power and train B of ESW. ECC does not provide support for Success Path A.

Emergency Service Water System, P45

Emergency service water (ESW) supplies cooling water to equipment required for both normal and emergency shutdown of the reactor. The source of water for ESW is Lake Erie. ESW is made up of three independent trains each consisting of a pump and motor-operated valves. Trains A, B and C receive a.c. power from the Class 1E buses for Divisions 1, 2 and 3 respectively.

ESW trains A and B support equipment and the removal of decay heat for Success Path B. ESW train C supports equipment used in Success Path A.

Safety Related Instrument Air, P57

The safety related instrument air system supplies clean, dry, oil free air to the ADS safety relief valves. The system is divided into two independent trains. Each train consists of air filters, air tanks and air headers for distribution. During normal plant operation, the system air pressure is maintained by a non-safety non-diesel backed air compressor. Following an event the air receiver tanks provide the air needed for ADS actuation. Trains A and B support the RPV Level Control Function for Success Path B in conjunction with ADS. No support is provided to Success Path A from safety related instrument air.

Class 1E D.C. Power

The Class 1E d.c. power system is divisionally separated into Divisions 1, 2 and 3. Each of the independent divisions contains batteries (the Unit 1 and Unit 2 batteries of each division may be cross-tied to extend battery life during an event, e.g., Unit 1/Division 1 and Unit 2/Division 1 may be cross-tied by procedure), battery chargers, and associated buses. During normal operation, the battery chargers provide the power to maintain the d.c. bus voltages. Upon loss of the a.c. power source to the chargers, the batteries maintain bus voltage until a.c. power is restored.

For the IPEEE, the duration of the event was assumed to be 72 hours, far longer than the capacity of the batteries. Restoration of a.c. power via diesel generators was assumed to be needed for long-term d.c. power availability. Divisions 1 and 2 support Success Path B and Division 3 supports Success Path A.

Class 1E A.C. Power

The Class 1E a.c. power system consists of three 4,160 VAC buses, their associated switchgear, 480 VAC buses, 120 VAC buses and motor control centers (MCCs). The system is divisionally separated into Divisions 1, 2 and 3. Normally, off-site power supplies the bus loads. Given an undervoltage signal, diesel generators start and automatically supply power to their buses. The diesel generators also start on a LOCA signal but do not supply the buses unless an undervoltage signal exists.

Divisions 1 and 2 support Success Path B. Division 3 is dedicated to HPCS and its support systems which form Success Path A.

Diesel Generators and Their Support Systems

As noted above, the three diesel generators start and supply power to the 4,160 VAC Division 1, 2 and 3 buses given an undervoltage condition on their associated bus. In addition to D/G building ventilation, ESW and d.c. power, the Division 1 and 2 diesel generators depend on five support systems for their operation: 1) standby D/G starting air, 2) standby D/G fuel oil, 3) standby D/G jacket water cooling, 4) standby D/G lube oil and 5) standby D/G exhaust/intake crankcase.

Although not explicitly modeled in the PRA, the components needed to support long-term diesel generator operation from these support systems were included in the Safe Shutdown Equipment List (SSEL) for the IPEEE seismic margins assessment. The Division 3 D/G has these same support systems incorporated into its design.

As with the other divisionally separated systems, Divisions 1 and 2 support Success Path B and Division 3 supports Success Path A.

Remote and Local Panels, H13, H22, & H51

Although remote and local control and instrumentation panels and racks do not strictly constitute a separate system, they do have their own equipment I.D. numbers. As such, those panels and instrument racks that support the systems that form Success Paths A and B were also included as part of a "super" system.

3.1.2.4 Safe Shutdown Equipment List

The Safe Shutdown Equipment List (SSEL) was developed based on the guidance provided in the Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment[3-7] developed by the Seismic Qualification Utility Group (SQUG) for unresolved safety issue USI A-46, Seismic Qualification of Equipment in Operating Plants.[3-10] Diversity, reliability and independence were considered during the development of the SSEL. No reliance was made on alignments of systems that are not included in plant instructions. The equipment identified for seismic evaluation included:

- Active mechanical and electrical equipment which operates or changes state to accomplish a safe shutdown function.
- Active equipment in systems which support the operation of identified safe shutdown equipment.
- Instrumentation needed to confirm that the four safe shutdown functions have been achieved and are being maintained.
- Instrumentation needed to operate the safe shutdown equipment.
- Tanks and heat exchangers used by or in the identified safe shutdown paths.

Individual cable trays, conduit, piping, instrument lines and HVAC ducts for the various distribution systems were not explicitly listed in the SSEL. However, these items were listed on a generic basis and a sample set of each type of distribution system was walked down.

The SSEL was generated by reviewing the plant drawings, including 1) piping and instrumentation drawings (P&ID), 2) electrical elementary drawings, 3) electrical one-line drawings and 4) HVAC drawings for each of the systems included as part of the success paths.

3.1.3 Analysis of Structure Response

This section summarizes analysis performed to develop comparisons between the PNPP SSE design spectra and the RLE spectra for structural response.

3.1.3.1 Earthquake Spectra Comparison

Three parameters are needed to define the RLE ground spectra. They are peak ground acceleration (pga), structural damping value, and spectrum shape. Based on a review of the PNPP USAR, the following is found:

- The SSE design ground acceleration response spectra are specified for the rock-supported structures at PNPP with peak ground accelerations of 0.15g and 0.1g for horizontal and vertical directions, respectively.

- The structural damping values for reinforced concrete structures are 4% and 7% for OBE and SSE, respectively.

According to Generic Letter 88-20, Supplement 4, PNPP is in the 0.3g peak ground acceleration focused-scope bin for the IPEEE program.

The recommended damping values for the seismic design of nuclear power plants are presented in Regulatory Guide 1.61. The recommended values for reinforced concrete structures are 4% and 7% for OBE and SSE, respectively, and therefore compare favorably with the USAR review findings. This recommended damping is based on the fact that maximum combined stresses due to static, seismic, and other dynamic loading are significantly lower than the acceptance criteria of 0.5 times yield stress and 1.0 times yield stress for the OBE and SSE respectively.

EPRI NP-6041-SL, Section 4, page 4-3 states that "[The Regulatory Guide 1.61] values used in seismic design are, for the most part, considered excessively conservative for a SMA." The recommended damping values for the SMA are presented in Table 4-1 of EPRI NP-6041-SL. The recommended damping values for reinforced concrete structures are 5% and 10% for stresses about 0.5 times yield stress and 1.0 times yield stress, respectively, and shown in Table 3.7.

Based on a review of the design-basis and published recommended structural damping values, it was decided to use a structural damping value of 7% for all Seismic Category I structures within the scope of the PNPP SMA.

3.1.4 Evaluation of Seismic Capacities of Components and Plant

3.1.4.1 Walkdown Results

The seismic capacity walkdowns for PNPP were conducted during an approximate two week period during Refuel Outage 4 and commenced in March 1994. The walkdowns were initiated by Cleveland Electric Illuminating SRT members with assistance from Gilbert/Commonwealth. Additionally, Dr. Paul Smith and Mr. Harry Johnson provided knowledgeable and experienced assistance to the SRT. This external expertise gave added assurance that walkdown techniques and methods were being performed in accordance with the intent of Generic Letter 88-20, Supplement 4.

To maximize efficiency, the walkdowns were planned and performed on an area-by-area basis rather than by system-by-system. They were structured to facilitate a review of components, piping and

supports, cable tray and supports, structures, and seismic interactions in an area during one trip rather than conduct repeated trips to the same area for different reviews. The components associated with both safe shutdown paths plus the containment isolation valves constituted the equipment evaluated during the course of the walkdowns. All of the buildings and structures included in the walkdown paths are considered Seismic Category I.

Walkdown findings were recorded on Seismic Evaluation Walkdown Sheets (SEWS) obtained from Appendix A of EPRI NP-6041-SL. The SEWS are organized to prompt the user on the various issues that should be considered rather than provide a narrowly focused checklist. Equipment anchorage adequacy, spatial interactions, ruggedness of mounted equipment, and unusual configurations were some of the issues that were noted during the walkdowns.

As discussed previously, when components are screened-out, it means that those items are considered to be seismically adequate with no additional evaluation required for the SMA evaluation. The screened-out components would have an expected High Confidence of Low Probability of Failure (HCLPF) number that is at or above the RLE of 0.3g. The screening process makes it unnecessary to calculate HCLPF numbers for screened-out components. The HCLPF capacity is intended to represent an earthquake level in which there is approximately 95% confidence of less than about 5% failure probability. There is no further review activity required for screened-out components. For those components not screened-out, or found to be outliers during the walkdowns, the seismic margin capability (expressed in terms of HCLPF) for any success path is then assessed to be equal to the seismic margin capability of the weakest component in that path.

The following is a summary of various equipment groups that were evaluated during the walkdown activity.

Motor-operated valves (MOVs) were evaluated using cantilever (height/weight) screening criteria and allowable stress limits. The height/weight screening is based on SQUG GIP criteria. It involves evaluating the size of the motor operator, cantilever distance, and actual pipe size. Care was taken to note large operators on small diameter piping. Two valves were considered questionable during the walkdowns. One, the shutdown volume drain valve 1C11-F0181, concerned a potential of instrument tubing flexibility and the other, valve 1E12-F0028A, concerned a junction box cantilevered off the motor operator by conduit. Upon subsequent evaluation, both of these issues were resolved and the components are considered acceptable and able to function for a RLE up to 0.3g.

Instrument racks were walked-down for anchorage adequacy, instrument mountings, and spatial interaction.

Several areas of focus were utilized in the walkdown of the Control Room. The raised floor details were reviewed to determine the adequacy of the anchorage to the embedments in the floor and connections to the termination and control panels. Relay mounting details were inspected and relay chatter is discussed further in Section 3.1.4.2. The seismically qualified ceiling details were reviewed and subsequently inspected to verify the adequacy of bracing. All non-seismically mounted furniture (file cabinets, desks, chairs, drawing racks) were reviewed for impact with panels. This is discussed further in Section 3.1.4.3.

Pumps, as expected from earthquake experience, were found to be seismically rugged. The vertical pumps employed at PNPP were found to have supports which meet the requirements of EPRI NP-6041-SL for long shaft pumps. Two Emergency Service Water pumps had high anchorage stress and were considered outliers during the walkdown. Two pumps had questionable anchorage capacity due to existing high stress levels. However, upon subsequent evaluation the anchorage was determined to be adequate for the RLE with a HCLPF value equal to 0.3g.

Air handlers and fans were found in all areas of the plant. Vibration isolators were targeted in seismic training course as possible weak links in fan anchorage details. No fans were considered outliers during the PNPP walkdown. Ductwork and air handlers appeared, in the judgment of the SRT, to have been built rugged enough to function properly after a RLE.

The hydraulic control units (HCU) for the Control Rod Drive (CRD) system were found to be very well braced for both vertical and horizontal motion. The HCUs are bolted onto common supports which effectively made each perform as part of a single seismic unit.

The air accumulator tanks for the Safety Relief Valves (SRV) were walked down. All of the equipment and anchorages were found to be seismically rugged.

The diesel generators were walked down to assess the adequacy of anchorage and attachment or auxiliary equipment. The auxiliary modules associated with the Division 1 and 2 diesel engines were was considered to be outliers by the SRT during the walkdown due to unknown anchorage details. The Division 3 diesel engine has no auxiliary module. Upon subsequent evaluation, the auxiliary module was determined to have a HCLPF value slightly greater than 0.3g and therefore is considered acceptable.

There were no flat bottom metal fluid storage tanks included in the safe shutdown paths. Existing flat bottom tanks were reviewed, however, for problems posed to safe shutdown equipment if the tanks did rupture during an earthquake. The tanks in the general vicinity of the plant buildings, the Condensate Storage Tank and the Fuel Oil Tank, have dikes to contain any liquid if a rupture occurs. Additionally, the dike for the Condensate Storage Tank, which is located adjacent to the Turbine Building, is seismically designed. Therefore, the rupture of these tanks have no impact on safe shutdown equipment during an RLE.

It was confirmed during the walkdown that there are no masonry or concrete block walls located near any equipment on the safe shutdown equipment list. All structures that house success path equipment or structures that could fail, fall and impact any success path equipment were screened out based on the EPRI NP-6041-SL screening guidelines and verification of the screening assumptions.

A question was raised whether a trolley and hoist on the emergency switchgear cabinets should be better secured. After further analysis and review this was determined to be unnecessary except for the hook which should not be left dangling.

All areas of the plant that contain safe shutdown equipment were checked for seismically induced flooding during the walkdowns. No problems were noted in relationship to this issue.

As mentioned previously, it is not the intent of this evaluation to determine the largest earthquake the site could withstand. However, for those items considered outliers during the walkdowns, HCLPF numbers were calculated, where appropriate, and compared to the RLE of 0.3g and found to meet or exceed that value. Table 3-8 provides a listing of the calculated HCLPF values.

Based on the previous discussion, the plant walkdowns have met the intent of the request in Generic Letter 88-20, Supplement 4, using the guidance in EPRI NP-6041-SL.

3.1.4.2 Relay Chatter Evaluation

The relay evaluation for a focused-scope seismic margins evaluation consists of locating and evaluating low seismic ruggedness relays.

Appendix E of EPRI NP-7148-SL[3-12] identifies the relays that have been determined to be "bad actors" or low seismic ruggedness relays. This list was reviewed against PNPP design drawings using the two safe shutdown success paths as a guideline to establish which drawings to utilize. The review revealed that the only low seismic ruggedness relay type within the safe shutdown success paths is the GE HFA relay. Two areas of concern were noted. The first area of concern was the High Pressure Core Spray (HPCS) diesel generator control circuitry. The second area of concern was in the Reactor Protection System (RPS) Motor Generator Set Control circuitry. Reviewing these two systems for impact which a RLE would have on safe plant shutdown showed that the HPCS circuitry required in-depth evaluation to determine the effects of relay chatter. The RPS system was screened out as not requiring additional evaluation since the low seismic ruggedness relay could result in an inadvertent trip of the recirculation pumps. Since a loss of offsite power is assumed, the recirculation pumps are already assumed lost.

3.1.4.2.1 Method of Analysis

As discussed previously, only the low seismic ruggedness relays within the HPCS diesel generator circuitry required additional evaluation to determine relay chatter consequences. The methodology used for this relay evaluation is specified in EPRI NP-7148-SL. Since this evaluation only focuses on the impact of relay chatter for this known specific low seismic ruggedness relay in a specific system, the approach is as follows:

- Develop the list of GE HFA relays and their associated contacts.
- Evaluate the relays/contacts to determine if chatter is acceptable.
- Evaluate those relays/contacts for which chatter is not acceptable for operator recovery action in the control room.

The first level of relay screening is to identify those relays for which relay chatter can be considered acceptable. Relay chatter can be considered acceptable if the following applies:

- The contacts only energize alarms or annunciator panels.

or

- The contact pairs change position in a portion of the circuit which is initially in a specific state (energized or de-energized) and the change in state of the contacts will not affect the state of that portion of the circuit.

or

- The electrical circuitry is insensitive to relay chatter (i.e., the relay controlled device may respond slowly enough that relay chatter may cause either no operation or only a temporary but acceptable spurious operation of the controlled device).

If chatter of specific relays is not found to be acceptable, the second screening criteria is whether the effects of relay chatter can be recovered from changes of state and associated false alarm by relying upon the operator's actions to reset the affected relays. This requires addressing the following questions:

- Does the control room operator have an alarm to warn of component malfunction?
- Can the control room operator tell from other indications, gauges, meters, etc., that the component has malfunctioned?
- If the control room operator can determine that a problem exists, can he also determine the cause of the problem from the information available?
- Do current procedures prevent the operator from determining the cause?

Once it has been determined that the control room operator can correctly assess the problem, the next step is to determine what means are available to the operator to correct the problem.

- Can the control room operator reset the relay from the control room?
- Does a plant procedure cover the problem?
- If there is no procedure can a new procedure correct the problem?
- Does the control room operator need outside assistance, such as a maintenance technician to correct the problem? If so, is a technician always available when the plant is operating?

If the evaluation determines that the electrical circuitry is sensitive to the relay chatter and the relays cannot be recovered from changes of state and associated false alarms by operator action, other additional recommendations as listed below may be pursued before re-placing the relay with a seismically qualified one.

- Test the relay and/or cabinet in question.
- Redesign and modify the circuit to make the relay function nonessential.
- Relocate the relay to reduce seismic demand imposed upon it.
- Stiffen the relay mounting or the cabinet in which the relay is located.

It should be noted, however, that in practice nearly any relay chatter impact can be easily corrected in a timely manner without compromising the safety functions. The automatic signal or operator action in the control room typically can be used to place the component in correct operation. Only a very few relays have seal-in circuits that cannot be reversed in a timely manner. Even if the relay has a seal-in circuit, there may be sufficient time for the operator to reset the relay at the electrical cabinet. In such a case, procedures should be available in order to take credit for the operator action.

3.1.4.2.2 Details of Analysis

A review of the HPCS diesel generator circuitry resulted in a list of 16 GE HFA relays to be considered. Screening was performed to determine the effects of the chatter on each relay/contact pair and identified those relay/contact pairs which can be considered as chatter acceptable relays or which require operator action to reset the relays. Each drawing was reviewed to determine the impact of chatter of the contacts associated with each important HFA relay. This review is essentially a failure mode and effects analysis (FMEA) of the relay.

Of the 16 HFA relays in the HPCS control circuitry, 11 are classified as chatter acceptable and five as requiring operator action after the RLE.

Four of these five relays which require operator action are associated with the diesel generator protective trips which energize the Engine Trip Lockout Relay K15. Annunciators will clearly identify actuation of the engine protective trips and lockout relay. In order to operate the diesel generator following the seismic event, an operator resets the protective trips by push-buttons which reset the individual relays and then performs a hand reset of relay K15. These actions are performed in the HPCS diesel generator room.

The fifth relay which requires operator actions to restore equipment if the contacts chatter is relay K14 which trips the HPCS D/G output breaker if its contacts close. The operator would be notified of the opening of the HPCS D/G output breaker by the annunicators and can manually reset the tripped breaker.

There are at least 25 minutes available to perform the above actions in accordance with existing Alarm Response Instructions prior to the water level in the RPV reaching the top of active fuel (TAF).

The above review only considers the effects of individual relay chatter. The most likely scenario is that the diesel Engine Trip Lockout Relay would be actuated preventing any start of the diesel. The actuation of this relay also ensures that the majority of the other chatter susceptible relay contacts in the

control portion of the circuits are in de-energized portions of the circuits and that their chatter would have no impact.

3.1.4.2.3 Summary of Results

The chatter of the contacts associated with the low ruggedness HFA relays used in the HPCS diesel generator control circuitry would not significantly impact the safe shutdown of the plant after a RLE event at PNPP. This conclusion is based on one of the following criteria:

- Chatter of the low ruggedness relays is acceptable due to one of the following reasons:

  a) The functions provided by the relay contacts are not required during the period of strong motion and chatter will not result in these functions being available after the strong motion subsides.

  b) The contacts are associated with de-energized circuits or are for indication only or a combination of the two, and will return to their normal state after the strong motion subsides.

or

- Chatter will result in lock out of HPCS diesel generator operation, however, there is sufficient indication what the problem is, the engine trip inputs are easily reset by a push-button switch, and the lockout relay can be manually reset.

or

- Chatter will result in trip of the HPCS 4.16 kV bus diesel generator breaker if initially closed. It is unlikely that the breaker would be closed but if trip does occur, there is sufficient indication what the problem is and the breaker can be manually reset.

3.1.4.3 Seismic Spatial Interactions

The potential for spatial system interaction was considered as part of the seismic walkdown. System interaction issues are considered and noted on the screening and evaluation walkdown sheets (SEWS Forms). The following provides examples of what was included:

- Proximity: The proximity of structures to components and components to components was considered during the walkdown. For example, the proximity of valve operators to structures and other components was considered.

- Seismic II over I: Although this was considered in the plant design basis, it also was considered during the walkdowns. Examples include considerations of instrument lines and the proximity of block walls to equipment.

- Seismic Spray and Flooding: The possibility of water spray and flooding impact on systems was considered during the walkdown.

Specific seismic interaction items looked for during the walkdowns included the following:

- Overhead potential safety impact hazards which warrant concern. This was anticipated not to be a concern due to a continuing II over I program.

- Good housekeeping practices, i.e., portable equipment storage, ladders, cleaning items, etc., properly stored/secured.

- Adequate anchorage of nearby Non-Safety Related equipment to prevent impacting item listed on the SSEL.

- Potential impact effects of swinging items, i.e., piping, lighting fixtures, doors, etc., on other Safety Related items on the SSEL.

- Adequate space between items listed on the SSEL and other items to prevent impact from differential displacement or out-of-phase response, i.e., adequate space between panels/racks and the wall. In lieu of this, adjacent items should be secured together, i.e., adjacent panels bolted together.

- Flexible connections between points of differential movement.

- Flex conduit attached to panels and instrument racks.

- Flex tubing/airline connections, i.e., there should be proper offsets in tubing runs to allow for differential movement.

3.1.4.3.1 Spatial Interaction Results

The seismic capability walkdown identified several suspect spatial interaction problems. These problems revolved around maintenance items, such as a storage cabinet, a test bench, office furniture, a fluorescent light fixture, raised floor impact, and a space heater, falling and impacting electrical cabinets. The interaction concern with these items, besides damage to the cabinets, was inadvertent relay chatter. The method utilized in evaluating this potential concern included overturn evaluation, raised floor rigidity evaluation, effect of impact, and review of essential relays.

The fluorescent light fixture, raised floor and space heater were subsequently evaluated not to be concerns. Resolution of the remaining items include: locking the wheels of the test bench, providing an alternate storage cabinet that is shorter, and relocation of the office furniture. These are considered housekeeping issues and as such are not design changes.

3.1.5 Analysis of Containment Performance

The process used for evaluation of containment integrity was similar to that used for the safe shutdown success paths. In this case the function was prevention of early containment failure. A Containment Integrity Equipment List (CIEL) was developed using plant drawings. This list consisted mostly of isolation valves. Some of the valves on the CIEL were also on the SSEL, primarily the injection valves

for ECCS. A separate walkdown for containment integrity was not performed, but containment integrity components were factored into the overall seismic walkdown.

Based on the Level 2 portion of the PRA, hydrogen control features, such as hydrogen igniters, do not contribute to early containment failure and were not included on the CIEL.

The main objective of the containment analysis is to identify vulnerabilities that involve early failure of containment functions. This includes consideration of containment integrity, containment isolation, and other containment functions.

The guidance in NUREG-1407 states that "generally containment penetrations are seismically rugged, a rigorous fragility analysis is needed only at review levels greater than 0.3g, but a walkdown to evaluate for unusual conditions (i.e., spatial interactions, unique penetration configuration) is recommended." With regard to containment systems, the guidance provided is that "seismic failures of actuation and control systems are more likely to cause isolation system failures and should be included in the examination." The major concern deals with relay chatter, and is addressed by the low seismic ruggedness relay review and the spatial interaction review performed during the walkdowns to preclude inadvertent relay activation.

The containment walkdown consisted of evaluating unusual conditions ,i.e., spatial interactions, unique penetrations, piping hard spots, items/components bridging the seismic gap between the containment liner and integral structure, etc. The containment walkdown was performed by the SRT at the time of the seismic capability walkdown. No unusual conditions or configurations were noted during the containment walkdown. All containment isolation valves met the screening criteria set forth by EPRI NP-6041-SL and all containment penetrations were deemed acceptable.

As stated previously, the main objective of the containment analysis is to identify vulnerabilities that involve early failure of containment functions. The SRT reviews and the walkdown performed of the containment did not identify any vulnerabilities. Therefore, the HCLPF for the containment at PNPP is greater than or equal to 0.3g, based on the results of this evaluation.

3.1.6 Fire/Seismic Interaction

Fire/Seismic interactions, effects of suppressants on safety equipment, and control systems interactions are addressed in the IPEEE. As noted on the SEWS forms, during the walkdown, no adverse interactions were found. Therefore, failures of fire protection systems that lead to release of water will not have a detrimental effect on the capability of PNPP to safely shut down the reactor.

Other system interaction issues relate to the potential for the earthquake to result in a fire. Consequently, a review of the potential fire source was performed to identify any vulnerabilities.

The only area of concern was the Fire Service Fuel Oil Tank associated with the Diesel Driven Fire Pump located in the Emergency Service Water Pumphouse. This equipment is Non-Safety Related, and therefore at the time of the walkdown no qualification test results, nor seismic design existed to determine the adequacy of the fuel oil tank and stand assembly associated with the diesel fire pump. However, subsequent evaluation has determined that the HCLPF value for the tank and stand assembly is 0.56g, which is well in excess of 0.3g.

Therefore, it is concluded that there are no vulnerabilities due to seismic induced fire or flooding concerns.

3.2 USI A-45, GI-131 and Other Seismic Safety Issues

The methodology used in the seismic IPEEE can also be used to address other ongoing programs where seismic events could cause licensee action. USI A-45 "Shutdown Decay Heat Removal Requirements," A-46 "Verification of Seismic Adequacy of Equipment in Operating Plants," GI-131 "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants," and "The Eastern U.S. Seismicity Issue" are all issues which are classified in this category. These have been subsumed into the IPEEE evaluation per the direction of GL 88-20, Supplement 4 and NUREG-1407. In addition to the above issues, PNPP committed to perform a seismic margin assessment following the 1986 Leroy Earthquake. This IPEEE evaluation has addressed and resolved those issues directly related to PNPP, which are discussed in additional detail below.

Two of the issues, A-46 and GI-131, are not related to PNPP and as such are not discussed. PNPP is not required to perform an A-46 review since the plant was constructed after implementation of the IEEE 344-1975 standard for qualification of Class I electrical equipment. As for GI-131, PNPP is not a Westinghouse plant, it is a General Electric BWR/6 with a Mark III containment. Therefore, this issue also does not apply to PNPP.

3.2.1 USI A-45 "Shutdown Decay Heat Removal Requirements"

The external events portion of USI A-45, "Shutdown Decay Heat Removal Requirements," was subsumed into the IPEEE. The decay heat removal issue is addressed by the fact that the SSEL contains the equipment necessary to maintain decay heat removal for a period of 72 hours. This issue was evaluated as part of the IPEEE to determine if the risk due to seismic events impacts the Category I vulnerability classification assessed in the IPE. No potential vulnerabilities were found that would prevent this system from functioning in the design heat removal role. There were no additional USI A-45 related concerns which developed from the application of the seismic methodology at PNPP. Therefore, this issue is considered complete for PNPP.

3.2.2 The Eastern U.S. Seismicity Issue (The Charleston Earthquake Issue)

The Eastern U.S. Seismicity Issue (formerly called the Charleston Earthquake Issue) was subsumed into the IPEEE. The performance of the IPEEE provides a resolution to this issue since the work performed on the probabilistic seismic hazard estimates by the USNRC, LLNL, and EPRI played a key role in determining the review level earthquake bins which each plant site was assigned for performing a SMA. The hazard results used in the binning process included those published in 1989 by LLNL. Recently, LLNL performed new hazard estimates, which the USNRC has published in NUREG-1488.[3-11] These new hazard estimates are lower than those published by LLNL in 1989 but remain higher than EPRI published estimates. Based on this new information and reviews conducted by the USNRC plants binned as 0.3g focused-scope plants may modify their evaluation to allow them to be equivalent to reduced-scope evaluations. This information was transmitted to the utilities in GL 88-20, Supplement 5. The PNPP IPEEE has adopted this approach by not including reactor

internals and soil-related evaluations. However, seismic capability walkdown outliers were still evaluated to a 0.3g RLE. Therefore, based on the new hazard estimates and the performance of the PNPP IPEEE to a 0.3g RLE level, this issue is considered complete for PNPP.

3.2.3 USI A-17 "System Interactions in Nuclear Power Plants"

The external events portion of USI A-17, "System Interactions in Nuclear Power Plants" was subsumed into the IPEEE. Seismic spatial interactions were evaluated in the seismic walkdowns performed for PNPP. These walkdowns were performed in accordance with guidance supplied by EPRI NP-6041-SL. PNPP did not identify any seismic vulnerabilities due to spatial interactions. Therefore, this issue is considered complete for PNPP.

3.2.4 January 31, 1986 Leroy Earthquake Issues

Due to the 1986 Leroy Earthquake, PNPP committed to perform an evaluation to determine the level of available seismic margin. The RLE for the PNPP IPEEE SMA evaluation, set at 0.3g is exactly twice the SSE of 0.15g utilized in the design of the plant. The successful completion of the PNPP SMA at a 0.3g RLE has shown that significant margin does exist for the site and adequately addresses the post-1986 earthquake commitments. Therefore, this issue is considered complete for PNPP.

3.3 References

3-1 USNRC Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

3-2 NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Final Report, June 1991.

3-3 EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Revision 1, August 1991.

3-4 NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," May 1978.

3-5 Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," December 1973.

3-6 Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," February 1987.

3-7 Seismic Qualification Utility Group, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 2, February 1992.

3-8 IEEE 344-1975, "IEEE Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations".

3-9 USNRC Generic Letter 88-20, Supplement 5, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," September 1995.

3-10 Unresolved Safety Issue, USI A-46, "Seismic Qualification of Equipment in Operating Plants," December 1980.

3-11 NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Sites East of the Rocky Mountains," April 1994.

3-12 EPRI NP-7148, "Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality," December 1990.

Table 3-1 - Success Path A Systems

Reactivity Control

Control Rod Drive system
Reactor Protection System
Redundant Reactivity Control System

Reactor Coolant System Overpressure Protection

Safety Relief Valves

Emergency Core Cooling

High Pressure Core Spray System
Division 3 d.c. Power
Division 3 a.c. Power
Emergency Service Water System, Train C
D/G Building Ventilation System, Division 3 Diesel Room
ECCS Pump Room Cooling System, HPCS Pump Room

Containment Overpressure Protection

Fuel Pool Cooling and Clean-up
Division 3 a.c. Power cross-tied to Division 2 a.c. Bus

Table 3-2 - Success Path B Systems

Reactivity Control

Control Rod Drive system
Reactor Protection System
Redundant Reactivity Control System

Reactor Coolant System Overpressure Protection

Safety Relief Valves

Emergency Core Cooling

Automatic Depressurization System (includes SRVs designated as ADS valves)
Residual Heat Removal System, Trains A, B and C (Low Pressure Coolant Injection Mode of Operation)
Low Pressure Core Spray System
Emergency Service Water System, Trains A and B
Emergency Closed Cooling System, Trains A and B
Division 1 and 2 d.c. Power
Division 1 and 2 a.c. Power
D/G Building Ventilation System, Division 1 and 2 Diesel Rooms
ECCS Pump Room Cooling System, RHR A, B & C and LPCS Pump Rooms
Safety Related Instrument Air System

Containment Overpressure Protection

Residual Heat Removal System, Trains A and B (Suppression Pool Cooling Mode of Operation)
Residual Heat Removal System, Trains A and B (Containment Spray Mode of Operation)
Emergency Service Water System, Trains A and B
Emergency Closed Cooling System, Trains A and B
Division 1 and 2 d.c. Power
Division 1 and 2 a.c. Power
D/G Building Ventilation System, Division 1 and 2 Diesel Rooms
ECCS Pump Room Cooling System, RHR A, B & C and LPCS Pump Rooms
Containment Venting via Fuel Pool Cooling and Clean-Up
Containment Venting via RHR, Trains A and B

Table 3-3 - Pre-Screening Basis for Structures

Types of Structures	Basis for Screening Out
Reinforced Concrete Category I Structures	Designed for SSE of 0.15g and construction details verified by drawing review.
Steel Containment Shell	Steel Pressure Boundary keyed to basemat screened-out by drawing and USAR review.
Containment internal structures.	Designed for SSE of 0.15g and construction details verified by drawing review.
Shear walls, footings, and Containment Shield walls.	Designed for SSE of 0.15g and construction details verified by drawing review.
Category I steel and concrete structures.	Designed for SSE of 0.15g and construction details verified by drawing review.
Masonry walls.	Not Applicable - No masonry walls utilized in any Category I structure (verified during walkdowns).
Control room ceilings.	Adequate bracing above ceiling confirmed by drawing review and walkdown.
Impact between structures.	Screened-out by drawing review and piping review.
Category II structures with safety related equipment or with potential to fail Category I structures.	No safety related equipment housed in Category II structures at PNPP.
Dams, levees, dikes	Not applicable to PNPP.
Soil failure modes	Not reviewed based on input from Generic Letter 88-20, Supplement 5.

Table 3-4 - Pre-Screening Basis for Equipment

Types of Equipment	Basis for Screening Out
NSSS Primary Coolant System (piping and vessel)	Screened-out based on review of intergranular stress corrosion cracking documentation.
NSSS supports	Screened-out based on design included both SSE and LOCA loading (confirmed by USAR review).
Reactor internals	Not evaluated based on input from Generic Letter 88-20, Supplement 5.
Control rod drive housings and mechanisms	Lateral seismic support (confirmed drawing review).
Category I piping	Walkdown of representative system performed.
Valves (Active and Passive)	Screened-out (caveats confirmed during walkdown, attention was directed to MOVs for 2 inch and under piping).
Heat Exchangers	Walkdown of anchorage and supports (anchorage calculation review).
Atmospheric storage tanks	Not applicable at PNPP, none in the success paths.
Buried tanks	Piping connection drawings reviewed.
Batteries and racks	Battery racks designed for seismic loading, rigid spacers and end restraints present, batteries supported by side rails (confirmed during walkdown).
Diesel Generators (includes engine and skid-mounted equipment)	Visual inspection of anchorage and mounting of peripheral equipment performed during walkdown.
Pumps (Horizontal and Vertical)	Screened-out (caveats confirmed by drawing review and walkdown).
Fans, air handlers, chillers, and compressors	None mounted on vibration isolators (confirmed during walkdown).
HVAC ducting and dampers	Walkdown of representative system performed.
Cable trays and conduit	Screened-out (caveats confirmed during walkdown).
Active and passive electrical power distribution panels, cabinets, switchgear, MCC's, cabinets, and racks	Visual inspection during walkdown of instrument mounting and anchorage.
Transformers	Visual inspection of anchorage and coil restraint for dry units during walkdown.
Battery chargers and inverters	Visual inspection of anchorage during walkdown and review of qualification testing results.
I & C panels and racks	Visual inspection of anchorage and equipment mounting during walkdowns along with review of qualification testing results.
Temperature, pressure, and level sensors	Screened-out based on review of qualification testing results.

Table 3-5 - PNPP Seismic Input Scaling Factors

Building	Horizontal Scaling Factors	Vertical Scaling Factors
Reactor	1.62	1.62
Auxiliary	1.62	1.88
Control Complex	1.62	1.69
Intermediate	1.62	1.62
Fuel Handling	1.62	1.62
Diesel Generator	1.62	1.88
Emergency Service Water Pumphouse	1.62	1.85

Table 3-6 - PNPP Dominant Natural Frequencies of Structures

Building	Horizontal Scaling Factors	Vertical Scaling Factors
Reactor	2.5 -- 9.0	2.5 -- 9.0
Auxiliary	2.5 -- 9.0	18.0
Control Complex	2.5 -- 9.0	11.0
Intermediate	2.5 -- 9.0	2.5 -- 9.0
Fuel Handling	2.5 -- 9.0	2.5 -- 9.0
Diesel Generator	2.5 -- 9.0	18.0
Emergency Service Water Pumphouse	2.5 -- 9.0	17.0

Table 3-7 - PNPP Percent Critical Damping Values Used for the SMA

Structure or Components	Percent Critical Damping
Piping Systems	5.0
Welded Steel Structural Supports	5.0
Bolted Steel Structural Supports	7.0
Welded Steel Components Including Electrical Cabinets	5.0
Cable Trays	15.0
HVAC Ducting and Supports	7.0
Welded Steel Tanks	
Impulsive Mode	3.0 to 5.0
Convective (sloshing) Mode	0.5
Massive Low-Stress Equipment (pumps, motors, etc.)	3.0
Lightly Loaded Welded I & C Racks	3.0

Table 3-8 - PNPP Calculated HCLPF Values

Description of Equipment Evaluated	Calculated HCLPF Value
Diesel Generator Engine Control Panel - 1H51-P0054A & B	0.58g
Diesel Generator Control Panel - 1H51-P0055A & B	0.86g
Diesel Generator Auxiliary Module - 1R46-S0001A & B	0.30g
Diesel Generator Fuel Oil Day Tank - 1R45-A0003A & B	1.18g
Fire Service Fuel Oil Tank - 0P54-A0002	0.56g
Emergency Service Water Pump Anchorage - 1P45-C0001A & B	0.30g
Shutdown Volume Drain Valve - 1C11-F0181	0.32g
Motor Operated Valve Junction Box - 1E12-F0028A	0.44g
Generic Electrical Equipment Foundation Pads	0.39g

Figure 3-1 - Success Path Logic Diagram (Success Path A)

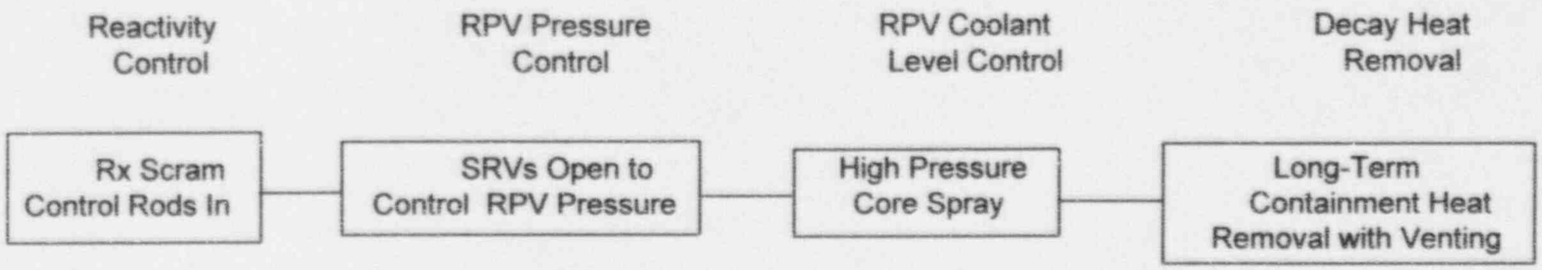
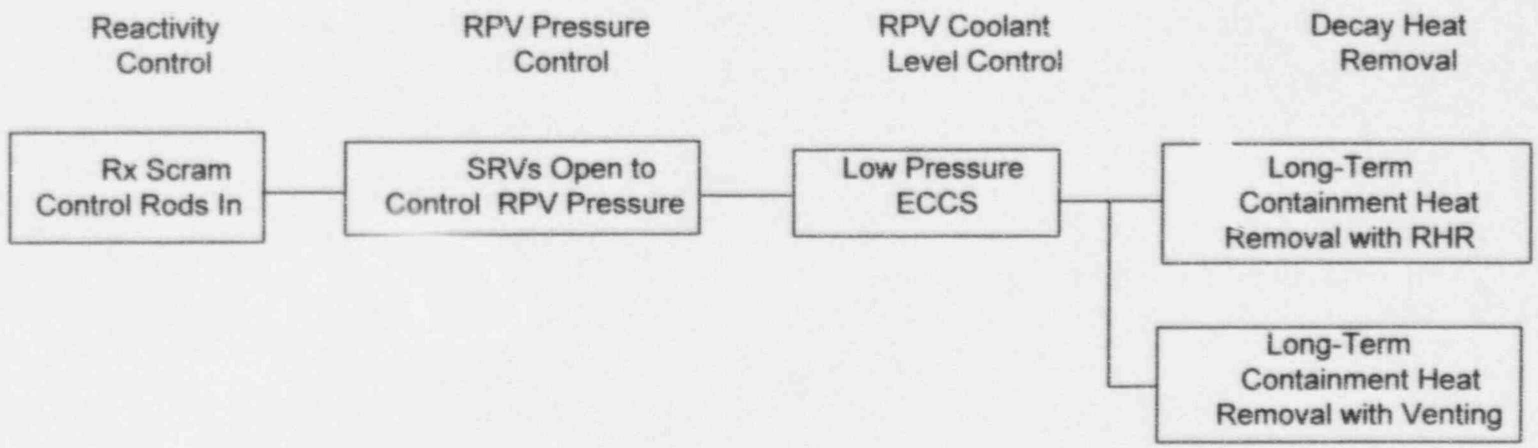


Figure 3-2 - Success Path Logic Diagram (Success Path B)



4 INTERNAL FIRES ANALYSIS

4.1 Methodology Selection

The EPRI Fire-Induced Vulnerability Evaluation (FIVE)[4-2] methodology was selected for the Perry Nuclear Power Plant (PNPP). Specific fire risk assessment issues raised in NUREG-1407[4-1] were addressed as described below.

Fire compartments of potential risk significance were identified using the initial qualitative and quantitative screening steps defined in the FIVE methodology document up to and including Phase I and Phase II, step 2.

Those fire compartments which did not screen out by Phase II, Step 2 were subject to refined fire damage impact evaluations similar to that prescribed in steps 3 to 7 of FIVE, Phase II. The FIVE data and fire modeling techniques were generally retained, supplemented as necessary with information derived from the Fire Risk Analysis Implementation Guide, EPRI NP 3385-01[4-3] and NSAC 181.[4-4] Inter-area or compartment propagation analysis was not required based on the integrity of the fire barriers as demonstrated by the review performed to address the Fire Risk Scoping Study, NUREG/CR-5088,[4-5] issues.

Fire frequencies in particular locations accounted for both U.S. plant generic experience[4-6] and area specific fixed ignition sources. The contribution of transient fuels and sources was accounted for by addressing plant specific procedures for the control of combustibles and ignition sources, as well as by considering periodic inspections for transients.

Coincident failure of accident mitigating equipment was modeled, accounting for random component failures and human error.

Fire Risk Scoping Study Issues were addressed through specifically tailored walkdowns as defined in the FIVE methodology, including seismic fire interactions, effects of fire suppressants on safety related equipment, fire fighter effectiveness, fire barrier effectiveness and control systems interactions.

4.2 Fire Hazard Analysis

4.2.1 Overview

The PNPP plant has already undergone an extensive deterministic fire hazards and safe shutdown review conducted under the 10 CFR 50 Appendix R program and was demonstrated to be in compliance. Although the plant information contained within the Appendix R submittals and supporting documentation provided much of the input to this IPEEE fire analyses, the underlying bases for the two studies are substantially different. Consequently, any findings or conclusions reached concerning potential fire vulnerabilities in no way contradicts or compromises the existing Appendix R analyses. Differences in the Appendix R and fire IPEEE methodologies include:

Issue	Appendix R	Fire IPEEE
Extent of equipment damage	Generally assumes all equipment in fire area is damaged	May use fire modeling to determine extent of damage from specific sources
Likelihood of fire	Assumes fire may occur regardless of sources present	Evaluates fire frequency as a basis for estimating actual risk
Coincident equipment failures	Assumes equipment unaffected by the fire will be available for plant shutdown	Considers random failures of unaffected equipment coincident with fire damage
Operator reliability	Assumes operators will take actions directed by procedures having demonstrated adequate time and access is available	Considers potential operator error and associated reliability
Offsite power	Assumes offsite power unavailable	Only assumes offsite power unavailable if shown to be damaged by the fire, otherwise considers random failure probability coincident with the fire
Fire protection systems	Has specific requirements regarding installation and operability depending upon fire hazard	Only addresses and credits fire protection system operability for risk significant fire scenarios

In theory the contribution to core damage frequency from fires anywhere in the plant may be assessed in detail. However this is impractical, due to the large number of possible scenarios, and is also unnecessary, since fires in many plant areas are incapable of causing significant damage no matter how severe they become. Consequently, the first stage in performing a fire analysis was to perform a systematic screening of all fire areas to identify those plant locations where fires may present a significant hazard.

The FIVE methodology qualitative and quantitative screening procedures were applied, as described below. The results of this screening are presented in Section 4.2.3 of this report.

4.2.1.1 Qualitative Screening Analysis of the Fire Areas

Steps 1 through 6, described below apply to all plant areas with the exception of the containment. For various reasons, principally low fire potential, FIVE permits a less stringent criteria to be applied to the screening of containment fires. Step 7 describes this approach. Further details of the methodology can be found in Sections 5.3 and 6.3.1 of the FIVE methodology document.[4-2]

The purpose of this task was to identify the boundaries of the plant fire areas, together with the location of equipment and cables which, if damaged by fire, would cause a plant shutdown and degradation of shutdown paths identified in the PNPP Appendix R Safe Shutdown Capability Report.[4-8] That information was then used in this subtask as a basis for systematically screening out fire areas from further consideration using the non-probabilistic criteria developed in the FIVE methodology document. Further use was made of the information in subsequent tasks.

Step 1 - Identify Plant Safe Shutdown Systems

The safe shutdown analysis described in the Appendix R Safe Shutdown Capability Report, and PRA models were reviewed to identify the PNPP safety related safe shutdown systems and trains. Both front line and support systems were listed. In the FIVE methodology, the target shutdown mode of operation selected should be consistent with the plant's PRA (FIVE, Section 2-10). In general, the PNPP event trees were constructed to model success paths which lead to hot shutdown. The combination of systems required to achieve this stable condition for a period of 24 hours, following various types of initiating events, is discussed in Section 3 of the IPE.[4-7]

Step 2 - Identify Fire Areas

The plant was initially divided into fire areas which are physically separated from one another and have boundaries which comply with the requirements of the FIVE methodology (Definition 2.2). That is, fire area boundaries must have at least a two hour rating with all penetrations sealed with assemblies which have an equivalent rating, or an engineering evaluation must have been performed to determine whether the boundary can withstand the fire hazards within the area and protect important equipment in the area from a fire outside the area.

In several cases, compartments within fire areas were also defined. The compartment boundaries, although not fire rated, were defined where barriers exist which may provide a substantial confinement for heat and products of combustion. Further evaluation of the effectiveness of designated compartment boundaries is performed, as necessary in Step 6.

Step 3 - Identify Safe Shutdown Equipment in Each Fire Area and Compartment

Appendix R safe shutdown equipment and the associated cabling located in each fire area are identified in Safe Shutdown Capabilities Report. The location of selected non-Appendix R safety related equipment and cables was determined through drawing review and cable tracing effort. A summary of the damaged and undamaged safe shutdown equipment in each area and compartment was documented.

Step 4/5 - Perform Fire Area vs Safe Shutdown System Screening

A review of potential fire induced initiating events (e.g., loss of offsite power, LOCA or manual shutdown) was performed. This information, combined with that obtained in Step 3 was then used to determine whether a fire area may be screened out using qualitative screening criteria defined within the FIVE methodology, as follows.

A fire area was screened out at this stage if:

- none of the associated compartment contains any Appendix R safe shutdown equipment or cable

  and

- a fire in any of the associated compartments may not cause a demand for shutdown

Step 6 - Compartment Identification, Interaction and Screening

At this stage the FIVE methodology also provides the option of evaluating individual fire compartments. A fire compartment interaction analysis was performed in this step to determine if inter-compartment propagation could be shown to be ruled out. The evaluation of fire spread between compartments was performed with aid of the screening criteria presented in Section 5.3.6 of the FIVE methodology. Factors such as the characteristics of the inter compartment barrier, combustible loading, and installed fire detection and suppression systems were considered.

Where all boundaries of a compartment could be screened out, i.e., risk of propagation is insignificant, then the screening criteria applied in Step 4/5 was re-applied considering only the potential for damage within the compartment.

Those compartments not screened out at this stage were analyzed further in the quantitative analyses.

Step 7 - Containment Fire Evaluation

FIVE indicates that containment fires are generally not expected to be risk significant and consequently do not provide data for quantitative evaluation. Several reasons are provided below.

- A hot gas layer is unlikely to form in most areas which can damage cables.

- The small number of historical events during plant operation, a large percentage of which were reactor coolant pump (RCP) fires, are unlikely to occur in the future due to oil collection system design improvements.

- Previous fire PRAs did not show containment fires to be risk significant.

Consequently, FIVE recommends a qualitative assessment to determine if there are any unique plant features which make the plant more susceptible than those plants examined previously. Specific issues to be addressed are:

- plant experience which might indicate fires have occurred frequently

and

- the potential for redundant trains of critical equipment to be exposed to the same fire plume or be in a confined space subject to damage by a hot gas layer.

4.2.1.2 Fire Frequencies

The purpose of this task was to evaluate the fire frequency for compartments which were not screened out in the qualitative screening process described above. These frequencies are intended for use in the quantitative screening evaluation and detailed fire analysis.

For PNPP, the fire frequency calculations were performed using the methods provided in the FIVE methodology, Phase II, Step 1, and generic fire data information provided in the Fire Events Data Base.[4-9] The approach requires the generic fire data to be weighted according to the specific types and quantity of ignition sources present in the area being evaluated. FIVE provides detailed guidance for determining both "Location Weighting Factors" and "Ignition Source Weighting Factors" and a formalized documentation process for recording input data and calculating fire frequencies.

The number, type and location of each ignition source was initially evaluated from PNPP drawings and PNPP Master System Operating Equipment data base. The information was modified as necessary as a result of plant walkdowns.

The area/compartment ignition sources and the fire frequency calculations were evaluated using the FIVE software package.

4.2.1.3 Quantitative Screening Analysis

The FIVE methodology permits screening of a fire area/compartment when either of the following can be shown to be less than 10^{-6}/year:

- the total area/compartment fire ignition frequency
- the fire ignition frequency multiplied by the conditional core damage probability given loss of all equipment/cable in the compartment

At this screening stage, the PRA model was used to determine the conditional core damage probability (CCDP).

4.2.2 Assumptions and Other Modeling Considerations for Screening

4.2.2.1 Success Criteria

The Appendix R analysis included equipment necessary to achieve cold shutdown. The IPEEE fire analysis was limited to achieving a stable hot shut down state for a general transient, represented as a loss of the power conversion system (PCS), and loss of offsite power using the same success criteria identified in the Perry PRA. The potential for fire induced LOCAs was reviewed and determined to be insignificant with respect to risk.

The first step of the FIVE methodology is to identify plant safe shutdown systems. This analysis primarily takes credit for the same shutdown systems as defined in the Appendix R Safe Shutdown Capability Report for PNPP to achieve stable hot shutdown since the location of equipment and cable associated with these systems is known. However, experience has shown that it is also advantageous at this stage in the analysis to also be able to take credit for some selected additional systems. For the IPEEE fire analysis, therefore, off-site power supplies to the emergency buses, high pressure core spray system, condensate/feedwater, condensate transfer alternate injection, fast firewater alternate injection and containment venting were also credited. Specific analyses were performed to identify and locate the necessary components and cables to support these functions.

In summary the following safety functions are required for achieving stable hot shutdown:

Reactivity Control

In both the Appendix R and the PRA models, reactivity control is accomplished through insertion of control rods. The effect of a fire will most likely cause rod insertion, through de-energization of the RPS, rather than inhibit its operation. There are also several proceduralized methods to manually de-energize the RPS by taking action either within or outside the control room. The potential for fire events to prevent adequate reactivity control is therefore insignificant.

Primary System Integrity

Reactor coolant system (RCS) system integrity is required to ensure proper RCS recirculation, pressure control and inventory control can be maintained and heat removed through a steam relief valves.

High Pressure Injection

The reactor core isolation cooling system (RCIC), high pressure core spray (HPCS) or condensate/feedwater provide a high pressure flow of water to reactor vessel to assure that sufficient water inventory is maintained in the reactor vessel permitting adequate core cooling to take place.

Automatic Depressurization System

The automatic depressurization system (ADS) employs nuclear boiler system pressure relief valves to relieve high pressure steam to the suppression pool. This system operates when the high pressure injection systems are unavailable. ADS reduces the reactor pressure vessel (RPV) pressure so that flow from the low pressure injection systems can enter the reactor vessel to cool the core and limit fuel damage.

Low Pressure Injection

Low pressure injection may be used to maintain RPV inventory in the event of high pressure injection failure and successful depressurization of the RPV. Low pressure injection may be accomplished using Trains A, B or C of the residual heat removal (RHR) system or the low pressure core spray (LPCS) system. In the event that the safety related low pressure injection systems are unavailable, low pressure RPV injection can be accomplished with the condensate transfer system or the fire protection system.

Long-term Heat Removal

Long term containment heat removal may be achieved using either Train A or Train B of the RHR system in the suppression pool cooling mode or containment spray mode of operation. For continued success of RCIC, the suppression pool cooling mode of operation is required. Alternatively, decay heat may be removed through containment venting via the fuel pool cooling and clean-up system lines or via the RHR lines and containment spray headers.

Process Monitoring

Instrumentation, both in the control room and at the remote shutdown panel, has been provided to monitor primary system variables. Additional indication has been provided to monitor and control the safe shutdown systems including support systems. This indication generally includes pump discharge flow and flow rate through essential heat exchangers. Necessary tank level indication is also provided.

Support Systems

Emergency service water, emergency service water screen wash, emergency closed cooling and various HVAC systems were credited in the Appendix R Safe Shutdown Capability Report as being required to support continued operation of the front line systems.

Passive Mechanical Components

Valves, heat exchangers, and piping systems, which are exposed to the fire, remain structurally intact as a pressure barrier or structural member of a system. Mechanical components that are exposed to a fire may be actively operated after the fire is extinguished if a local operational capability exists, i.e., a handwheel. However, if there are restrictions for entering the fire affected location, e.g., restriction due to severe smoke levels, then the restrictions on performing the local operation noted in the Safe Shutdown Capability Report were also accounted for in this analysis.

In addition to the above functions, two more support functions have to be provided during a fire event, namely emergency lighting and communication. These support systems will not be included in this analysis since emergency lighting is located in each fire area with sufficient Appendix R analyses to establish reliability. Portable lighting can also be utilized as needed. Emergency lighting may be incorporated, as necessary, in the human reliability analysis performed as part of the PRA process of the quantitative analysis. Emergency communication can be achieved by utilizing portable communication systems.

4.2.2.2 Fire Initiated Events

Consistent with the FIVE methodology it has been assumed that a fire will cause a demand for plant shutdown unless it can be shown with confidence that the damage sustained will not result in an automatic plant trip or the degradation of safety related equipment is such that a shutdown would not be required within 8 hours in order to meet technical specifications.

The analysis utilizes the PNPP PRA initiating event categorization. However, those initiating events which cannot be induced as a result of fire can be ignored. In summary, the categories of fire induced initiating events which are evaluated include:

- Loss of Off-Site Power (Type T1)
- Transient with Loss of Power Conversion System (PCS) (Type T2)
- Transient with PCS Available (Type T3A)
- Fire induced LOCA due to stuck open valve

Loss of Offsite Power: One area where the fire may result in a loss of offsite power has been specifically identified. This fire area is the Unit 1 Turbine Power Complex, 1TPC.

Transient with Loss of PCS: A transient with the loss of PCS, i.e., a transient in which the MSIVs close, can occur as a result of many different fire induced faults and may even be initiated by the operator in response to spurious signals or technical specification limitations. A loss of PCS transient is assumed to occur given a fire in an area unless specific discussion is provided which excludes any type of fire induced initiating event or identifies a more onerous event, i.e., LOCA or loss of offsite power.

Transient with PCS Available: A transient with PCS available, i.e., the MSIVs remain open and the condenser is available as a heat sink, can occur as a result of the same type of events described in the Transient with Loss of PCS. A transient with PCS available is assumed to occur given a fire in a specific area when no circuits or components are involved that would cause closure of the MSIVs.

Loss of Coolant Accidents: Fire induced LOCA events may occur as a result of electrically operated valves in the RCS spuriously operating. In the Safe Shutdown Capability Report, an evaluation is also performed to demonstrate that the RCS integrity will not be compromised as a result of spurious valve operations resulting from fire induced electrical shorts.

Since the FIVE methodology requires the consideration of random equipment failures in addition to fire induced faults (whereas the Appendix R analysis does not), the Safe Shutdown Capability Report analysis of LOCA events was reviewed in order to identify any specific areas where the conclusions reached in the Appendix R study may not necessarily satisfy the IPEEE requirements.

4.2.2.3 Ignition Sources

Ignition source information was initially gathered from plant layout drawings and a print out of the Perry Equipment Master System data base, sorted by location. The ignition information was then verified or adjusted as necessary based on plant walkdowns. The approach for counting and categorizing plant locations and ignition sources followed the guidelines provided in the FIVE methodology and supporting documentation.

All electric cable construction utilized at PNPP satisfies the requirements pass the IEEE No. 383 flame test. Thus there is no contribution to fire frequency from self ignited fires in non qualified cable or associated junction box fires at PNPP.

4.2.3 Analysis Results

4.2.3.1 Qualitative Screening of Fire Areas and Compartments (Phase I)

The Updated Safety Analysis Report (USAR) designates all fire area barriers at PNPP as 3 hour rated thus satisfying the requirements of the FIVE methodology for designation as independent areas with no potential for as inter-area fire propagation. Four of these areas were subsequently subdivided into compartments, namely the Control Complex elevation 599'-0", the Fuel Handling Building, the Intermediate Building and the Unit 1 Turbine Power Complex. The interfacing boundaries of these compartments were subsequently examined using the fire compartment interaction analysis (FCIA) approach and were found to meet the screening criteria. A list of fire areas and associated compartments is included as Table 4-1.

The systems which are potentially lost due to a fire in each area/compartment are summarized in Table 4-2.

The qualitative screening analysis was completed using the FIVE methodology, as discussed in Section 4.2.1. The following compartments were screened out:

Fire Area/Compt	Description
1ABSTWN	Auxiliary Building Unit 1 North Stairwell
1ABSTWS	Auxiliary Building Unit 1 South Stairwell
1CC3b	Control Complex Elev 620 Unit 1 Div 3 Switchgear Room
1CC4i	Control Complex Elev 638 Unit 1 Computer Room
1DG1b	Diesel Building Elev 620 Unit 1 Div 3 D/G Room
1TPC/2	Unit 1 Turbine Power Complex Sub-basement
2CC3a	Control Complex Elev 620 Unit 2 Div 1 Switchgear Room
2CC3b	Control Complex Elev 620 Unit 2 Div 3 Switchgear Room
2CC3c	Control Complex Elev 620 Unit 2 Div 1 Switchgear Room
2CC3d	Control Complex Elev 620 Unit 2 Remote Shutdown Panel
2CC3e	Control Complex Elev 620 Unit 2 Switchgear Room Corridor
2CC4 638/654	Control Complex Elev 638/654 Unit 2 Elevator Corridor
2CC4f	Control Complex Elev 638 Unit 2 Div 1 Cable Chase
2CC4i	Control Complex Elev 638 Unit 2 Computer Room
2CC5a	Control Complex Elev 654 Unit 2 Control Room
2CCSTW	Control Complex South Stairwell
2DG1a	Diesel Building Elev 620 Unit 2 Div 2 D/G Room
2DG1b	Diesel Building Elev 620 Unit 2 Div 3 D/G Room
2DG1c	Diesel Building Elev 620 Unit 2 Div 1 D/G Room
CC2 (cmpt CC2/3)	Control Complex Elev 599 (Chemistry Lab)
ESW1b	ESW Pump House - Diesel Fire Pump Room
FH (cmpt FH2b)	Fuel Handling Building (Elev 599 Refueling Equipment)
IB (cmpt IB5)	Intermediate Building (Elev 582)
RWB	Rad Waste Building
UNIT2 (various)	Unit 2 (Turbine Bldg, TPC, Aux Bldg, and Steam Tunnel)

The screening results through Phase II are provided in Table 4-3.

4.2.3.2 Fire Ignition Frequencies for Quantitative Screening

For each of the fire compartments located within the fire areas that were not screened out in the previous steps, estimates of the fire ignition frequency were prepared for use in the quantitative screening and detailed analyses. These estimates were based on data from the Fire Events Database for US Nuclear Power Plants and adjusted for PNPP using information from plant arrangement drawings or other documentation and equipment databases. A summary of the database appears in the FIVE methodology document. The frequencies were then updated based on the plant walkdowns that were performed for this purpose. Table 4-3 contains a summary of the fire area/compartment ignition frequencies.

4.2.3.3 Quantitative PRA Screening Analysis (Phase II, Step 2)

The FIVE methodology includes a second level of screening which provides for a conservative estimation of the contribution to core damage frequency. Generally, all equipment and cable in an area/compartment were assumed to fail due to a fire. However, in the case of compartment CC2/1 credit was taken for Appendix R separation to limit the damage, in accordance with the FIVE approach. Using an event tree representative of the most significant failure, the conditional core damage probability was calculated. If the product, F2, of the area/compartment ignition frequency, F1, and the conditional core damage probability, P2, was less than 10^{-6}/yr, the area/compartment was screened out.

The majority of the fire areas/compartments were screened out using this approach. The results are shown in Table 4-3. More refined analyses of the unscreened plant areas are discussed in Sections 4.3 to 4.6.

4.3 Review of Plant Information and Walkdown

4.3.1 Plant Information Sources

For this analysis, plant information was obtained from plant drawings, plant instructions, and other documents such as the PNPP Individual Plant Examination (IPE),[4-7] the Appendix R Safe Shutdown Capability Report[4-8] and the Updated Safety Analysis Report (USAR).[4-9]

The USAR defines the fire area boundaries, fire protection features, combustible loadings as well as location maps showing the position of Appendix R safe shutdown components. The Appendix R Safe Shutdown Capability Report defines the post fire safe shutdown design philosophy and evaluates the potential damage to safe shutdown systems within each fire area. This report also addresses associated circuits and the Remote Shutdown System capability which was specifically installed to mitigate transients and equipment loss due to fires in the control room. In addition, assessments for individual fire zones and fire areas are provided and it contains a complete discussion of the plant's fire protection program including: organizational responsibilities: fire prevention abilities (control of combustibles and ignition sources, and control of fire protection system impairments); employee training; fire brigade manning, response, training, drills, and equipment; and fire protection systems (detection, alarm and suppression systems).

Implementation of the fire protection program is governed by specific Plant Administrative Procedures (PAPs) and Periodic Test Instructions (PTIs) which were also reviewed as part of this IPEEE project. Examples include; PAP-1910 (Fire Protection Program), PAP-1914 (Fire Protection System Operability) and PTI P54-P00041 (Semi Annual Fire Door Inspection).

The above reports provide information on the method of Appendix R compliance, combustible loading analysis, exemption requests and engineering analyses. The existence of these reports is pre-supposed by the FIVE methodology. They were used to obtain the fire area boundary definitions, the safe shutdown and other safety related functions which may be disabled in each fire zone, and the combustible loading characteristics, including cable tray loadings and transient material inventories.

For compartments requiring detailed analyses, specific power and control wires were identified through reviews of electrical schematics and control wiring diagrams. Cable routes were then obtained from the PNPP cable pull sheets. Plant cable tray and conduit drawings were utilized to identify physical locations.

Plant drawings were also used for locating equipment to obtain information about the number and type of ignition sources and targets in each fire area. The plant specific data were used to relate generic fire frequency data obtained from the EPRI fire events database to specific PNPP fire compartments.

Post fire safe shutdown procedures embodied within the Emergency Operating Procedures, Off Normal Instructions, e.g., ONI-C61 "Evacuation of Control Room Unit 1," were utilized in defining applicable recovery actions.

References to plant specific documents are identified under applicable sections of the submittal.

4.3.2 Outstanding Modifications

NUREG-1407 requires licensees to provide a discussion of the status of Appendix R modifications. There are no outstanding Appendix R modifications at PNPP with the exception of the Thermo-Lag™ issue.

The ongoing modifications involving Appendix R are enveloped in the resolution of the Thermo-Lag fire barrier issues stated in the Perry response to USNRC Bulletin 92-01. This involves the isolation and evaluation of 1 hour rated raceway barriers. The resolution of this issue includes an evaluation of the function of the Appendix R Safe Shutdown circuits protected to determine if a fire rated barrier is needed to meet the requirements of Appendix R, Section III G. This will also eliminate some of the components and circuits presently identified as Appendix R Safe Shutdown circuits. The overall impact of the Thermo-Lag project on the IPEEE would result in the IPEEE internal fires evaluation to be conservative, i.e., in general, the IPEEE did not take credit for 1 hour rated raceway barriers.

There were two exceptions to this. In Fire Compartments DG1d and CC2/4, circuits from each of the Appendix R Safe Shutdown divisions pass through these fire areas. Credit was taken for a 1 hour rated raceway barrier for one of the divisions in DG1d. The Thermo-Lag project is evaluating this area for upgraded fire protection barriers. The circuits in CC2/4 are being assessed in the Thermo-Lag project. If necessary, upgraded fire protection barriers will be placed on some circuits.

4.3.3 Plant Walkdowns

Several plant walkdowns were performed for the PNPP fire analysis. The main objective of these walkdowns was to gather plant data which cannot be readily derived from documented sources in order to perform the screening and detailed analyses, as well as complete the evaluation of Fire Risk Scoping Study issues. Another objective was to confirm that the information which was obtained from documented sources is consistent with the as-built, as-operated plant. The main walkdown activities are discussed below.

Walkdowns were carried out to verify plant conditions for the Fire Risk Scoping Study evaluation. Information pertaining to potential seismic-fire interactions, e.g., seismically induced fires from hydrogen, or from storage of diesel oil, fuel oil or lubricating oil; or seismic actuation of fire suppression systems, were obtained.

A walkdown was also performed primarily to verify the information in the qualitative and quantitative screening analyses and obtain specific information on the type and location of ignition sources in each compartment.

Several additional walkdowns were performed on an as-required basis with the aim of obtaining information regarding specific plant configurations. For example, information was obtained on:

- Type of sealing and venting of electrical cabinets
- Type of confinement provided for potential oil spills
- Separation of redundant components/wireways provided within control room cabinets
- Type and proximity of fire detectors to specific fire sources
- Proximity of exposed combustibles to ignition sources
- Location and concentration of combustibles with respect to inter compartment boundaries

All walkdowns were carried out by PNPP, Vectra and NUS personnel. The participants were either fire protection engineers or PRA specialists who, between them, possessed the following qualifications:

- Familiarity with the Appendix R Safe shutdown paths, equipment and cable raceway layouts and Post Fire Shutdown Procedures
- Familiarity with the plant fire protection design and standards, including fire barrier characteristics, fire detection and suppression systems and fire prevention measures
- Understanding of PRA models and assumptions made in fire induced conditional core damage probability analysis

4.4 Fire Growth and Propagation Modeling

The second part of the fire analysis dealt specifically with the potentially significant fire areas and compartments which could not be eliminated as part of the qualitative and quantitative screening process.

As previously stated, the initial quantifications assumed all vulnerable equipment in the fire area/compartment would be damaged in the event of fire. This can obviously be very conservative in many cases. For example, fire damage to an elevated cable tray from a small to medium size fire, on the opposite side of a room, with no intervening combustibles, is highly unlikely if not impossible. Using fire damage calculations, many of the fire sources can be shown to be benign based on their size and target range, and can be screened from further consideration. Furthermore, if a fire compartment is protected by an automatic fire suppression system the initial estimate of the probability of equipment damage due to fires can often be substantially reduced. Through a process of eliminating many of the ignition sources as potential causes of significant equipment damage or reducing the estimated probability of such damage, a more realistic (less conservative) estimate of the fire induced risk can be obtained.

4.4.1 Fire Scenario Identification and Evaluation

The detailed fire analysis is intended to refine the predicted extent of fire damage, as well as take credit for fire detection and suppression systems. Fire damage is refined by studying in more detail, the ignition source/target relationships to identify and evaluate potential fire spread and more realistically predict the likelihood of equipment damage.

Fire areas/compartments requiring additional evaluation are listed in Table 4-10.

In performing the detailed fire analysis the following steps were generally applied for each individual fire area/compartment.

1. A fire scenario assessment was made for each individual fire ignition source identified during the ignition frequency analysis, considering the spatial relationship of the ignition sources to equipment required for safe shutdown and intervening combustibles as well as the size of the enclosure and the fragility of the targets in question. Those ignition sources which were incapable of causing significant damage to safe shutdown equipment, either directly or as a result of fire propagation, were screened out.

   If the product of the residual fire frequency, fires capable of causing some damage, and the original conditional core damage probability, P2, was less than 10^{-6} then the area was screened out and the analysis terminated .

2. An assessment of the probability of failure of suppression prior to damage was made based on the type of detection and suppression systems available, P_{fs}.

3. The specific damage resulting from individual fires source and propagation scenarios was evaluated and a scenario specific conditional core damage probability evaluated, P2'.

4. Finally the frequency of core damage due to individual fire scenarios F_3' was then evaluated from:

   $F3' = F1' \times P_{fs} \times P2'$ Where F1' is the ignition frequency of a specific fire source.

If the sum of all the F3' frequencies for a given area/compartment is less than 10^{-6} then the area is screened out. For those areas/compartments which do not screen out, containment performance, decay heat removal and potential vulnerability issues are addressed.

4.4.2 Fire Modeling

The detailed fire modeling was performed in accordance with the FIVE Methodology. Additional guidance was obtained from the EPRI Fire PRA Implementation Guide,[4-3] as needed. The ignition sources, and their respective initiating event frequencies, F1', and the targets were previously determined for each fire area/compartment during the quantitative screening process. For the detailed analysis the ignition source size and burning characteristics, the target damage criteria, and the source-target spatial relationship are also required. Once these items are known, a fire model can be developed. Four general types of fire damage phenomena need to be addressed:

- Damage to an elevated target located in the fire plume directly above a fire source
- Damage to an elevated target located within the ceiling jet but outside the plume
- Damage to a target located in the hot gas layer, but outside the plume and ceiling jet
- Damage to a target located next to the fire source, exposed to direct thermal radiation

The effects of these phenomena may compound, e.g., a target placed in the plume of a fire may receive a sub-critical heat flux from plume as well as a sub-critical heat flux from a hot gas layer, but the combined heat flux is critical. For these situations, multiple analyses were necessary to evaluate the ultimate potential for damage.

These phenomena were analyzed by utilizing Worksheets 1, 2 and 3 of the FIVE methodology. The worksheets, and instructions for their use, can be found in Attachment 10.4 of the EPRI FIVE document. Input into these worksheets includes physical dimensions, and physical characteristics of sources, intervening combustibles and targets. The physical dimensions required include the spatial relationship between the sources and targets, component room location, e.g., next to wall, in corner, etc., existence and location of barriers, e.g., fire shielding, and room size. These dimensions were obtained mainly from walkdowns and drawings. The relevant physical characteristics are principally the heat release rate and duration of the sources and intervening combustibles, as well as the damage and ignition threshold temperatures and fluxes for targets. The physical characteristics discussed below were selected for this screening analysis.

4.4.3 Characterization of Ignition Sources

For the areas/compartments analyzed in this report, the ignition sources previously determined can be separated into six main groups:

- large electrical cabinets
- large pumps and compressors
- cable and junction boxes
- miscellaneous small fixed sources
- transient sources
- welding sources.

These ignition sources are characterized in the following paragraphs.

Large Electrical Cabinets

The total frequency of fires associated with electrical cabinets in each area/compartment was determined as part of the screening analysis. Each cabinet was then allocated a fraction of the total cabinet frequency based on its relative size number of sections or bays or the inverse of the total number of cabinets in the area/compartment. The latter approach was used where cabinets were approximately of equal size. For the purpose of this analysis all cabinet fires are conservatively assumed to result in cable ignition inside the cabinet.

All cabinets at PNPP contain IEEE qualified cable. For all cabinets containing IEEE qualified cable only, a heat release rate (HRR) of 65 Btu/s is suggested in Reference 4-3 based on a review of the Sandia Cabinet Fire Tests.[4-10] However, since electrical cabinets and panels come in a wide range of sizes, a single HRR was considered to be unrealistic.

For segregated motor control centers (MCC) and small electrical panel fires, the 65 Btu/s HRR was applied. For larger vertical cabinets and benchboards a HRR of 400 Btu/s was assumed based on the screening value for closed door vertical cabinet fires containing unqualified cable (Appendix I of Reference 4-3). Furthermore, the following assumptions were made concerning the cabinet configuration.

Cabinet Configuration	Fire Modeling Assumption
No ventilation	Cannot propagate (self limiting due to oxygen starvation)
No top penetration	Fire source at height of ventilation louvers, limited radiation, subtract 20% of HRR.
Open top penetration	Source at top of cabinet.
Sealed top penetration - "Fire rated" or judged to be capable of confining a fire	Same as no top penetration.
Sealed top penetration - "Non-fire rated"	Same as open top penetration.
Top penetration is a conduit, and either: a) D < 2" and L > 1', b) D = 2" and L > 3', or c) Conduit has rated seal	Same as no top penetration if otherwise ventilated. Same as no ventilation if not otherwise ventilated.

The total combustible loading for all electrical cabinets in each compartment was in some cases available in the USAR.[4-9] From this reference, and walkdown information, an estimate was made as to the total combustible loading of an individual cabinet. If a cabinet combustible loading was not available, individual cabinet combustible loadings were estimated based on known loadings of other sources, and cabinet size information gained during the walkdown.

The fire duration is dependent on the HRR, combustible loading and mass burnout fraction, i.e., the fraction of mass consumed prior to burnout. The mass burnout fraction for electrical cabinets was set to 0.7 based on review of experimental data obtained in the Sandia Cabinet Fire Tests, and documented in NUREG/CR-4527, Volume 1 of 2.[4-10]

Large Pumps and Compressors

Two types of ignition can arise from pump and compressor fires; the motor windings can ignite due to some electrical fault, or bearing grease and oil may leak and burn. In either case, the HRRs are not easily defined.

For motor fires a conservative bounding heat release rate equivalent to a small electrical cabinet is recommended, i.e., 65 Btu/s.[4-3] The combustible loading was obtained from the USAR, or assumed from other known loadings.

For oil fires, the burning rate was determined on a case by case basis, and is dependent upon the physical properties of the oil, and the spill size. Confined spill areas are dependent upon the amount of oil in the sump, and any confinements, including trays, dikes, floor slopes and drains, etc. Unconfined spill areas can be determined using Table 3 of FIVE. The HRR was then determined as the product of the spill area, and the unit heat release rate and combustion efficiency for the oil. The burn duration is dependent on the amount of spilled oil and the HRR.

Based on the Fire Events Data Base,[4-3] 18 percent of pump motor fires and 2 percent of compressor fires involved oil spills. These fractions were be factored into the analysis as necessary where such fires were determined to be significant.

Cable and Junction Boxes

As indicated earlier self ignited cable fires for PNPP were ruled out due to the cables being IEEE 383 rated. For similar reasons significant junction box/cable splice fires and welding induced cable fires which do not involve transient materials were eliminated in the detailed analysis. This approach is consistent with that adopted in NSAC 181.[4-4]

Miscellaneous Small Fixed Sources

There are numerous fixed fire sources with a low combustible content which are legitimate fire ignition sources. These sources include small pumps/compressors with either sealed bearings or lubricated by small amounts of grease or oil, small dry transformers and small ventilation system fans. Due to their low combustible loading, a conservative bounding heat release rate equivalent to a small electrical cabinet, or motor fire is recommended, i.e., 65 Btu/s. If the combustible loading for these sources was not known, one was estimated based on other known sources. The estimated values were from 50,000 to 100,000 Btu.

Transient Sources

Transient ignition sources are only of concern if they are exposed to a critical loading of transient combustibles. The probability of having an unsuppressed critical loading of a transient combustible exposed in the compartment, P_{tc}, can be estimated as follows:[4-2]

$$P_{tc} = P_{fst} * u * p * x/2 * \ln(1/x)$$

where:
- P_{fst} = probability of non-suppression
- u = critical floor area ratio
- p = probability of the critical combustible being exposed
- x = F_{ccl}/F_w
- F_{ccl} = the frequency of having a critical combustible loading present
- F_w = frequency of inspections for combustibles

The PNPP "Control of Transient Combustibles" procedure, PAP-1913, severely restricts the unattended storage of transient ignition sources and combustible loads in the generating facility. In general, transient combustible material quantities are restricted, are controlled by permit, are not taken into the plant until required to support the work activity and are promptly removed when no longer required. Flammable and combustible liquids are only used in well ventilated areas using only U.L. listed or F.M. approved containers if not in the original manufacturer's container. Smoking is not permitted in areas where flammable and combustible liquids and gases are used, stored or handled. The storage of flammable/combustible materials adjacent to safety related areas is prohibited unless permitted by USAR Section 9A, or are administratively controlled by PAP-1913.

Given the strict rules pertaining to transient combustibles, it is highly unlikely that large amounts of debris or liquid combustible would be present for any duration of time. However, smaller amounts of

maintenance refuse, e.g., cleaning rags, protective clothing (anti-contamination), may not be uncommon in some safety related areas. Therefore, the transient type analyzed for PNPP will consist of moderate amounts of Class A/B mixed combustibles, i.e., paper, oily rags, polyethylene bottles, etc.

A review of heat release test data for fires involving transient material packages selected to characterize typical nuclear power plant transients has been performed by EPRI.[4-3] The fire types were as follows:

Fire Zone Type	Typical Fire	Worst Case Fire Size
Frequently occupied by plant personnel	Human Occupancy: Polyethylene bag, paper cups, towels	310 Btu/s
Only occupied for maintenance, inspection and occasional operational reasons	Maintenance Refuse: Polyethylene wash bottles, buckets, cardboard, Kimwipes, small amounts of acetone	140 Btu/s
RRA fire zone where used protective clothing (anti-contamination) may be temporarily stored	Protective Clothing: Protective clothing stored in bins or polyethylene bags	107 Btu/s

The total combustible loading associated with each of these representative fires was approximately 100,000 Btu.

The type of transient fire, e.g., human occupancy trash, must be determined for each fire compartment on a case by case basis. In general, human occupancy trash fires are consistent with normally occupied or heavily trafficked areas, e.g., control room, offices, etc. However, none of the fire compartments analyzed in this detailed analysis, with the exception of the control room which is discussed separately, were of the type expected to contain human occupancy trash, so only maintenance refuse or protective clothing transients were modeled. Since the maintenance refuse and protective clothing fires are relatively close in size, the former was conservatively chosen to represent both. A bounding 150 Btu/s HRR was chosen to encompass all transient combustible fires.

Based on the transient fire type determined for the compartment, critical vertical and horizontal separation distances were developed for all targets in the room and used to determine the critical floor area ratio, u. A transient combustible exposure probability, p, of 0.1 was selected based on the guidance provided in FIVE. A Critical Combustible Loading Frequency, F_{ccci}, of 2/compartment/year was applied. This value was judged to be conservative based on a review of the Periodic Fire inspection Tracking log coupled with discussions with plant personnel. Finally, all plant areas are inspected for transient combustibles once per week, giving an a value of 52 for F_w. Given these values, P_{tc} is equal to $P_{fst} * u * 6.266x10^{-3}$, where P_{fst} and u are specific to individual fire areas/compartment.

Welding Sources

Hot work sources include work involving welding, brazing, open flame, soldering, grinding and thermal cutting processes. Any activity producing sufficient heat or having the potential for producing sparks or hot material which could cause combustible materials to ignite in the generating facility. Given the stringent administrative control governing hot work at PNPP defined in PAP-1912, it is not probable for a welding fire to go undetected for any great period of time. Both the welder and the fire watch would be available to detect and extinguish any fires related to "hot work" prior to escalation beyond an incipient stage. This is supported by the Fire Events Database[4-6] which shows that most welding fires were quickly extinguished. Hence, timely manual suppression was generally considered feasible for both welding cable fire and welding/cutting fire ignition sources.

Lastly, since both welding/cutting fires and transient fires involve the ignition of transient combustibles, they were analyzed together. That is, welding/cutting fires will be treated as just another transient ignition source, albeit a readily extinguishable one.

4.5 Evaluation of Component Fragilities and Failure Modes

Potential targets are separated into three main categories:

- cables serving safety related equipment
- electrical equipment
- intervening combustibles

4.5.1 Cables Serving Safe Shutdown Equipment

This category is limited to cable serving equipment modeled in the PNPP PRA, i.e., cable that if lost would directly effect the value of P2. All other cable is categorized as an intervening combustible since it may indirectly lead to loss of safe shutdown cable. All cable at PNPP is IEEE qualified cable. The threshold damage temperature was assumed to be 700°F for cables subjected directly to the plume, ceiling jet or hot gas layer, and 1 Btu/s/ft^2 for cables subjected to radiant heat.[4-2, 4-3]

4.5.2 Electrical Equipment

For electrical cabinets in general, the threshold damage temperature was assumed to be 248°F when subjected directly to the plume, ceiling jet or hot gas layer, and 0.88 Btu/s/ft^2 when subjected to radiant heat.[4-3] For cabinets containing temperature sensitive electrical equipment, e.g., solid state relays, the threshold damage temperature was assumed to be 150°F when subjected directly to the plume, ceiling jet or hot gas layer, and 0.19 Btu/s/ft^2 when subjected to radiant heat.[4-3]

Electric motors are assumed to fail at temperatures approximately 20 percent over their operating limit.[4-3] Since only some electric motors are targets, i.e., many electric motors are associated with non-safe shutdown components, and each motor may be qualified for a different temperature, they were analyzed on a case-by-case basis.

4.5.3 Intervening Combustibles

The main source of intervening combustible is cable insulation; both safe shutdown and non-safe shutdown. In this case, the most critical parameters are the piloted and spontaneous ignition temperatures. Any other intervening combustible noted during the walkdowns was dealt with on a case-by-case basis. The piloted ignition temperature of IEEE qualified cable is assumed to be 932°F.[4-3] Furthermore, the time to piloted ignition of IEEE qualified cable is on the order of 5 to 10 minutes of direct flame impingement.[4-3] The spontaneous ignition temperature is conservatively chosen to be equal to the piloted ignition temperature.

4.5.4 Characterization of Secondary Fire Sources

Overhead cable trays represent the major source of exposed combustible material that may ignite and become secondary fire sources. The size of this secondary fire source depends upon the number and size of the cable trays exposed to the primary fire source, and the extent of fire spread. Based on actual tests, fire growth in a cable tray stack is primarily in the vertical direction; however, horizontal spread does occur as the fire progresses from one level to another.

Where required, the effects of secondary ignition of overhead cable trays was modeled in accordance with Appendix K of the EPRI Fire PRA Implementation Guidelines.[4-3]

4.6 Fire Detection and Suppression

Unless the source and target are one and the same, there is the potential to extinguish the fire prior to damage to the target occurs. Targets which are also sources are assumed to be damaged upon ignition. Fire suppression can be automatic, e.g., wet pipe sprinkler, or Halon system, or manual. In either case, some form of detection must precede suppression. The following sub-sections describe how both automatic and manual fire detection and suppression was incorporated into this analysis.

4.6.1 Automatic Suppression

Credit was taken for automatic fire suppression systems (AFSS) if it could be shown that the critical equipment in a fire area/compartment would not be damaged prior to successful suppression. In some cases, engineering judgment was used to supply justification for credit of an AFSS. For example, AFSS credit can be given for a small to medium size fire, with respect to room size, well separated from the target if the AFSS is activated by a smoke detector installed on the ceiling. In some cases, more detailed analyses were performed. This involved determining the time to detector response as described in Worksheet A-1 of FIVE, adding any actuation time delay, e.g., most total flooding CO_2 and Halon systems have a time delay and warning signal to allow personnel time to evacuate the compartment, and comparing this to the damage time. If the time to damage is greater than the sum of the detection time and the actuation time delay, credit for the AFSS can be given.

For cases where AFSS can be credited, a suitable automatic suppression failure rate, P_{fsta} was assigned. Automatic fire suppression system failure rates obtained from previous studies are published in of Attachment 10.3 of FIVE, and are as follows:

Halon	0.05
Deluge or Pre-Action Sprinklers	0.05
Wet Pipe Sprinkler Systems	0.02
CO_2	0.04

For sources where the use of the AFSS was not justified based on inadequate response time or coverage, the P_{fsta} was assigned a value of one.

4.6.2 Manual Suppression

As with automatic suppression, credit was only taken for manual fire suppression if it could be shown that the time to damage is greater than the combination of the times to detection and suppression. The time to suppression includes the detection time and the time required for fire fighters to assemble at the scene and extinguish the fire. Detection means can be either manual or from automatic detectors.

A fire watch equipped with a portable fire extinguisher is required during, and 30 minutes after completion of any work involving open flames, welding, grinding or temperatures that would exceed the heat of ignition of materials in contact with that work to ensure the work area is safe from fire danger. All welding and cutting fires in the Fire Events Database were detected and extinguished within a few minutes by either the fire watch or the welder. This implies that welding fires detected in the incipient stage are easily extinguished by the fire watch. Given the above information, manual suppression was considered possible, with an associated failure probability of 0.1, for all welding fires.

In all other cases, detection by automatic means, and suppression by the fire brigade was required. The time to automatic detection is calculated as described in Section 4.6.1. In these cases, manual suppression was credited if the time to damage is greater than the sum of the time to detection and the fire brigade response time to the affected fire compartment. The fire brigade response times was estimated based on those times recorded during unannounced fire drills.

If the drill response time is less than the detection-damage interval, the probability of manual suppression failure, P_{fstm}, was assigned a value of 0.1. This assumes that if the fire brigade arrives prior to damage, 90 percent of the fires will be extinguished prior to further damage. If the detection-damage interval falls in the range of drill response times, Pf_{stm} was assigned a value of 0.5, i.e., the fire brigade has a 50 percent chance of arriving prior to damage. If the drill response time is greater than the detection-damage interval, manual suppression was not deemed credible.

4.7 Analysis of Plant System Sequences and Plant Response

A representative detailed fire scenario analysis (Fire Compartment 1CC3a) employed for general plant areas is presented in Section 4.7.1. The control room analysis, which uses a somewhat unique approach, is discussed in Section 4.7.2. Section 4.7.3 presents a review of the results obtained for all plant areas/compartments considered in the detailed analysis.

4.7.1 Detailed Analysis for Unit 1, Division 2 Switchgear Room

4.7.1.1 General Description of Area

Fire Compartment 1CC3a is the Unit 1, Division 2 Switchgear Room. It is located at elevation 620'-6" along the north wall of the control complex. It houses the Unit 1 Division 2 4.16 kV and 480 V Class 1E switchgear, the Division 2 remote shutdown panel, and the Class 1E 480 V MCCs for power distribution to Unit 1 Division 2 safety-related equipment. Fire detection equipment for the compartment consists of ionization detectors that activate alarms in the control room. Manual carbon dioxide hose reels, water type hose stations and fire extinguishers are provided for fire suppression. A simplified sketch showing the layout of the room is given as Figure 4-1.

4.7.1.2 Ignition Source and Screening Results Summary

The fixed fire sources in this compartment include electrical cabinets, junction boxes, an RPS MG set, and transformers. Transient sources, including welding/cutting fires, are credible as well. The total ignition frequency for these sources, F1, was determined in the screening analysis to be 7.994×10^{-3}, and was distributed as follows:

Electrical Cabinets	3.750×10^{-3}
Junction Box in Qualified Cable	2.554×10^{-5}
RPS MG Sets	2.750×10^{-3}
Transients	1.887×10^{-4}
Transformers	7.802×10^{-4}
Welding/Cutting Fires	5.000×10^{-4}
Total	7.994×10^{-3}

The conditional core damage probability, P2, given loss of all components in this fire area is 7.01×10^{-3}. The product of P2 and F1 for this fire compartment results in an initial estimated fire induced core damage frequency, F2, of 5.604×10^{-5}/yr. Since this is greater than prescribed screening criteria of 10^{-6}/yr, further analysis was warranted.

4.7.1.3 Fire Scenario Evaluation

Fire scenarios associated with each of the ignition sources are considered in turn in the following subsections.

Electrical Cabinets

The electrical cabinets include 4.16 kV switchgear associated with bus EH-12, 480 V buses EF-1-C and EF-1-D, 480 V MCCs EF1C07, EF1C08, EF1C09, EF1D07, EF1D08 and EF1D09 and a power supply cabinet for post LOCA shutdown, M51-S002. All cabinets, except the post LOCA shutdown power supply cabinet, which is a single cabinet, are constructed from a series of vertical cabinets bolted together at the sides. For clarity, each of these vertical cabinets will be referred to as a bay. The total area cabinet fire frequency $F1_C$ must be divided between the electrical cabinets. However, since some are larger than others, an equal split is not warranted. Therefore, a weighted fire frequency, wF1, was calculated based on the fraction of the total number of sections associated with each electrical cabinet, f, as shown below:

Electrical Cabinet Weighted Fire Frequencies (wF1)

Cabinet EID	Cabinet Description	No. of bays in Cabinet	Panel Fraction (f)	Weighted Fire Freq (wF1)
EH-12 (R22-S006)	4.16kV SWGR	18	0.225	3.069×10^{-4}
EF-1-C/EF-1-D	480 V Bus	7	0.088	1.194×10^{-4}
EF1C07 (R24-S023)	480 V MCC	14	0.175	2.387×10^{-4}
EF1C08 (R24-S024)	480 V MCC	4	0.050	6.820×10^{-5}
EF1C09 (R24-S025)	480 V MCC	7	0.088	1.194×10^{-4}
EF1D07 (R24-S026)	480 V MCC	19	0.238	3.240×10^{-4}
EF1D08 (R24-S028)	480 V MCC	5	0.063	8.525×10^{-5}
EF1D09 (R24-S036)	480 V MCC	5	0.063	8.525×10^{-5}
M51-S002	Post LOCA Power Supply	1	0.013	1.705×10^{-5}

4.16kV switchgear and 480V Bus cabinet fires

The 4.16 kV switchgear and 480 V buses have a combined combustible content of approximately 31 MBtu. Given the similarity in section size of these electrical cabinets, it was assumed the 31 MBtu can be equally distributed, i.e., each cabinet section contains about 1.25 MBtu of combustible. Furthermore, since each of the cabinets is ventilated and/or has open penetrations, in accordance with Section 4.4.3 of this report, a screening HRR of 400 Btu/s was applied.

Various simple fire modeling calculations were performed using the FIVE worksheets to determine potential target damage and time to damage. The results were subsequently used to construct fire scenarios with and without timely fire suppression.

All cable within the plume (directly above the cabinet) are subject to immediate damage if their elevation is less than about 635'-6", i.e., about 7'-7" above the switchgear or Bus. All cable within the plume are subject to ignition if their elevation is less than about 634'-2", i.e., about 6'-3" above the switchgear or Bus.

For cable targets not in the plume or ceiling jet but exposed to the hot gas layer (HGL), the time to damage from the 400 Btu/s fire source alone, i.e., ignoring additional HRR from burning overhead cable, is about 35 minutes.

Furthermore, the ceiling jet interface is at about elevation 635'-9", i.e., about 7'-10" above the switchgear or Bus. For cable targets in the ceiling jet, a minimum of 2' from the plume, the time to damage is over 17 minutes.

If the HGL were to substantially descend below the source, the potential for damage to temperature sensitive electrical equipment (TSEE), and other electrical equipment exists.

Damage to TSEE, which is limiting with respect to other electrical equipment, will occur in approximately 15 minutes.

With respect to the increased HGL temperature rise from burning overhead cable, damage is not expected to occur any earlier than the 15 minutes predicted above based on the following. The time to piloted ignition of IEEE qualified cable is on the order of 5 to 10 minutes of direct flame impingement.[4-3] Some amount of time is also required for both fire growth of the ignition source, and growth of the secondary cable fire. Plots of HRR vs. time of large cabinet fires from tests at Sandia plotted in Attachment 10.3 of FIVE, suggest that several minutes are required for the cabinet fire to reach high HRRs. This same growth period is anticipated for the overhead cable as well. Thus, the sum of the ignition source fire growth period, the 5 to 10 minutes required to ignite the cable and the additional cable fire growth period is probably on the order of 10 to 15 minutes. Furthermore, if the HGL were to truly descend below the elevation of the fires, the burning efficiency would likely be reduced as oxygen is replaced by the products of combustion. And lastly, heat loss factors from test data were on the order of 0.79 to 0.93 at 2.5 minutes, 0.94 to 0.98 at 5 minutes, and were increasing at test termination. Hence, the 0.7 and 0.8 heat loss factors used in Tables A-2 and A-4 are conservative values. Therefore, it is judged that no damage to TSEE will occur until at least 15 minutes into the scenario.

Damage to TSEE is also possible from radiant exposure. Radiant exposure is only assumed credible for non-shielded source-target pairs. Hence, for a 4.16 kV switchgear fire, only adjacent 4.16 kV cabinet sections and the 480 V Bus are exposed. Likewise for a 480 V Bus fire, only adjacent 480 V cabinet sections and the 4.16 kV switchgear are exposed.

The critical radiant heat flux distance was determined to be less than 8'-3". However, the 480 V Bus and 4.16 kV switchgear are perpendicularly separated by about 8'-6". Hence, radiant damage to the 480 V bus from the 4.16 kV switchgear is not predicted, and vice versa.

If the fire can be promptly extinguished, only the source, adjacent cabinet sections and targets in the plume are assumed damaged. There is no automatic suppression in this fire compartment. The limiting time for manual suppression, T_{crit}, is 15 minutes based on the time to damage of TSEE. If the time to detector actuation, T_d, plus the fire brigade response time, T_r, is less than T_{crit}, manual suppression can be

credited. The detector response time was evaluated using FIVE worksheets as described in the following paragraph.

The maximum distance from any 4.16 kV switchgear or 480 V Bus section to the nearest smoke detector is 30 feet. Based on the methods described in FIVE, the estimated time to smoke detector response, T_d, is on the order of 16 seconds. This calculation was based on the predicted target temperature rise, a detection device rated temperature rise (38°F), and detector time constant of 10 seconds recommended by FIVE. Therefore, manual suppression can be credited if the fire brigade response time is less than about 15 minutes.

Fire brigade response times for Fire Compartment 1CC3a are on the order of 10 minutes. Therefore, T_{crit} is greater than $T_d + T_r$, and manual suppression can be credited. In accordance with Section 4.6.2, a probability of manual suppression failure, P_{fstm}, of 0.1 was applied.

Given the above information, numerous fire scenarios exist involving the 4.16 kV switchgear and 480 V Bus. However they can be categorized into two main types, namely; those where manual suppression is successful, and those where it is not. When manual suppression is successful, damage is still credible for the source, all targets in the plume, and all targets in the ceiling jet within two feet of the source cabinet. If manual suppression fails, all components in the room are assumed to fail.

Based on this rationale, eight potential fire damage states and associated frequencies were defined for the 4.16 kV switchgear. Similarly seven scenarios were defined for the 480 V Bus. These scenarios together with their contribution to fire induced CDF are described in Table 4-4.

<u>Other Electrical Cabinets</u>

The other electrical cabinets include the MCCs and the post LOCA power supply cabinet, M51-S002. As previously stated, the MCCs are constructed of several vertical cabinets bolted together to form an elongated vertical cabinet. Each vertical cabinet, or section, consists of several stacked cubicles which are open to one another. A 65 Btu/s fire source was used in this case.

There is a total of 24,960,000 Btu in the Fire Compartment 1CC3a MCCs. Assuming each of the 54 MCC bays is equally loaded, and has a mass burnout fraction of 0.7, the total heat released from any one MCC cabinet section is approximately 325,000 Btu. Some initial screening calculations were performed on these fires as described below.

A 65 Btu/s MCC fire will not produce a hot gas layer sufficient to damage overhead cable. The ceiling jet interface is at about elevation 636'-2". Also, for cables outside the plume, but in the ceiling jet, no damage is expected unless the conduit is within 7 inches of the MCC. This assumes that the MCC is against the wall thereby increasing the effective heat release rate. Next, by using the same source height and heat loss fraction used for the 4.16 kV switchgear and 480 V Bus, it was shown that TSEE is not susceptible to damage from a hot gas layer formed from an MCC fire which descends to near the floor.

Finally, the critical radiant heat flux distances to electrical equipment and TSEE from any 65 Btu/s source are about 1'-6" and 3'-4", respectively. Note that the two closest MCCs, i.e., electrical equipment, EF1D07 and EF1D08, are about 2 feet apart. Also, the closest distance between the 4.16 kV switchgear or

480 V Bus, i.e., TSEE, and any MCC is about 4 feet. Therefore, the only concern for MCC fires are targets in the plume.

With respect to the post LOCA shutdown power supply, an estimate of about 15 pounds of combustible material, plastics and cable insulation, was made based on visual inspection during the walkdown resulting in a combustible content of about 150,000 Btu. Assuming a 0.7 mass burnout ratio, a total heat release of about 100,000 Btu is expected. Furthermore, as with the MCC, the small size would indicate that a heat release rate of 65 Btu/s is credible. Since M51-S002 is smaller than an MCC, all damage ranges are bounded by that of the MCC except for the ceiling jet damage range.

The ceiling jet temperature is a function of the effective heat release rate, and since this cabinet is in a corner, its effective heat release rate is higher than that of any MCC. Damage is not expected for cables in the ceiling jet at least 6 inches away from M51-S002. Note that although the effective heat release rate for M51-S002 is greater than that for an MCC fire due to the location factor, the horizontal separation distance from the ceiling jet is actually 1 inch less. This is due to the fact that the post LOCA shutdown panel is a much shorter cabinet than the MCCs, and therefore, the distance between the source and ceiling is greater for M51-S002.

It was noted during the walkdown that the MCCs were constructed with no openings between adjacent individual bays, and cable bundles were generally vertical. Therefore, based on review of cabinet fire tests [Appendix H of Ref. 4-3], fire spread between open top MCC sections is not likely, and was assumed to not occur. It was also noted during the walkdown that the conduit penetrations in three of the six MCCs, EF1C07, EF1C08 and EF1D09, were sealed.

Since each MCC panel is constructed of several cubicles filled with electrical equipment and cable, very little free volume is available. Therefore, a fire in an MCC section would likely self extinguish due to oxygen starvation before propagation to an adjacent section. Therefore, cabinet fire propagation for the sealed MCCs was assumed to not occur.

As stated above, MCCs EF1C07, EF1C08 and EF1D09 are sealed. Therefore, damage is limited to the loss of the safe shutdown equipment served by these MCCs. These scenarios are included Table 4-4.

EF1C09 is an open top MCC located in the center of the room. Two conduits, 1R22C205X and 1R33C5520X, pass over the MCC, but do not contain safe shutdown cable. Three trays would be in the plume of a fire from this MCC. Trays 1356 and 1649 contain cable for safe shutdown equipment and would be damaged. Simple FIVE fire modeling demonstrates that tray 1649 would be the only tray with undamaged cable. Piloted ignition of only tray 273 is likely, i.e., plume temperature rise at target is much greater than 932°F. However, since cable in trays 1356 and 273 would be damaged, both Methods A and B would fail; hence, MCC EF1C09 is not screened. This scenario (# 17) is included in Table 4-4.

EF1D07 and EF1D08 are open top MCCs located against the south wall of the room. Several conduits pass over or near these MCCs, but do not contain safe shutdown cable. For a fire in either MCC, three trays would always be in the plume, 271, 1321 and 1320, all of which contain cable for Methods A and B. These trays have vertical separation distances of 1'-11", 3'-3" and 4'-7", respectively, from either MCC. Also, depending on which panel the fire occurred in, an additional vertical tray (1371, 1372, 1373, 1374, 1375, 1376, 1377, 1378, 1384 or 1385) carrying Method B cable would be in the plume.

Simple FIVE fire modeling demonstrates that cable in all of the trays would be damaged and trays 271 and 1321 are likely to ignite, thereby damaging safe shutdown equipment. Therefore, neither MCC EF1D07 nor EF1D08 was screened. These scenarios (#'s 18 and 19) are included in Table 4-4.

M51-002 is an open top cabinet located near the northwest corner of the room. Several conduits pass over or near this cabinet, but only 1R33C5423B, which is vertically separated from M51-002 by 7'-0", contains safe shutdown cable. For a fire in this cabinet, five trays would be in the plume, 363, 1605, 1809, 1317 and 272. These trays have vertical separation distances of 3'-8", 3'-8", 5'-0", 6'-4" and 7'-6", respectively. Trays 1317 and 272 contain safe shutdown cable, the others contain do not.

Simple FIVE fire modeling demonstrates that all cable with at least a 6'-8" vertical separation distance would not be damaged. Trays 363, 1605 and 1809 would most likely ignite, thereby probably damaging tray 272 and conduit 1R33C5423B as well. Therefore, since safe shutdown equipment would be damaged, M51-S002 was not screened. This scenario (# 20) is included in Table 4-4.

Junction Boxes

Although "Junction Box in Qualified Cable" is listed as a potential ignition source as part of the FIVE Phase II, Step 1 evaluation, it is not considered a significant risk as discussed in Section 4.4.3.

RPS MG Sets

An RPS MG Set is contained in a totally enclosed room within this fire area. The room has an approximate area of 170 ft^2. There is very little combustible material in this room, only a small amount of cable, two small transformers and the MG Set. The MG Set is lubricated by a small amount of grease. Also, the plant walkdown confirmed that there was no concentration of combustibles near the boundary and no combustible pathway exists between the MG Set room and the remainder of the fire area.

The total combustible content in the RPS MG Set room is estimated to be less than 3 MBtu. Therefore, the combustible loading for the room is less than about 18,000 Btu/ft^2. Since this room has automatic smoke detection, the combustible loading is less than 20,000 Btu/ft^2, and there is no concentration of combustibles near the boundary or combustible pathway across the boundary, the MG Set room was treated like a separate fire compartment.

A fire in the MG Set room would at most cause a reactor scram trip. Since no shutdown components would be disabled, the RPS MG Set and the transformers in the room (see Transformers subsection below) were screened from the analysis.

Transformers

There are eight transformers in the fire area. Two are large 4,160/480 V transformers, EHF-1-C (R23-S011) and EHF-1-D (R23-S012), and six are small (less than about 50 KVA) transformers, R25-S034, R25-S5002, R25-S027, R25-S035, R25-S128 and R71-S076. All of the transformers are vented near or at the top, and are dry type. The transformer fire frequency, $F1_{TR}$, was equally divided among all eight transformers (9.753×10^{-5}/transformer year).

R25-S034 and R25-S5002, are contained in the MG Set room, and as discussed above are screened from the analysis.

R23-S011 and R23-S012 are large transformers with a fairly large amount of combustible in the form of cable insulation. Much of the insulation is in the interior windings and is not accessible; however, these sources will be assumed equal to the 400 Btu/s cabinet sources. Therefore, the 480 V Bus cabinet sections they are attached to would be considered damaged, i.e., adjacent cabinet sections are assumed damaged. Due to the separation distance of at least 8'-6", radiative damage to the 4.16 kV switchgear is not predicted.

Both transformers are within the 30 foot range used to calculate smoke detector response. Therefore, in the same way as employed for the large electrical cabinets, a P_{ms} of 0.1 was applied for these large transformer fires. Recall that even if suppression is successful, damage is still credible for the source, adjacent cabinet sections, all targets in the plume, and all targets in the ceiling jet within two feet of the source transformer. If manual suppression fails, all components in the room are assumed to fail. The R23-S011 and R23-S012 fire scenarios can be characterized as follows.

R23-S011 is the large 4,160/480 V transformer attached to the west end of, and is the normal power supply to, 480 V bus EF-1-C. A fire in this transformer would result in a loss of power to EF-1-C if the EF-1-C to EF-1-D cross-tie breaker could not be closed. Since a section of EF-1-C is adjacent to R23-S011, a ground fault was assumed to occur to the bus, and therefore, re-energization of the bus by closing the cross-tie breaker is not credible. Targets in the plume or ceiling jet of a potential fire in R23-S011 include cables in trays 272, 366, 1317 and 1608 and six conduits. Trays 272, 1317 and 1608 contain shutdown cable, whereas tray 366 does not. The conduits include 1C95R157B, 1C95R158B, 1R22C204X, 1R25B266X, 1R33B27X and 1R33R1438B; none contain cable serving shutdown equipment.

As in the case of cabinet fires, transformer fires are categorized as suppressed and non-suppressed. The resultant two R23-S011 fire scenarios (#'s 25 and 26) are listed in Table 4-4.

R23-S012 is the large 4,160/480 V transformer attached to the east end of, and is the normal power supply to, 480 V bus EF-1-D. A fire in this transformer would result in a loss of power to EF-1-D if the EF-1-C to EF-1-D cross-tie breaker could not be closed. Since a section of EF-1-D is adjacent to R23-S012, a ground fault was assumed to occur to the bus, and therefore, re-energization of the bus by closing the cross-tie breaker is not credible.

Targets in the plume or ceiling jet of a potential fire in R23-S012 include cables in trays 272, 366, 597, 1317, 1608 and 1649 and six conduits. All but tray 366 contains cable associated with safe shut down equipment. The conduits include 1R33C1219X, 1R33C1220X, 1R33C1221X, 1R33C1222X, 1R33R1338X and 1R33R1369C; none contain safe shutdown cables.

The resultant two R23-S012 fire scenarios (#'s 27 and 28) are listed in Table 4-4.

The remaining transformers are all relatively small transformers with minimal combustible material. They are assumed to contain about 200,000 Btu of burnable combustible (based on comparison of other combustibles in the room - 100,000 Btu for M51-S002, and 325,000 Btu/MCC cabinet), and are expected

to burn with a HRR of about 65 Btu/s. Therefore, they are bounded by the criteria of the other 65 Btu/s sources, i.e., only targets in the plume are of concern.

R25-S027 is a located in the center of the room. Several conduits pass near this transformer, but do not contain safe shutdown cable. This source is, therefore, screened out as having little or no impact on the fire induced core damage frequency for 1CC3a.

R25-S035 is located against the north wall of the room between EF1C07 and EF1C08. R25-S035 is no closer than 2 feet to either MCC. Since the critical radiant heat flux distance for electrical equipment is about 1'-6", no radiant damage is expected to either MCC from a potential fire in R25-S035. There are no conduits passing near this transformer. The closest trays in the plume are 370 and 1318 which have respective vertical separation distances of 4'-2" and 5'-6".

Simple FIVE fire modeling demonstrates that cables in tray 1318 are not subject to damage. Those in tray 370 may be damaged, but would not ignite, i.e., plume temperature rise at target is less than 932°F - T_{amb}. However, tray 370 contains no shutdown cables and the source can therefore be screened out.

Transformers R25-S128 and R71-S076 are located against the north wall of the room more than 5 feet from the 480 V bus. Since the critical radiant heat flux distance for sensitive electrical equipment is about 3'-4", no radiant damage is expected to the 480 V Bus from a potential fire either transformer. There are no conduits passing near these transformers containing shutdown cables. The only tray in the plume of either transformer is 597 which does contain shutdown cable. Tray 597 has a vertical separation distance of 6'-2" above both transformers.

Simple FIVE fire modeling shows that cables in tray 597 are not subject to damage. Therefore, since no safe shutdown components are damaged, R25-S128 and R71-S076 can be screened from further analysis.

Transients and Welding/Cutting Fires

In accordance with Section 4.4.3, transient combustibles left in the area are likely to be maintenance refuse with a combustible loading of about 100,000 Btu, and a HRR of 150 Btu/s. Using the FIVE worksheets, critical cable heights of 6'-4", 8'-0" and 10'-3" were calculated for transient fires in the center of a room, against a wall, and in a corner, respectively. Out-of-plume damage only occurs inches from the source. Direct radiant damage was found to be possible when the source was within 2'-2" of cable.

With respect to electrical equipment, not even TSEE, damaged at 150°F, would be subject to damage from the HGL formed by a transient fire. However, direct radiant damage was found to be possible when the transient source was within 2'-4" of electrical equipment, and within 5'-0" of TSEE.

Also, since there is no automatic suppression in this fire compartment, P_{fst}, was set to 1.0 for transient fires, i.e., no automatic or manual suppression, and 0.1 for welding/cutting fires (see Section 4.6.2). Finally, based on the previously calculated critical vertical and horizontal separation distances discussed above, the critical floor area ratio, u, was conservatively calculated to be 0.46.

Inserting the values listed above for P_{fst} and u into the equation for P_{tc} (see Section 4.4.3), a probability of 2.882×10^{-3} was calculated for P_{tc} for transients, and 2.882×10^{-4} for welding/cutting fires. The product of F1 and the sum of P_{tc} for both transient and welding/cutting fires, 6.876×10^{-7} ($1.887 \times 10^{-4} \times 2.882 \times 10^{-3} + 5.000 \times 10^{-4} \times 2.882 \times 10^{-4}$) is the probability of having an unsuppressed critical loading of a transient combustible exposed to a transient or welding/cutting ignition source in the compartment. For the purposes of this screening analysis, these unsuppressed transient fires are assumed to damage all equipment in the room. The transient and welding/cutting fire scenario (# 33) is listed in Table 4-4.

4.7.1.4 Determination of Core Damage Frequency

In order to determine the effects of each fire scenario, it was necessary to determine which safe shutdown equipment would be lost due to fire damage to the associated individual components cable trays and conduits.

The worst case impact due to fires in this area was determined during the screening analysis which evaluated the CCDP given all equipment disabled as 7.01×10^{-3}. In this case the dominant core damage sequence involved successful injection followed by random failure of long-term heat removal (via RHR or venting). Consequently, emphasis was placed on identifying the precise level of damage to the heat removal functions (-W and -Y) in each scenario. Table 4-4 summarizes the results of this investigation in terms of the functions served by individual target cable trays and components. As a result four unique damage states were identified and assigned to particular fire scenarios:

- Fire Damage State 0 is identical to the worst case condition identified in the screening and results in loss of RHR B heat removal via suppression pool cooling and containment spray, as well as containment venting paths via FPCC, MSIVs and containment spray train B. This may occur due to fires which occur in the 4.16 kV switchgear, non suppressed electrical fires which disable specific overhead trays (with no opportunity for suppression) or electrical fires with failure of timely manual suppression which result in room temperatures beyond the limits for sensitive electrical equipment.

- Fire Damage State 1 results in loss of RHR B heat removal via suppression pool cooling and containment spray, as well as containment venting via MSIV paths B and D. However, venting via both containment spray paths A, FPCC and MSIV paths A and C remain operable. This may occur due to fires in MCCs EF1C09, EF1D07 or EF1D08 which damage trays directly overhead but is suppressed prior to causing excessive room temperatures.

- Fire Damage State 2 results in loss of RHR B heat removal via suppression pool cooling and containment spray, as well as venting via FPCC and all MSIV paths. However, venting via both containment spray paths A and B remain operable. This may occur due to fires in 480 V Bus EF-1-C which damage trays directly overhead but is suppressed prior to causing excessive room temperatures.

- Fire Damage State 3 does not result in significant damage.

The total core damage frequency for all 33 Fire Area 1CC3a fire scenarios is 1.06×10^{-5}/yr.

4.7.2 Control Room Analysis

4.7.2.1 General Approach

The general philosophy for fire evaluation of control room fires follows the approach suggested in NSAC 181[4-4] and EPRI document RP 3385-01.[4-3] It is similar to that adopted in other areas but differs in two respects.

1. Detailed fire propagation analysis was not performed since there are no acceptable models for modeling propagation within and from cabinets. Instead various assumptions were made, supported by the results of the Sandia cabinet fire tests[4-10] in which all tested fires self-extinguished, and by the reports of control room fires in the Fire Events Data Base.[4-6] The following general assumptions were adopted:

    - No inter-cabinet fire spread is assumed since individual cabinets are separated by double steel walls.

    - Fire spread between cabinet bays, which are separated by a single thickness steel wall, may occur. However, the delay time for propagation (> 15 minutes) is such that control room evacuation would have occurred at or about the same time. Therefore, inter-bay fire propagation is generally not a significant concern from a risk perspective.

    - No damage to equipment adjacent cabinets is assumed due to the intervening double steel walls.

    - Fire damage to equipment in adjacent cabinet bays may occur if the fire is not extinguished in a timely manner.

2. Regardless of the level of damage which is actually sustained as result of a fire, the production of smoke may necessitate the evacuation of the Control Room. Under such circumstances the operators will isolate the Control Room and shutdown the plant using the Division 1 Remote Shutdown Panel (RSP). Re-entry into the control room is credited for the operation of long term heat removal functions (providing that the associated controls are not damaged) which are not required for several hours.

The evaluation of control room fires required analysis to determine those cabinets or combination of cabinet bays in which enclosed fires might cause significant degradation of accident mitigating systems. Fire scenarios in such cabinets were evaluated individually. Fires in the remaining cabinets were evaluated as a group for their potential to cause the operators to evacuate the Control Room and shutdown the plant using the Remote Shutdown Panel, or locally operated equipment.

The methods for frequency analysis, propagation analysis and suppression analysis are discussed in the following sections.

4.7.2.2 Description of Area and Associated Fire Protection

The general layout of the Control Room is shown in Figure 4-2.

The Control Room (Fire Area 1CC5a) is situated on the fifth level of the control complex at elevation 654'-6" and is separated from other adjacent fire areas by three-hour rated fire barriers. All penetrations have 3 hour-rated seals. The combustible loading in the Control Room is approximately 171,818 Btu/ft^2.

Safe shutdown equipment located in this fire area consists of consoles and control panels. All circuits enter and leave the control room via termination cabinets located against the north and south walls of the control room. Cables are routed between various panels via cable ducts located in the steel plate raised floor sections. Redundant divisions of cabling are routed in separate cable ducts and generally in separate bays of the termination cabinets.

The control room HVAC system, M25, and the Control Room Emergency Recirculation System, M26, operate in conjunction with each other using shared ducts and fans. In the smoke clear mode, the M25 supply and exhaust fans operate with a flow rate of 30,000 cfm equivalent to a purge rate is about 5 air changes per hour.

Fire detection in the Control Room consists of heat detectors and ionization smoke detectors covering the floor section modules and ionization detectors covering the control room proper. There is no automatic fire suppression system installed in the Control Room. Fire suppression equipment consists of manually activated carbon dioxide total flooding system for the cable ducts in the sub-floor sections. The sub-floor is divided into three main sections and should a fire occur in may one of these the ducts, all the ducts in the entire section are flooded simultaneously. Manual water type hose stations and water, Halon, dry chemical and carbon dioxide type fire extinguishers are also provided for back-up suppression.

In the event of a fire in this area, normal control of all accident mitigating systems could be lost or operators could be forced to evacuate the Control Room. In such cases operators would preferably use the Division 1 Remote Shutdown Panel (RSP) located in Fire Area 1CC3d to shutdown the plant from outside the main control room. Control functions are isolated from the main control room and control power supplies are duplicated/protected as necessary such that a fault in the main control room will not interfere with control from the RSP. The back-up Division 2 RSP, in conjunction with individual breaker switches in the Division 2 switchgear room, may also be utilized if necessary. However, the same degree of control power protection is not provided in this case. In such cases operators would follow integrated operating instruction IOI-11.[4-4]

4.7.2.3 Fire Hazard Review

The fire frequency in the Control Room was evaluated in the screening analysis (FIVE Phase II Step 2) as 1.033×10^{-2}/yr, i.e., about one control room fire for every 100 years of reactor operation. This fire frequency is comprised of the following contributions:

Control room location frequency	9.500×10^{-3}/yr
Plant Wide Sources	8.300×10^{-4}/yr

In fact, the Fire Events Data Base[4-6] on page 3-17 indicates that plant wide component based fires should not be applicable to the Control Room since no control room fires were considered in plant-wide bins. The plant wide sources include transient combustibles, welding cable fires and welding/cutting fires. The PNPP FIVE analysis is, therefore, overly conservative and a fire frequency of 9.5×10^{-3}/yr is more appropriate.

The Control Room fire frequency is based on twelve fires which actually occurred in control rooms, eleven of which were cabinet fires and one of which was a kitchen fire. None of the fires have been of significant severity and all were extinguished or self extinguished within a few minutes. No control room fires to date have required evacuation of the Control Room.

Potential sources of fire in the PNPP control room include the control panels, termination cabinets, electrical junction boxes, and transient sources/combustibles.

In the control room, transient fires do not pose a significant risk because it is continuously occupied and the likelihood that a transient fire would not be detected and suppressed in its incipient stage is very small. Furthermore, since there are no exposed fixed combustible materials in the room there is no possibility of a growing transient induced fire. Transient fires are therefore discounted.

Fires induced by electrical termination faults located in the termination cabinets or junction boxes are not considered significant fire sources given the low operating currents and over-current protection devices. Even if a small fire were to develop in the termination cabinets or the under floor ducts the fire would not propagate due to the lack of a fresh air supply and the use of IEEE 383 cable. A comprehensive fire analysis of the raised floor sections,[4-12] including full scale fire tests, found that a ventilated fire initiated in one section, using simulated transient material, will neither propagate through the installed combustible material (cable insulation) nor will any damage to cable in an adjacent floor section occur. Thus in the highly unlikely event that a fire did occur, damage would be confined to one electrical division.

Thus, the only potentially significant control room fires postulated are those which occur in control panels. This is consistent with the approach adopted in NSAC 181.[4-4]

4.7.2.4 Fire Induced Equipment Failures for Enclosed Cabinet Fires

Fire-induced equipment failures in a cabinet were evaluated as described in the following subsections.

Determine Critical Cabinets

Determine critical cabinets whose fire-induced failure would degrade safety related equipment required for hot shutdown.

Information derived from control room design drawings which associate one or more electrical divisions with each cabinet, supplemented with additional cable tracing efforts were utilized to define or bound the potential system loss due to a fire in each control room cabinet. Cabinets serving Offsite Power supplies, RCIC, HPCS, ADS valves and the FPCC valves and MSIVs which could be used for containment venting were specifically identified. Other system failures were assumed if the particular cabinet under consideration contained associated divisional control equipment, e.g., LPCS was assumed to fail due to a fire in any cabinet designated as Division 1.

All control room fires were assumed to result in a permanent loss of main feedwater/condensate and other balance of plant systems.

As a first step in identifying significant fire locations, cabinets have been grouped according to the potential fire impact on shutdown capability, as shown in Table 4-5.

The specific cabinets which fall into each group are listed in Table 4-6.

Determine the likelihood of a fire occurring in a critical cabinet

The frequency of fire in each individual cabinet is evaluated as a function of the relative size of each cabinet and the overall cabinet fire frequency for the entire control room. This approach attempts to account for the relative number of potential ignition sources in cabinets of different size.

Each of the cabinets (see Figure 4-2) are divided into separate bays, each bay being of approximately similar size, with the exception of the main console. Consequently, the relative size of each cabinet can be measured in terms of the number of associated bays.

Therefore, for example, the frequency of fire for the Emergency Core Cooling Bench Board (P601) is calculated as follows:

$$= \text{F1 (control room)} \times \frac{\text{No. of bays in P601}}{\text{Total No. of bays in all cabinets (183)}}$$

$$= 9.500\text{x}10^{-3}/\text{yr} \times 7/183 = 3.634\text{x}10^{-4}/\text{yr}$$

The number of bays in each cabinet and the associated fire frequency is shown in Table 4-6.

Determine Fire Severity to Fail Critical Functions

Determine how severe a fire would have to be to fail the critical functions supported by a particular cabinet bay or combination of connected cabinet bays.

The equipment located within a PNPP control room cabinet is separated from adjacent cabinets by double steel walls. The cabinets themselves are subdivided into separate bays. In many cases a single bay will only contain equipment associated with one electrical division. However, in cases where this separation is not maintained, conductors associated with different divisions are separated according to IEEE 384 separation criteria.

The only ignition sources present within the electrical cabinets occur due to electrical faults. If the damage can be confined locally to the site of the overload, which is in fact the most likely situation given historical experience with control room fires, i.e., the faulted component or associated wiring, the resulting impact will be bounded by the random failure of the component itself, which has already been accounted for in the internal events PRA model. Consistent with the Sandia cabinet fire tests, it was assumed that a fire in one of the control room cabinets will generally not impact equipment in another (see discussion above). However, a fire in one bay of a cabinet, if not extinguished in a timely manner, may damage components in adjacent bays. All equipment in the cabinet bay in which the fire originates, or propagates to, was assumed to fail immediately.

It therefore remains to determine what opportunity exists for extinguishing a fire before it can develop and damage components in an adjacent bay. The evidence from the Sandia cabinet fire tests can be used to establish the time required for the progression of fire damage. These tests indicate that between 8.5 and 11.5 minutes, i.e., about 10 minutes, elapsed between smoke first being observed coming from a cabinet initiation and significant heat generation, 10 to 20 kW. This is termed the pre-growth fire development phase. The tests also indicate that once fire growth begins it may progress rapidly, as may the rise in cabinet air temperature.

Ionization fire detectors located in the PNPP control room ceiling will detect the fire shortly after the initial smoke emission. Thus, allowing 3 minutes for the detectors to actuate, credit will be given for preventing inter-bay fire damage propagation within the first 8 minutes after fire detection. This is termed the pre-growth phase. Once the fire damage has propagated to other cabinets bays all functions associated with those bays are assumed to fail.

Since cabinet bays are separated by steel walls, fire propagation (as opposed to damage propagation) beyond the original bay is expected to progress slowly and until this occurs no damage is expected in bays beyond the bays directly adjacent to the bay in which the fire originates. Therefore, the fire duration required for damage to occur beyond directly adjacent bays is expected to be considerably longer than 10 minutes. Such damage is considered to be bounded by the control room evacuation scenarios. Control room evacuation, according to the modeling assumptions, is required in all cases, within 20 minutes, if manual suppression has not been implemented. At this time all equipment served by the entire cabinet in which the fire occurs is assumed to be permanently disabled unless alternate controls are available on the Division 1 Remote Shutdown Panel.

Determine the Probability of Critical Component Damage within a Cabinet or Adjoining Cabinets

The probability and extent of component damage is dependent upon the probability of non-suppression during the pre-growth phase discussed above. The probability of non-suppression of control room fires as a function of time was obtained using a model to interpret the control room fire durations in the EPRI data base. Such a model is developed in Appendix J of the EPRI Fire PRA Implementation Guide.[4-3] The probability of non-suppression derived from the model (Case 1 is recommended) is:

$$p(\text{non-suppression within 7 minutes}) = 4.9\text{x}10^{-2}$$

4.7.2.5 Adverse Effects of Smoke

Determine the Level of Smoke which Will Impair the Effectiveness of the Operators

The Sandia cabinet fire tests[4-10] indicate that fires were self-sustaining and did produce sufficient quantities of smoke to cause visual impairment with purge rates as high as 14 room air changes per hour. All of the actual control room fires reported in the Fire Events Data Base (FEDB) were small, but this may have been because they were extinguished early. Since there are no tools available for assessing smoke production and the evidence from the historical fires is not conclusive, it was assumed that any fire is capable of producing sufficient smoke given that it is allowed to continue burning for a sufficient period of time.

Determine Time Available to Suppress Fire

Determine how much time is available to suppress the fire before the smoke concentration reaches the level of visual impairment, at which time the operators are assumed to evacuate the Control Room. There are eleven Sandia tests for which information is available for smoke build up.

Six tests were performed in a small enclosure, 11,016 ft^3 with ventilation rates about 14 room air changes per hour. However, only one of these was electrically initiated (Test PCT5) and indicated visual obscuration within 13 minutes with time zero being the point at which smoke was first observed from the cabinet. Five tests were performed in larger enclosures of 48,000 ft^3, two of which were electrically initiated. In both of the electrically initiated, large enclosure tests, the control room control board was obscured within 20 minutes after smoke was first observed based on visual observations. The ventilation rate in one case was 1 room air change per hour, and in the other, 8 room changes per hour. For the large enclosure tests, the ventilation system did not appear to substantially effect the rate of smoke build up.

The volume of the PNPP Unit 1/Unit 2 control room envelope is approximately 367,000 ft^3. However, the effective volume of the Unit 1 control room is less than half, i.e., about 100,000 ft^3 (compared with 48,000 ft^3 in the SNL large enclosure tests).

The normal number of air changes per hour is one . However, this may be increased to 5 changes per hour in the smoke purge mode.

Based on the above discussion it is concluded that the rate of smoke build up in the PNPP control room will not be any faster than that observed for large test enclosure. That is smoke obscuration of the control board will not occur for at least 20 minutes after smoke begins to escape from the cabinet. Again allowing

3 minutes to activate one of the area ionization detectors, 17 minutes would be available to extinguish the fire prior the necessity for control room evacuation.

Ten minutes was selected by Sandia and used in the NUREG-1150 fire studies for Peach Bottom and Surry. EPRI selected a time interval of 15 minutes for their NSAC 181 control room analyses.

Determine the Probability of Detection and Suppression Prior to Visual Impairment

Using the HCR approach to determine probability of non-suppression within the time frames available prior to the control boards being obscured, the probability is determined to be:

$$p(\text{non-suppression within 17 minutes}) = 2.0\times10^{-3}$$

4.7.2.6 Screening of Potentially Significant Cabinet Fires

Based on the review performed in Section 4.7.2.4, the control cabinets were grouped according to the immediate impact of fires on accident mitigating systems. An initial screening of cabinets was performed assuming that a fire originating in any bay of a given cabinet destroys the entire cabinet, i.e., inter-bay fire propagation was assumed to be 1.0.

The frequency of fires associated with each cabinet group was evaluated in Table 4-6. The conditional core damage probability given the damage associated with each group was evaluated using the internal events PRA model as discussed in Section 4.7.2.8. The screening core damage frequency results are summarized in Table 4-7.

It was concluded that the contribution from fire scenarios confined to cabinets associated with groups 1a, 1b, 1c, 1d, 2a, 2c, 3c, 4a, and 4b do not present a significant fire risk regardless of the extent of inter-bay propagation. However, fires in groups 2b, 3a and 3b cabinets may be significant and therefore inter-bay propagation scenarios were addressed for those cases. Fire growth in all cabinets leading control room evacuation was also addressed.

4.7.2.7 Examination of Non-Screened Cabinet Fire Scenarios

The various growth stages, probabilities and fire damage states were defined and evaluated for each cabinet utilizing a fire growth event tree approach similar to that shown in the example provided as Figure 4-3. The end points of each event tree represents a particular fire damage state and associated frequency. Eighteen unique damage states were identified and are summarized in Table 4-8.

Fire scenarios originating in Group 2b cabinets

Group 2b consists of one cabinet, P870, which is the bench board located on the south side of the horseshoe. This cabinet consists of 9 bays only two of which, A and B, contain any equipment credited for shutdown. Bays C to J include controls for the neutral division and are assumed to result in a loss of feedwater and the Power Conversion System (PCS) only.

A fire in Bay A may lead to a loss of offsite power. If the fire is not extinguished prior to growth, i.e., within 8 minutes, equipment in adjacent Bay B, is assumed to also fail, resulting in a loss of Division 2

control. If the fire is not extinguished within a further 9 minutes, control room evacuation is assumed to be required. Similarly a fire in Bay B may lead immediately to a loss of control of Division 2 systems, and damage may propagate to Bay A if the fire is not extinguished in a timely manner. Fires in Bay C, although not significant immediately, may propagate to Bay B.

Fire scenarios originating in Group 3a Cabinets

Group 3a consists of 18 cabinets which were initially identified as potentially giving rise to a loss of control of Division 1 and 2 systems from the control room in the event of a fire. However, on closer inspection the function of equipment associated with many of these cabinets, although safety related, is not required for safe plant shutdown. The list of Group 3a cabinets finally identified as containing safe shutdown equipment was as follows:

P680	Unit 1 Control Console
P691	Division 1 RPS and Instrument Auxiliary Relay Panel
P692	Division 2 RPS and Instrument Auxiliary Relay Panel
P693	Division 3 RPS and Instrument Auxiliary Relay Panel
P694	Division 4 RPS and Instrument Auxiliary Relay Panel
P800	HVAC Control Panel
P868	Analog Loop Div 2 Instruments
P877	Diesel Generator Bench Board
P904	Common HVAC
P623	Auxiliary Relay Panel
P869	Divisional Instruments

P680 Unit 1 Control Console

Although this panel is designated as containing safe shutdown equipment designated Division 1 and Division 2, only the RPS system would be affected. Since this system is fail safe with respect to fire damage and there are other means of initiating reactor scram which do not rely on this equipment, the impact of fires which are confined to this console is negligible. The fire growth event tree for this cabinet therefore addresses fire suppression prior to control room evacuation only.

P691 Division 1 RPS and Instrument Auxiliary Relay Panel
P692 Division 2 RPS and Instrument Auxiliary Relay Panel
P693 Division 3 RPS and Instrument Auxiliary Relay Panel
P694 Division 4 RPS and Instrument Auxiliary Relay Panel

These cabinets contains two bays both of which are assumed to result in an immediate loss of control of Division 1 and Division 2 systems in the event of a fire. The fire growth event tree for these cabinets, therefore, addresses fire suppression prior to control room evacuation only.

P800 HVAC Control Panel

This cabinet contains 5 bays. Bays A and E contain neutral division equipment only. Bay D contains Division 2 designated equipment only. Bay C contains Division 1 designated only. Bay B contains both Division 1 and Division 2 designated equipment.

P868 Analog Loop Div 2 Instruments

This cabinet contains 3 bays. Bay A and Bay B includes Division 1 and Division 2 designated equipment. Bay C contains Division 2 designated equipment only.

P877 Diesel Generator Bench Board

This cabinet contains 2 bays. Bay A contains Division 1 designated equipment. Bay B contains Division 2 designated equipment.

P904 Common HVAC

This cabinet contains 3 bays. Bay A contains Division 2 designated equipment. Bay B contains Division 1 designated equipment. Bay C controls neutral division equipment.

P623 Auxiliary Relay Panel

This is a single bay cabinet containing Division 1 and Division 2 designated equipment. A fire in this cabinet is assumed to result in an immediate loss of control of all Division 1 and Division 2 equipment.

P869 Divisional Instruments

This cabinet contains 3 bays. Bay A and Bay B contain Division 1 and Division 2 designated equipment. Bay C contains Division 1 designated equipment.

Fire scenarios originating in Group 3b cabinets

Group 3b consists of one cabinet, P601, which is the Emergency Core Cooling benchboard located on the north side of the horseshoe. This cabinet consists of 7 bays. Bay A contains equipment designated Division 1. This bay was also determined to contain equipment associated with HPCS. Bays B and C contain equipment designated Division 2. Bays D and E contain Division 2 designated equipment. Bay F contains Div 1 and 2 designated equipment. Bay G contains neutral division equipment and is assumed to have no immediate impact on shutdown systems other than the loss of feedwater and PCS.

4.7.2.8 Fire Scenario Quantification

The quantification of the PRA model for control room fire sequences was performed in an identical manner to that adopted for the quantitative screening analysis.

The contributions to the frequency of each damage state from each fire scenario, including the cabinet groups screened earlier are summarized and totaled in Table 4-9.

The conditional core damage probability associated with each damage state is evaluated by requantifying the internal events model accounting for the equipment disabled due to fire as well as any pertinent recovery actions. In performing this analysis, the following assumptions were followed:

- The fire induced initiating events modeled were loss of offsite power (Type T1 event tree) or transient with a loss of PCS (Type T2 event tree).

- Human error probabilities for short term operator actions (within the first hour) were increased by a factor of 5 to account for the increased stress and distraction arising from the control room fire and the need, in some cases, to operate equipment from the Division 1 RSP.

- In the event of damage to ADS Division 1 and 2 equipment or the need control room evacuation RCS depressurization and low pressure injection is not feasible since only 3 SRVs can be controlled from outside the control room. A minimum of four valves was assumed to be required for success.

- In the event of control room evacuation, operators may re-enter the control room or use local switches in order to establish long term heat removal providing the specific fire damage to control room circuits does not prevent this. Division 1 suppression pool cooling is always operable from the Division 1 RSP, regardless of control room damage. However, the Division 1 containment spray and all Division 2 systems may not be available to function.

The product of the aggregated fire damage state frequencies and the corresponding CCDPs give the fire induced core damage frequencies for each state shown in Table 4-9.

4.7.3 Results

The results of the detailed fire analysis for the seventeen fire area/compartments which did not screen during the screening phases of FIVE are summarized in Table 4-10. Following the detailed analysis, the fire induced core damage frequencies for seven of the areas/compartments remain above the screening criteria of 10^{-6}/yr. These are:

1CC5a	Unit 1 Control Room	$1.06x10^{-5}$	33.9
1CC3a	Unit 1 Division 2 Switchgear Room	$1.05x10^{-5}$	33.5
1TPC/1	Unit 1 Turbine Power Complex - Switchgear Room	$3.38x10^{-6}$	10.5
CC1	Control Complex Elevation 599'	$2.03x10^{-6}$	6.5
1CC3c	Unit 1 Division 1 Switchgear Room	$1.98x10^{-6}$	6.3
FH3	Fuel Handling Building Elevation 620'	$1.63x10^{-6}$	5.2
1TB	Unit 1 Turbine Building	$1.30x10^{-6}$	4.2

Fire Zone 1CC5a, the Unit 1 Control Room and Fire Zone 1CC3a, the Unit 1 Division 2 Switchgear Room, are the most significant contributors. The top fire induced core damage sequences are summarized in the following subsections.

The contributions from the Unit 1 Turbine Power Complex, the Fuel Handling Building and the Unit 1 Turbine Building essentially represent screening values. Additional detailed modeling would most certainly reduce their contribution below 10^{-6}/yr.

4.7.3.1 Fire Zone 1CC5a - Unit 1 Control Room

The contributions to core damage frequency arising from fires in the Unit 1 Control Room are evaluated in Table 4-8. The fire induced core damage frequency, F3, for the Unit 1 Control Room is $1.06x10^{-5}$/yr. The most significant fire scenarios are described below.

<u>Fire in Any Control Room Panel causing Control Room Evacuation</u>

A fire in any Unit 1 Control Room panel which is not suppressed prior to the need for control room evacuation is assumed to result in an initial loss of control for all systems from the Unit 1 Control Room. Operators attempt to shut down using RCIC from the Division 1 Remote Shutdown Panel. Operators may subsequently use Train A, or Train B in some cases, of suppression pool cooling. Control would be from the remote shutdown panel or by re-entering the control room after extinguishing the fire. The dominant fire damage scenarios (FDS) involve random failure of the RCIC system. The CDF is $5.66x10^{-6}$/yr for FDS 3, FDS 6, FDS 17 and FDS 18 for a fire involving control room evacuation.

Fire in Panel P601

A fire in Panel P601, Bays C, D, E or G, which is not suppressed prior to damage being sustained in the adjacent bays or a fire in Bay F which is suppressed is assumed the result in damage to Division 1 and Division 2 systems. These systems could include RCIC, ADS and those components associated with containment venting via the fuel pool cooling and cleanup system. The operators attempt to shut down using HPCS (controlled from the Panel P601, Bay A) or RCIC (controlled from the remote shutdown panel and Train A of RHR in suppression pool cooling (also controlled form the remote shutdown panel). Containment venting via the MSIVs could be used in conjunction with HPCS. These systems fail due to random faults resulting in a CDF of 1.18×10^{-6}/yr for FDS 7.

Fire in Panel 625 or P873 or in Bays A or B of P601

A fire in Group 1b single bay Panels P625 or P873 or confined to Bays A or B of Panel P601 is assumed to result in the loss of HPCS and all Division 1 systems except RCIC. Operators attempt to shutdown using RCIC or ADS and low pressure systems associated with Train B of RHR. However, these systems fail due to random faults. The fire induced CDF is 7.08×10^{-7}/yr from FDS 13.

Fire in Bays B or C of Panel P870 or in Bays A, B or C of Panel P871

A fire in Panel P870 confined to Bays B or C or in Panel P871 in Bays A, B or C which is suppressed prior to control room evacuation is assumed to result in damage to Division 2 systems. These include the valves used for containment venting via the fuel pool cooling and cleanup system. Operators attempt to shutdown using either RCIC or HPCS and Train A of RHR in suppression pool cooling or containment spray modes of operation. However, these systems fail due to random faults principally from maintenance unavailability and human interaction difficulties. The CDF is 6.24×10^{-7}/yr for FDS 15.

Fire in Bays D or E of Panel P800, Bay C of Panel P868, Bay B of Panel P877, Bay A of Panel P904 or in Group 2a Panels

A Fire in Bays D or E of Panel P800, Bay C of Panel P868, Bay B of Panel P877, Bay A of Panel P904 or in Group 2a Panels is assumed to lead to the loss of all Division 2 systems. This excludes the RCIC isolation valve. Operators attempt to shutdown using RCIC and Train B of ADS and low pressure injection systems. However, these systems fail due to random faults. The CDF is 6.04×10^{-7}/yr for FDS 14.

4.7.3.2 Fire Zone 1CC3a - Unit 1 Division 2 Switchgear Room

The risk is dominated principally by fires in the 4.16kV switchgear and 480 V bus sections which damage overhead cable. Such fires lead to a permanent loss of RHR Train B as well as the loss of containment venting paths via the fuel pool cooling and cleanup system, Train B of containment spray, and MSIVs (FDS 0). Random failures of RHR Train A subsequently lead to core damage. This is somewhat conservative for two reasons:

- Overhead cable trays located in the plume of the fire are assumed to be damaged with a probability of 1.0. No credit was taken for fires being extinguished prior to such damage (either from self or manual extinguishment).

- All fires are assumed to lead to a permanent loss of equipment. In reality, it is likely that buses may be re-energized once the fire has been extinguished and the damaged sections repaired or isolated.

Other less important contributors include fires in the 480 V MCCs and the transformers which damage overhead cable trays.

4.8 Containment Performance

The evaluation of containment performance following core damage resulting from fires requires the consideration of mechanisms which may lead to containment bypass via hi-lo interfaces, failure of containment isolation or degradation of the availability of heat removal systems. NUREG-1407, Section 4.1.4 indicates that the focus should be on identifying containment failure mechanisms which are significantly different from those found in the internal events IPE.

4.8.1 Containment Bypass

Containment bypass pathways are identified and evaluated in the IPE. Likewise, the Appendix R Safe Shutdown Capability Report, Section 2.2, identifies and dispositions such pathways with respect to the potential for fire damage to initiate LOCAs via these paths. Fire induced mechanical failures are not considered credible. Spurious valve operation due to control and power circuit damage caused by fires was examined, as discussed in Section 4.2.2. No significant fire induced bypass mechanisms were identified.

4.8.2 Containment Isolation

Containment isolation failure was also considered in the IPE, the principal mechanism being associated with isolation valve failure to close. Containment penetration pathways determined to be a potential concern in the IPE analysis are isolated by mechanical check valves, air operated valves which fail closed on loss of air supply or power to their solenoid valves or electrically operated (a.c./d.c.) valves. Fire damage will not impact the operation of check valves and is likely to lead to actuation of containment isolation in the case of air operated valves. In the event of fire induced damage which does not prevent closure of air or electrically operated containment isolation valves, manual recovery actions may be taken. Since fire induced accidents do not lead to vessel failures, several hours would be available to take such action. Furthermore, the most risk significant fire scenarios are located in the switchgear room and the control room which are well away from the containment penetration areas where the isolation valves are located. Thus, access to the containment isolation valves would not be impaired due to the effects of the fire. The likelihood of failure of containment isolation following fire scenarios is not significant and no new mechanisms of isolation failure were identified.

4.8.3 Containment Pressure Suppression and Heat Removal

The availability of containment heat removal systems including RHR and various containment vent paths following core damage was explicitly modeled in the PRA event trees, and their status reflected in the plant damage state binning process The contribution of the dominant Unit 1 Control Room and Unit 1 Division 2 Switchgear Room fire scenarios involving loss of these systems is as follows:

1CC5a	Control Room	2.2×10^{-6}
1CC3a	Division 2 Switchgear Room	1.0×10^{-5}

4.8.4 Containment Performance Summary

In conclusion, containment performance issues have been specifically addressed in the fire analysis. No vulnerabilities which could cause early failures of containment or containment bypass were identified. The frequency of core damage scenarios involving loss of containment heat removal systems is estimated to be 1.2×10^{-5}/yr. This estimate is dominated by the Unit 1 Division 2 Switchgear Room fire scenarios, which as discussed above, have certain inherent conservatisms particularly when considering the potential for recovery in the time frame for initiation of long term containment heat removal (on the order of 15 to 20 hours).

4.9 Treatment of Fire Risk Scoping Study Issues

4.9.1 Background

Under the USNCR-sponsored Fire Protection Research Program, Sandia National Laboratories developed the Fire Risk Scoping Study: Current Perception of Unaddressed Fire Risk Issues (NUREG/CR-5088), hereafter referred to as the "FRSS." The objectives of this study were to:

- Reassess certain fire risk scenarios, in light of the availability of enhanced fire event databases and improved fire modeling techniques.
- Identify significant fire risk issues that may not have been addressed adequately, or at all, under earlier fire risk assessments, and to attempt to quantify the impact of these issues.
- Review current regulatory criteria and guidance, and plant fire protection programs, to assess whether the identified risk scenarios are adequately enveloped by these programs.

The issues identified and addressed by the FRSS include six categories:

- Potential seismic/fire interactions.
- Fire barrier qualification issues.
- Manual fire fighting effectiveness.
- Total environment equipment survival.
- Potential control systems interactions.
- Improved analytical codes.

The above issues, which were not addressed by earlier "fire" probabilistic risk assessments (PRAs), are required to be assessed as an integral part of the Individual Plant Examination for External Events (IPEEE). A structured approach to addressing the first five of these issues is presented in the "Fire-Induced Vulnerability Evaluation (FIVE)," EPRI Report TR-100370. The FIVE report provides an overall methodology for addressing the "fire" portion of the IPEEE process; the FRSS issues are but one element of the IPEEE process.

The sixth FRSS issue, concerning analytical codes, does not require a plant-specific evaluation or response, as the use of current-day analytical codes, i.e., COMPBRN IIIe, is incorporated as an integral part of the FIVE, Phase II, methodology. Accordingly, this analysis is limited to a Perry Nuclear Power Plant (PNPP) specific assessment of only the first five issues.

4.9.2 Seismic/Fire Interactions

This issue involves three concerns:

- The potential for seismically-induced fires.
- The potential for seismically-induced actuation of fire suppression systems.
- The potential for seismically-induced failure or rupture of fire suppression systems.

The above events have obvious implications on both postulated fire scenarios and potential for disruption of the safe-shutdown capability.

4.9.2.1 Seismically Induced Fires

This issue considers the potential leakage or rupture of flammable or combustible liquid gas lines or tanks/containers during a seismic event, which could create fire hazards. The potential hazards to be addressed include:

- Hydrogen piping
- Diesel fuel oil piping, day tanks, and storage tanks.
- Turbine lubricating oil storage tank(s) and associated piping.
- Turbine generator (hydrogen envelope).
- Hydrogen seal oil unit and associated piping and tanks.
- Hydrazine storage tanks and associated piping.

The specific location of these and similar hazards are identified through the Fire Walkdown Phase of the IPEEE process, and the seismic ruggedness of each identified component was addressed under the Seismic Walkdown Phase. No potential vulnerabilities were identified. See Section 3.1.6.

4.9.2.2 Seismic Actuation of Fire Suppression Systems

This issue considers the potential for inadvertent actuation of suppression systems during a seismic event, and the resultant effects on safety/safe-shutdown related components and systems. The effects of concern include both flooding and wetting effects caused by runoff/spray.

Automatic fire suppression systems located in fire areas/zones containing safety/safe-shutdown related equipment include the following:

Fire Area/Zone	Suppression System Type
1AB1c	Automatic Sprinkler System
1AB3b	Automatic Sprinkler System
CC1a	Automatic Sprinkler System
CC1b	Automatic Sprinkler System
CC1c	Automatic Sprinkler System
CC2a	Automatic Sprinkler System
CC2b	Automatic Sprinkler System
CC2c	Automatic Sprinkler System
1CC4a	Automatic Sprinkler System
1CC4b	Automatic Sprinkler System
1CC4e	Automatic Sprinkler System
1CC4f	Automatic Sprinkler System
1DG1a	Automatic CO_2 System
1DG1b	Automatic CO_2 System
1DG1c	Automatic CO_2 System
IB2	Automatic Sprinkler System
IB3	Automatic Sprinkler System

The cables installed at PNPP are capable of withstanding water spray and/or immersion hence, no problems should arise with the operation of a water spray system in areas containing only redundant cables. All other sprinkler areas containing safe shutdown components contain only one train of equipment.

Automatic fire suppression systems installed within a fire area containing a single train of shutdown components and cables are designed to activate and operate within one fire area only. Therefore, an inadvertent activation of the fire suppression system will only effect one component or one train of cables, and as in the case of a single fire, redundant divisions are not affected by this single event.

In addition, these water type fire suppression systems are designed to operate where the fire is located; this is accomplished by the use of closed heads operated by a fusible link. Therefore, wholesale operation is averted. In the event of a pipe rupture, floor drains are provided to adequately drain water away from affected equipment without jeopardizing the redundant train.

By PNPP internal memoranda, a review of the IEN 83-41 issues (Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment) was documented. This review concluded the concerns raised in the Information Notice fell into the categories of design, sealing, shielding and drainage. The concerns related to design were considered to have already been adequately addressed. Furthermore, with the implementation of actions pertaining to sealing and to shielding and drainage, the remaining concerns were adequately addressed. Consequently, this review, and the resolution of the subsequent commitments, is considered to adequately envelope the issue of seismically-induced actuation of PNPP automatic fire suppression systems.

4.9.2.3 Seismic Degradation of Fire Suppression Systems

This issue addresses the seismic installation of suppression system piping and appurtenances, and the potential for seismically-induced mechanical failure of these systems. The issue is focused on the potential effects on the safe-shutdown capability caused by suppression system equipment dislodging during a seismic event, and falling onto the subject equipment.

The location of fire suppression piping with respect to safe-shutdown equipment, and the potential effects, from the perspective of possible impact of equipment falling onto safe-shutdown components, is addressed under the Seismic Walkdown Phase of the IPEEE. No potential vulnerabilities were identified. See Section 3.1.6 for further discussion.

The potential for an earthquake to result in a fire was evaluated as part of the seismic walkdown effort. One area of concern was identified with the Fire Service Fuel Oil Tank located in the ESW Pumphouse Diesel Fire Pump Room. The tank was evaluated as having a HCLP of 0.3g. See Section 3.1.6 for further discussion.

4.9.3 Fire Barrier Qualifications

This issue is primarily concerned with the installation and maintenance of fire barriers and fire barrier penetration seals, including electrical and mechanical seals, as well as fire doors and fire dampers.

4.9.3.1 Fire Barriers

Fire barriers, defined as "any fire-rated wall or radiant heat shield protecting safety-related equipment and cable," are inspected at least once per 18 months, or every refueling outage, in accordance with PTI P54-P0054. Appendix R cable wrap is inspected every 18 months, or every refueling outage in accordance with PTI P54-P0075.

Any barriers found unacceptable are immediately reported to the Control Room Unit Supervisor, who shall take appropriate actions as defined in PAP-1914. Fire barriers which do not meet the inspection criteria may be deemed acceptable if a Fire Barrier Removal Permit is active on the barrier. PAP-1914 causes a Fire Barrier Removal Permit to be issued on any fire barrier removal, and ensures appropriate compensatory measures are taken to protect the area.

4.9.3.2 Fire Doors

The fire door maintenance and inspection program consists of inspections, verifications and operability tests. Periodic Test Instruction PTI P54-P0040 describes the semi-annual operability check of magnetically held open fire doors. PTI P54-P0041 describes the semi-annual fire door inspection. Daily fire door verifications are performed in accordance with PTI P54-P0044. These tests are all performed in accordance with the requirements of PAP-1923, NFPA-80 and Section 9.5.1.4 of the USAR.

4.9.3.3 Penetration Seal Assemblies

Penetration Seal Inspection and Surveillance Program

The surveillance of fire barrier penetration seals is performed in accordance with PTI P54-P0056 which provides a visual inspection of 10 percent of the penetration seals every 18 months, or every refueling outage. An inspection schedule is maintained to ensure every penetration is inspected at least once every 15 years.

Evaluation and Implementation of Applicable NRC I&E Notices

The FIVE methodology identifies three NRC I&E Information Notices (IEN) which have specific applicability to fire barrier penetration seals:

- IEN 88-04, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals."
- IEN 88-04 Supplement 1, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals."
- IEN 88-56 Supplement 1, "Potential Problems with Silicon Foam Fire Barrier Penetration Seals."

Evaluations were made of the specifications and the procedures for fire barrier penetrations related to IEN 88-04. These evaluations created concerns related to weaknesses in the implementation of USNRC requirements and guidelines. In response to these concerns, a penetration seal design and installation program is being implemented at PNPP including:

- A definition of design documents showing the function of the seal
- a definition of seal design criteria
- a review of present seal installations
- a definition of the installer qualification requirements

This is considered to adequately address the concerns regarding qualification of fire barrier penetration seals at PNPP.

4.9.3.4 Fire Dampers

Fire Damper Inspection and Maintenance Program

Fire dampers are visually inspected in accordance with Periodic Test Instructions PTI P54-P0053A through R once every 18 months, or every refueling outage.

In addition, following repair or maintenance of an HVAC system fire damper, both a static and dynamic functional test are performed on the damper to verify operability in accordance with PTI P54-P0022. Failed damper testing is a violation of the Approved Fire Protection Program as described in PAP-1914, and must either be reported or declared inoperable and formally impaired with approved compensatory actions per PAP-1914, Attachment 4, Section 6.D.

Evaluation and Implementation of Applicable NRC I&E Notices

The FIVE methodology identifies two NRC I&E Information Notices which have specific applicability to fire dampers:

- IEN 89-52, "Potential Fire Damper Operational Problems"
- IEN 83-69, "Improperly Installed Fire Dampers at Nuclear Power Plants"

With respect to IEN 89-52 deficient dampers were identified affecting PNPP Unit 1. Worst case samples were tested and modifications were made to ensure closure of all required dampers under system operating conditions. PNPP documentation states that "This issue was an NRC open item (50-440/85002 EE) and was satisfactorily closed by the NRC in their Inspection Report 50-440/85090."

By PNPP internal memoranda, a review of the IEN 83-69 issues were documented. These reviews concluded that many of the PNPP fire dampers were not in conformance with NFPA-90A at the time of the review. In response to these reviews, the fire damper non-conformities were resolved and the issue closed.

4.9.4 Manual Fire Fighting Effectiveness

This issue is focused on the adequacy of training and preparedness of the plant fire brigade, and on the general orientation of appropriate plant personnel to fire response requirements. The objective of this issue is to determine the adequacy of the plant's manual fire suppression capability, and thereby determine the degree to which this capability should be credited in the IPEEE Fire PRA.

4.9.4.1 Reporting Fires

Orientation of Plant Personnel to the Use of Portable Fire Extinguishers

As described in the Fire Protection Training Program, PAP-1917, a program is in place to train selected plant personnel in the use of portable fire extinguishers. Personnel formally trained include the Fire Brigade members, First Responders and Fire Watches.

In accordance with PAP-1911, individuals discovering a fire are allowed to attempt to extinguish an incipient fire only if trained to do so, and if the attempt will not jeopardize their own safety or the safety of others.

Availability of Portable Extinguishers Throughout the Plant

Portable fire extinguishers are located in all fire zones except where radiological concerns exist, e.g., under condenser. For fire areas where radiological concerns exist, extinguishers are staged outside the area for fire brigade use. The location of fire extinguishers for every fire zone is shown in the FPI Procedure, i.e., Pre-Fire Plan Instructions, for that zone. Operability of the fire extinguishers is ensured by periodic testing and maintenance as described in PAP-1914.

Plant Procedure for Reporting Fires

The initial reporting of fires and subsequent notification and dispatching of a First Responder and the Fire Brigade, is addressed by Plant Administrative Procedure PAP-1911. Furthermore, the reporting of fires by assigned Fire Watches is addressed in PAP-1916. Lastly, as part of the PNPP General Employees Training Program, the Fire Protection Training for General Employees trains all personnel with unescorted access the proper procedures for reporting a fire.

Communication System to Allow Contact With the Control Room

As specified in PAP-1911, any individual discovering a fire is required to promptly notify the Secondary Alarm Station (SAS) or Control Room by the nearest communication system. PNPP contains an internal phone system with extensions located throughout the plant.

4.9.4.2 Fire Brigade

Size of Fire Brigade

As stipulated in PAP-1910, a fire brigade of at least five members, including the fire brigade leader, is maintained on site at all times.

Brigade Members Knowledgeable in Plant Systems and Operations

The plant Fire Brigade will consist of a Fire Brigade Leader with either a reactor operators license, or equivalent knowledge of plant safety related systems, i.e., simulator certification with plant systems familiarity, and four individuals from the Operations Section on all shifts. One of these individuals shall be the Fire Attack Leader assigned prior to the start of the shift.

Annual Physical Examinations for Brigade Members

In accordance with PAP-1917, brigade members must satisfactorily complete a medical physical examination, including an electrocardiogram and a respiratory examination, and a sub-maximal stress test prior to participation in the Initial Fire Brigade Training Course, and thereafter, on an annual basis as part of the Fire Brigade requalification requirements.

Minimum Equipment Provided/Available to Fire Brigade

In accordance with Safety/Fire Instruction SFI-0060, all fire equipment in place in the Equipment Storage Areas and Fire Brigade Stations is visually and functionally inspected in full on a monthly basis, and in part when equipment is used for drills, practices, or emergencies to verify the equipment is in satisfactory condition for emergency use.

The turnout gear includes helmets, hoods, coats, pants, boots and gloves as described in PAP-1919, and flashlights. Portable ventilation equipment is provided in the Vent Paks which include smoke ejection equipment and accessories mounted on a portable cart. Portable fire extinguishers available to the Fire Brigade include the foam cart, as well as local portable fire extinguishers. Although not stated in SFI-0060, PAP-1911 states that the Fire Brigade Leader should have access to two portable radios. Therefore, emergency communications equipment available to the Fire Brigade should include both the PNPP internal communications system, and portable radios. In accordance with PAP-1922, the type and location of communication equipment within and nearby each fire zone, including an indication of any possible communication problem such as high noise levels, or poor radio communication, is listed in each Pre-Fire Plan Instructions. Therefore, the equipment available to the fire brigade is consistent with the FRSS criteria.

4.9.4.3 Fire Brigade Training

Initial Classroom Instruction Program

The fire brigade classroom training program, as described in PAP-1917, provides the following elements, consistent with the FIVE evaluation methodology:

- Indoctrination in the plant fire fighting plan and identification of individual responsibilities of the brigade members.
- Identification of the fire hazards and associated types of fires that may occur in the plant.
- Identification of the location of fire fighting equipment for each fire area, and familiarization with the layout of the plant, including access and egress routes.
- The proper use of available fire fighting equipment, and the correct method of fighting each type of fire. The types of fires covered should include electrical fires, fires in cable trays, hydrogen fires, flammable liquids, waste/debris fires, and record file fires.
- The proper use of communication, lighting, ventilation, and emergency breathing equipment.
- The proper method for fighting fires inside buildings and confined spaces.
- Detailed review of fire fighting strategies and procedures, Pre-Fire Plans.

In summary, the PNPP fire brigade classroom training program is in compliance with the FIVE/FRSS criteria.

Practice

As described in PAP-1917, Fire Brigade requalification requirements include an annual practice session which must meet the following criteria:

- Practice sessions are held on the proper methods used for fighting the various types of fires that could occur at a nuclear power plant.
- Practice sessions to provide members with first hand experience in actual fire extinguishment.
- Practice sessions to provide the members with first hand experience in the use of emergency breathing apparatus under strenuous conditions encountered in fire fighting.

In summary, the PNPP fire brigade hands-on training program is in compliance with the FIVE/FRSS criteria.

Drills

PNPP Fire Brigade drills provide the following elements consistent with the FIVE evaluation methodology:

- In accordance with PAP-1918, fire drills are performed in the plant so that the brigade can practice as a team.

- These fire drills are held at regular intervals (quarterly), per shift, with intervening intervals no more than five months nor less than one month for any shift.

- Each brigade shift participates in at least one unannounced drill per year as prescribed in PAP-1918.

- At least one drill per year is performed on a backshift for each shift fire brigade as prescribed in PAP-1918.

- As prescribed in PAP-1918, drills are pre-planned to establish training objectives, and drills are critiqued to determine how well the training objectives have been met.

- On a triennial basis, drills are critiqued by qualified individuals independent of the utility's staff as prescribed in PAP-1918.

- Pre-fire plans have been developed in accordance with PAP-1922 for areas containing safety-related equipment and areas presenting a hazard to safety-related equipment. In addition, PAP-1922 can be used to develop Pre-Fire Plan Instructions for other plant areas.

- Pre-fire plans are updated in accordance with PAP-1922, and used as part of fire brigade training activities as prescribed in PAP-1917.

- As prescribed in PAP-1919, the Fire Brigade equipment is inventoried, inspected and maintained in accordance with SFI-0060.

In summary, the PNPP fire brigade drills and drill schedules are in compliance with the FIVE/FRSS criteria.

Records

In accordance with PAP-1917, all fire protection training records documenting successful completion of each qualification requirement are maintained in accordance with TMG-1008 and TMA-4201. Qualifications are tracked through the use of the Perry Training Section Records and Retrieval System.

4.9.5 Total Environment Equipment Survival

4.9.5.1 Potential Adverse Effects on Plant Equipment by Combustion Products

The FIVE/FRSS methodology does not provide criteria for assessment of the potential effects of non-thermal products of combustion on safety/safe-shutdown related equipment. However, for the relatively short duration of the fire event and early recovery period, these effects are considered to be insignificant by FIVE.

4.9.5.2 Spurious/Inadvertent Fire Suppression Activity

The potential effects of spurious/inadvertent suppression system actuation are enveloped by Section 4.9.2.2 of this document.

4.9.5.3 Operator Action Effectiveness

<u>Post-Fire Safe-Shutdown Procedures</u>

Off-Normal Instruction ONI-P54 describes the actions required by Control Room personnel to mitigate the effects of a fire. The actions taken by the Secondary Alarm Station personnel, the first responder, support personnel, and the Fire Brigade personnel are outlined in PAP-1911. In the event of a required Control Room evacuation, ONI-P54 directs the operators to enter ONI-C61 which describes the operator actions related to control room evacuation. Furthermore, ONI-C61 directs entry to IOI-11 which provides the detailed instructions necessary to achieve and maintain a stable reactor shutdown condition from outside the Control Room with a possible loss of offsite power.

<u>Operator Training in Post-Fire Safe-Shutdown Procedures</u>

Periodic operator training in post-fire shutdown procedures is conducted as part of regular operator refresher training for both licensed and non-licensed plant operators. Additional operator training in the use of the shutdown procedures may be indirectly given during fire drills.

<u>Operator Reentry Into Affected Fire Area: Respiratory Protection</u>

The PNPP procedures do not specifically address operator reentry in smoke filled areas, but the following apply:

- Self contained breathing apparatus (SCBA) equipment is provided in the control room and in the Fire Brigade Stations.
- Fixed, battery-backed emergency lighting units are installed along post-fire shutdown access/egress routes and at equipment operating stations.

4.9.6 Control Systems Interactions

The Control Room contains components and circuits for both Method A and B systems. Redundant components and circuits cannot be protected in accordance with Section III.G.2 of Appendix R. Therefore, alternative shutdown capability in accordance with Section III.G.3 of Appendix R is provided.

For a fire in the Control Room, safe shutdown Method A is relied upon for shutdown. A review of all components required for Method A was performed to determine the ability to safely shut down the plant given a fire in the Control Room. The results of this review show:

- All necessary components can be electrically isolated from the Control Room

  or

  there is sufficient time to operate the component manually.

- No spurious failure will prohibit safe shutdown.

Plant procedures reflect the actions necessary to isolate components from the control room, or manually operate components to safely shutdown the plant. Shutdown will be monitored and controlled from the Division 1 Remote Shutdown Panel located in Fire Area 1CC3d.

Circuits necessary to scram the reactor from the control room have not been protected. However, if reactor scram can not be achieved from inside the Control Room, plant procedures address how to accomplish the scram from outside the control room.

In conclusion, the PNPP remote shutdown features provide independent remote, or manual, control and monitoring features. Therefore, the design of the PNPP remote shutdown capabilities is generally immune to the effects of "control systems interactions" as defined within the scope of the FIVE methodology.

4.9.7 Conclusions

The results of the topical assessments performed under the FIVE Fire Risk Scoping Study indicate that all of the FRSS issues have been adequately addressed by PNPP, and the applicable aspects of the PNPP Fire Protection Program therefore are in conformance with the intent of the FRSS guidelines, as tabulated in Attachment 10.5 of the FIVE methodology.

4.10 USI A-45 and Other Safety Issues

As discussed in Section 4.6, the frequency of core damage scenarios involving loss of containment heat removal systems is estimated to be 1.2×10^{-5}/yr. This estimate is dominated by the Unit 1 Division 2 Switchgear Room fire scenarios, which have certain inherent conservatisms particularly when considering the potential for recovery in the time frame for initiation of long term containment heat removal (on the order of 15 to 20 hours). Further discussion of the conservatisms is provided in Section 4.7.3.2.

4.11 References

4-1 NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Final Report, June 1991.

4-2a EPRI TR-100370s, "Fire-Induced Vulnerability Evaluation (FIVE)," Final Report, April 1992.

4-2b "Revision 1 to EPRI Final Report, dated April 1992, TR-100370, Fire-Induced Vulnerability Evaluation Methodology," Letter from William H. Rasin, NUMARC, Nuclear Management and Resource Council, September 29th 1993.

4-3 EPRI RP-3385-01, "Fire PRA Implementation Guide," January 1994 (Draft).

4-4 NSAC-181, "Fire PRA Requantification Studies," Final Report, March 1993.

4-5 NUREG/CR-5088, "Fire Risk Scoping Study," January 1989.

4-6 NSAC-178L, "Fire Events Database for U.S. Nuclear Power Plants," Final Report, June 1992.

4-7 "Individual Plant Examination for Perry Nuclear Power Plant," Cleveland Electric Illuminating Company, July 1992.

4-8 "Appendix R Safe Shutdown Capability Report."

4-9 Updated Safety Analysis Report (USAR) for Unit 1, Perry Nuclear Power Plant, Appendix 9A, Rev 2.

4-10a NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part I: Cabinet Effects," April 1987.

4-10b NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part II: Room Effects Tests," November 1988.

4-11 IOI-11, "Integrated Operating Instruction, Shutdown from Outside the Control Room," Rev 6.

4-12 NEDO-10466, "Power Generation Control Complex -- Design Criteria and Safety Evaluation," Rev 1, General Electric Company, 1977.

Table 4-1 - PNPP Fire Areas and Associated Compartments

Fire Area	Fire Compartment	Description
1AB1a	1AB1a	AX 568 Unit 1 LPCS Pump Room
1AB1d	1AB1d	AX568 Unit 1 RHR C Pump Room
1AB1f	1AB1f	AX 568 Unit 1 HPCS Pump Room
1AB1g	1AB1g	AX 568 Unit 1 Corridor
1AB2	1AB2	AX 568/599 Unit 1 RCIC (Fire Zones 1AB1c & 1AB2)
1ABE	1ABE	AX 568/620 Unit 1 RHR A (Fire Zones 1AB1b & 1AB3a)
1ABW	1ABW	AX 568/620 Unit 1 RHR B (Fire Zones 1AB1e & 1AB3b)
1ABSTWN	1ABSTWN	AX 574/652 Unit 1 North Stairwell
1ABSTWS	1ABSTWS	AX 574/652 Unit 1 South Stairwell
1CC3a	1CC3a	CC 620 Unit 1 Division 2 Switchgear Room
1CC3b	1CC3b	CC 620 Unit 1 Division 3 Switchgear Room
1CC3c	1CC3c	CC 620 Unit 1 Division 1 Switchgear Room
1CC3d	1CC3d	CC 620 Unit 1 Remote Shutdown Room
1CC3e	1CC3e	CC 620 Unit 1 Switchgear Room Corridor
1CC4 638/654	1CC4 638/654	CC 638 Unit 1 Elevator Corridor
1CC4a	1CC4a	CC 638 Unit 1 Division 2 Cable Spreading Room
1CC4b	1CC4b	CC 638 Unit 1 Division 2 Cable Chase
1CC4c	1CC4c	CC 638 Unit 1 Division 2 DC Switchgear Room
1CC4d	1CC4d	CC 638 Unit 1 Division 2 Battery Room
1CC4e	1CC4e	CC 638 Unit 1 Division 1 Cable Spreading Room
1CC4f	1CC4f	CC 638 Unit 1 Division 1 Cable Chase
1CC4g	1CC4g	CC 638 Unit 1 Division 1 DC Switchgear Room
1CC4h	1CC4h	CC 638 Unit 1 Division 1 Battery Room
1CC4i	1CC4i	CC 638 Computer Room
1CC5a	1CC5a	CC 654 Unit 1 Control Room
1CC5b	1CC5b	CC 654 Unit 1 Control Room Office
1CC5c	1CC5c	CC 654 Unit 1 Control Room Corridor
1CC6	1CC6	CC 679 Unit 1/2 Division 2 HVAC Room
1CCSTW	1CCSTW	CC 574/679 North Stairwell
2CC3a	2CC3a	CC 620 Unit 2 Division 2 Switchgear Room
2CC3b	2CC3b	CC 620 Unit 2 Division 3 Switchgear Room
2CC3c	2CC3c	CC 620 Unit 2 Division 1 Switchgear Room
2CC3d	2CC3d	CC 620 Unit 2 Remote Shutdown Room
2CC3e	2CC3e	CC 620 Unit 2 Switchgear Room Corridor
2CC4 638/654	2CC4 638/654	CC 638 Unit 1 Elevator Corridor
2CC4a	2CC4a	CC 638 Unit 2 Division 2 Cable Spreading Room
2CC4b	2CC4b	CC 638 Unit 2 Division 2 Cable Chase
2CC4c	2CC4c	CC 638 Unit 2 Division 2 DC Switchgear Room
2CC4d	2CC4d	CC 638 Unit 2 Division 2 Battery Room
2CC4e	2CC4e	CC 638 Unit 2 Division 1 Cable Spreading Room
2CC4f	2CC4f	CC 638 Unit 2 Division 1 Cable Chase
2CC4g	2CC4g	CC 638 Unit 2 Division 1 DC Switchgear Room
2CC4h	2CC4h	CC 638 Unit 2 Division 1 Battery Room
2CC4i	2CC4i	CC 638 Unit 2 Computer Room
2CC5a	2CC5a	CC 654 Unit 2 Control Room
2CC5b	2CC5b	CC 654 Unit 2 Control Room Corridor
2CC6	2CC6	CC 679 Unit 1/2 Division 1 HVAC Room
2CCSTW	2CCSTW	CC 574/679 South Stairwell
CC1	CC1	CC 574 (Fire Zones CC1a, CC1b & CC1c)
CC2	CC2/1	CC 599 Northwest portion of CC 599 (Equivalent to CC2b)
	CC2/2	CC 599 Corridor (Corridor through CC2c & CC2a)
	CC2/3	CC 599 Chemistry Lab (Equivalent to CC2c)

Fire Area	Fire Compartment	Description
	CC2/4	CC 599 NCC Pump Room (Southern portion of CC2a)
	CC2/5	CC 599 NCC Heat Exchanger Room (Northern portion of CC2a)
CC6	CC6	CC 679 Unit 1/2 HVAC Chase
1DG1a	1DG1a	DG 620 Unit 1 Division 2 Diesel Generator Room
1DG1b	1DG1b	DG 620 Unit 1 Division 3 Diesel Generator Room
1DG1c	1DG1c	DG 620 Unit 1 Division 1 Diesel Generator Room
2DG1a	2DG1a	DG 620 Unit 2 Division 2 Diesel Generator Room
2DG1b	2DG1b	DG 620 Unit 2 Division 3 Diesel Generator Room
2DG1c	2DG1c	DG 620 Unit 2 Division 1 Diesel Generator Room
DG1d	DG1d	DG 620 Diesel Generator Building Corridor
DG1e	DG1e	DG 620 Low Level Radwaste Room
DGDB1	DGDB1	Diesel Generator Duct Bank 1 (Division 1)
DGDB2	DGDB2	Diesel Generator Duct Bank 2 (Division 2)
ESW1a	ESW1a	Emergency Service Water Pumphouse - ESW Pump Room
ESW1b	ESW1b	Emergency Service Water Pumphouse - Diesel Fire Pump Room
ESDB1	ESDB1	ESW Duct Bank 1 (Division 1)
ESDB2	ESDB2	ESW Duct Bank 2 (Division 2)
FH	FH1	Fuel Handling Building 574
	FH2a	Fuel Handling Building 599 - Unit 1
	FH2b	Fuel Handling Building 599 - Unit 2
	FH3	Fuel Handling Building 620
IB	IB1	Intermediate Building 574
	IB2	Intermediate Building 599
	IB3	Intermediate Building 620
	IB4	Intermediate Building 654/665
	IB5	Intermediate Building 682
1TB	1TB	Unit 1 Turbine Building
1TPC	1TPC/1	Unit 1 Turbine Power Complex - Switchgear Rooms & Amertap
	1TPC/2	Unit 1 Turbine Power Complex - Sub-basement
RWB	RWB	Radwaste Building
Yard (LOOP)	Yard	Yard Transformers
UNIT2	UNIT2	Unit 2 Aux Bldg, Turb Bldg & Stm Tnl, & Turb Power Complex

Table 4-2 - Fire Induced System Loss and Initiating Event Matrix

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
1AB1a	1AB1a		Low Pressure Core Spray	A2
1AB1d	1AB1d		Residual Heat Removal C - Low Press Coolant Injection Mode	B
			RHR C Pump Room Cooler	B
1AB1f	1AB1f	X	Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
1AB1g	1AB1g		Residual Heat Removal A - Steam Condensing Mode	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Residual Heat Removal B - Suppression Pool Cooling Mode	B
			Residual Heat Removal C - Low Press Coolant Injection Mode	B
			Low Pressure Core Spray	A2
			Reactor Core Isolation Cooling	A1
			Emergency Service Water A	A
			Electric Power for Method A	A
			Instrumentation for Method A	A
			Instrumentation for Method B	B
1AB2	1AB2	X	Rx Press Vessel Pressure Control (MSIVs & ADS)	A
			Residual Heat Removal A - Steam Condensing Mode	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal C - Low Press Coolant Injection Mode	B
			Low Pressure Core Spray	A2
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			RHR A Pump Room Cooler	A
			RHR B Pump Room Cooler	B
			RHR C Pump Room Cooler	B
			RCIC Pump Room Cooler	A1
			Emergency Service Water A	A
			Safety Related Instrument Air A	A
			Safety Related Instrument Air B	B
			Electric Power for Method A	A
			Instrumentation for Method A	A
1ABE	1ABE	X	Rx Press Vessel Pressure Control (MSIVs & ADS)	A
			Residual Heat Removal A - Steam Condensing Mode	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Low Pressure Core Spray	A2
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			RHR A Pump Room Cooler	A
			RCIC Pump Room Cooler	A1
			Emergency Service Water A	A
			Safety Related Instrument Air A	A
			Instrumentation for Method A	A

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
1ABW	1ABW	X	Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Residual Heat Removal B - Suppression Pool Cooling Mode	B
			Residual Heat Removal C - Low Press Coolant Injection Mode	B
			Reactor Core Isolation Cooling	A1
			RHR B Pump Room Cooler	B
			RHR C Pump Room Cooler	B
			Emergency Service Water B	B
			Safety Related Instrument Air B	B
			Electric Power for Method A	A
			Instrumentation for Method B	B
1ABSTWN	1ABSTWN		None	
1ABSTWS	1ABSTWS		None	
1CC3a	1CC3a	X	Cntrl Cmplx Chilled Water A	A
			Cntrl Cmplx Chilled Water B	B
			ESW Screen Wash B	B
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Rx Recirculation High/Low Press Interface	B
			Reactivity Control for Method A	A
			Reactivity Control for Method B	B
			Rx Protection System for Method A	A
			Rx Protection System for Method B	B
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Residual Heat Removal B - Suppression Pool Cooling Mode	B
			Residual Heat Removal C - Low Press Coolant Injection Mode	B
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	B
			MCC Swtchgr & Misc Elec HVAC B	B
			Battery Rooms Exhaust B	B
			Control Room HVAC A	A
			Control Room HVAC B	B
			Control Room Emergency Recirculation B	B
			ECC Pump Area Cooling B	B
			ESW Pumphouse Ventilation B	B
			RHR B Pump Room Cooler	B
			RHR C Pump Room Cooler	B
			D/G Bldg Ventilation B	B
			Emergency Closed Cooling B	B
			Emergency Service Water B	B
			Safety Related Instrument Air B	B
			Electric Power for Method A	A
			Electric Power for Method B	B
			Division 2 Diesel Generator	B
1CC3b	1CC3b		None	
1CC3c	1CC3c	X	Cntrl Cmplx Chilled Water A	A

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A2	A2
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Rx Recirculation High/Low Press Interface	A
			Rx Recirculation High/Low Press Interface	B
			Reactivity Control for Method A	A
			Reactivity Control for Method B	B
			Rx Protection System for Method A	A
			Rx Protection System for Method B	B
			Residual Heat Removal A - Steam Condensing Mode	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Low Pressure Core Spray	A2
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			MCC Swtchgr & Misc Elec HVAC A	A
			Battery Rooms Exhaust A	A
			Control Room HVAC A	A
			Control Room Emergency Recirculation A	A
			ECC Pump Area Cooling A	A
			ESW Pumphouse Ventilation A	A
			RHR A Pump Room Cooler	A
			RCIC Pump Room Cooler	A1
			Emergency Closed Cooling A	A
			Emergency Service Water A	A
			Safety Related Instrument Air A	A
			Electric Power for Method A	A
			Electric Power for Method B	B
			Division 1 Diesel Generator	A
			Instrumentation for Method A	A
1CC3d	1CC3d		Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Reactivity Control for Method A	A
			Residual Heat Removal A - Steam Condensing Mode	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			D/G Bldg Ventilation A	A
			Emergency Closed Cooling A	A
			Emergency Service Water A	A
			Electric Power for Method A	A
			Electric Power for Method B	B
			Division 1 Diesel Generator	A
			Instrumentation for Method A	A
1CC3e	1CC3e		ESW Screen Wash B	B
			Control Room HVAC B	B
			D/G Bldg Ventilation A	A
			D/G Bldg Ventilation B	B
			ECC Pump Area Cooling A	A
			ECC Pump Area Cooling B	B

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
			Electric Power for Method A	A
			Division 1 Diesel Generator	A
			Division 2 Diesel Generator	B
1CC4	1CC4		Electric Power for Method A	A
638/654	638/654		Electric Power for Method B	B
1CC4a	1CC4a	X	Cntrl Cmplx Chilled Water B	B
			ESW Screen Wash B	B
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Rx Recirculation High/Low Press Interface	B
			Reactivity Control for Method A	A
			Reactivity Control for Method B	B
			Rx Protection System for Method A	A
			Rx Protection System for Method B	B
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Residual Heat Removal B - Suppression Pool Cooling Mode	B
			Residual Heat Removal C - Low Press Coolant Injection Mode	B
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	B
			MCC Swtchgr & Misc Elec HVAC B	B
			Battery Rooms Exhaust B	B
			Control Room HVAC A	A
			Control Room HVAC B	B
			Control Room Emergency Recirculation A	A
			Control Room Emergency Recirculation B	B
			ECC Pump Area Cooling B	B
			ESW Pumphouse Ventilation B	B
			RHR B Pump Room Cooler	B
			RHR C Pump Room Cooler	B
			D/G Bldg Ventilation B	B
			Emergency Closed Cooling B	B
			Emergency Service Water B	B
			Safety Related Instrument Air B	B
			Electric Power for Method A	A
			Electric Power for Method B	B
			Division 2 Diesel Generator	B
			Instrumentation for Method B	B
1CC4b	1CC4b	X	Cntrl Cmplx Chilled Water A	A
			Cntrl Cmplx Chilled Water B	B
			ESW Screen Wash B	B
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Rx Recirculation High/Low Press Interface	A
			Rx Recirculation High/Low Press Interface	B
			Reactivity Control for Method B	B
			Rx Protection System for Method A	A
			Rx Protection System for Method B	B
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
			Residual Heat Removal B - Suppression Pool Cooling Mode	B
			Residual Heat Removal C - Low Press Coolant Injection Mode	B
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			Rx Water Clean-Up High/Low Press Interface	B
			MCC Swtchgr & Misc Elec HVAC B	B
			Battery Rooms Exhaust B	B
			Control Room HVAC B	B
			Control Room Emergency Recirculation B	B
			ECC Pump Area Cooling B	B
			RHR B Pump Room Cooler	B
			RHR C Pump Room Cooler	B
			Emergency Closed Cooling B	B
			Emergency Service Water B	B
			Safety Related Instrument Air B	B
			Electric Power for Method A	A
			Electric Power for Method B	B
			Instrumentation for Method B	B
1CC4c	1CC4c		Reactor Core Isolation Cooling	A1
			Electric Power for Method A	A
			Electric Power for Method B	B
1CC4d	1CC4d		Electric Power for Method B	B
1CC4e	1CC4e	X	Cntrl Cmplx Chilled Water A	A
			Cntrl Cmplx Chilled Water B	B
			ESW Screen Wash A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A2	A2
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Rx Recirculation High/Low Press Interface	A
			Rx Recirculation High/Low Press Interface	B
			Reactivity Control for Method A	A
			Rx Protection System for Method A	A
			Rx Protection System for Method B	B
			Residual Heat Removal A - Steam Condensing Mode	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Low Pressure Core Spray	A2
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			Rx Water Clean-Up High/Low Press Interface	B
			MCC Swtchgr & Misc Elec HVAC A	A
			Battery Rooms Exhaust A	A
			Control Room HVAC A	A
			Control Room HVAC B	B
			Control Room Emergency Recirculation A	A
			Control Room Emergency Recirculation B	B
			ECC Pump Area Cooling A	A
			ESW Pumphouse Ventilation A	A
			RHR A Pump Room Cooler	A
			RCIC Pump Room Cooler	A1
			D/G Bldg Ventilation A	A

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
			Emergency Closed Cooling A	A
			Emergency Closed Cooling B	B
			Emergency Service Water A	A
			Emergency Service Water B	B
			Safety Related Instrument Air A	A
			Electric Power for Method A	A
			Electric Power for Method B	B
			D.C. Power for Method A	A
			D.C. Power for Method B	B
			Instrumentation for Method A	A
1CC4f	1CC4f	X	Cntrl Cmplx Chilled Water A	A
			Reactivity Control for Method A	A
			Rx Protection System for Method A	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			MCC Swtchgr & Misc Elec HVAC A	A
			Battery Rooms Exhaust A	A
			Control Room HVAC A	A
			Control Room Emergency Recirculation A	A
			ESW Pumphouse Ventilation A	A
			Emergency Closed Cooling A	A
			Emergency Service Water A	A
			Safety Related Instrument Air A	A
			Instrumentation for Method A	A
1CC4g	1CC4g		Rx Press Vessel Press Control (MSIVs & ADS) for Method A2	A2
			Reactivity Control for Method A	A
			Electric Power for Method A	A
1CC4h	1CC4h		Electric Power for Method A	A
1CC4i	1CC4i		None	
1CC5a	1CC5a	X	Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Rx Protection System for Method A	A
			Rx Protection System for Method A1	A1
			Rx Protection System for Method B	B
			Instrumentation for Method A	A
			Instrumentation for Method B	B
1CC5b	1CC5b		Control Room HVAC B	B
1CC5c	1CC5c		Control Room HVAC B	B
			Electric Power for Method B	B
1CC6	1CC6		Cntrl Cmplx Chilled Water A	A
			MCC Swtchgr & Misc Elec HVAC A	A
			MCC Swtchgr & Misc Elec HVAC B	B

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
			Battery Rooms Exhaust A	A
			Battery Rooms Exhaust B	B
			Control Room HVAC A	A
			Control Room HVAC B	B
			Control Room Emergency Recirculation A	A
			Control Room Emergency Recirculation B	B
			Electric Power for Method A	A
1CCSTW	1CCSTW		Electric Power for Method A	A
2CC3a	2CC3a		None	
2CC3b	2CC3b		None	
2CC3c	2CC3c		None	
2CC3d	2CC3d		None	
2CC3e	2CC3e		None	
2CC4	2CC4		None	
638/654	638/654			
2CC4a	2CC4a		Electric Power for Method A	A
			Electric Power for Method B	B
2CC4b	2CC4b		Cntrl Cmplx Chilled Water B	B
			ECC Pump Area Cooling B	B
			Emergency Closed Cooling B	B
			Emergency Service Water B	B
2CC4c	2CC4c		Electric Power for Method B	B
2CC4d	2CC4d		Electric Power for Method B	B
2CC4e	2CC4e		Electric Power for Method A	A
2CC4f	2CC4f		None	
2CC4g	2CC4g		Electric Power for Method A	A
2CC4h	2CC4h		Electric Power for Method A	A
2CC4i	2CC4i		None	
2CC5a	2CC5a		None	
2CC5b	2CC5b		Control Room HVAC B	B
2CC6	2CC6		Cntrl Cmplx Chilled Water A	A
			MCC Swtchgr & Misc Elec HVAC A	A
			MCC Swtchgr & Misc Elec HVAC B	B
			Battery Rooms Exhaust A	A
			Battery Rooms Exhaust B	B
			Control Room HVAC A	A

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
			Control Room HVAC B	B
			Control Room Emergency Recirculation B	B
			Electric Power for Method B	B
2CCSTW	2CCSTW		None	
CC1	CC1	X	Cntrl Cmplx Chilled Water A	A
			Cntrl Cmplx Chilled Water B	B
			ECC Pump Area Cooling A	A
			ECC Pump Area Cooling B	B
			Emergency Closed Cooling A	A
			Emergency Closed Cooling B	B
			Emergency Service Water A	A
			Emergency Service Water B	B
CC2	CC2/1	X	Cntrl Cmplx Chilled Water A	A
			Cntrl Cmplx Chilled Water B	B
			ECC Pump Area Cooling A	A
			ECC Pump Area Cooling B	B
			Emergency Closed Cooling A	A
			Emergency Closed Cooling B	B
			Emergency Service Water A	A
			Emergency Service Water B	B
			Electric Power for Method A	A
CC2	CC2/2	X	Cntrl Cmplx Chilled Water A	A
			Cntrl Cmplx Chilled Water B	B
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Residual Heat Removal A - Steam Condensing Mode	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Low Pressure Core Spray	A2
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			ECC Pump Area Cooling A	A
			ECC Pump Area Cooling B	B
			RHR A Pump Room Cooler	A
			RCIC Pump Room Cooler	A1
			Emergency Closed Cooling A	A
			Emergency Closed Cooling B	B
			Emergency Service Water A	A
			Emergency Service Water B	B
			Electric Power for Method A	A
			Electric Power for Method B	B
			Instrumentation for Method A	A
CC2	CC2/3		None	
CC2	CC2/4	X	Cntrl Cmplx Chilled Water A	A
			Cntrl Cmplx Chilled Water B	B
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Residual Heat Removal A - Steam Condensing Mode	A

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Low Pressure Core Spray	A2
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			ECC Pump Area Cooling A	A
			ECC Pump Area Cooling B	B
			RHR A Pump Room Cooler	A
			RCIC Pump Room Cooler	A1
			Emergency Closed Cooling A	A
			Emergency Closed Cooling B	B
			Emergency Service Water A	A
			Emergency Service Water B	B
			Safety Related Instrument Air A	A
			Electric Power for Method A	A
			Electric Power for Method B	B
			Instrumentation for Method A	A
CC2	CC2/5	X	Cntrl Cmplx Chilled Water A	A
			Cntrl Cmplx Chilled Water B	B
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Residual Heat Removal A - Steam Condensing Mode	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Low Pressure Core Spray	A2
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			ECC Pump Area Cooling A	A
			ECC Pump Area Cooling B	B
			RHR A Pump Room Cooler	A
			RCIC Pump Room Cooler	A1
			Emergency Closed Cooling A	A
			Emergency Closed Cooling B	B
			Emergency Service Water A	A
			Emergency Service Water B	B
			Safety Related Instrument Air A	A
			Electric Power for Method A	A
			Electric Power for Method B	B
			Instrumentation for Method A	A
CC6	CC6		MCC Swtchgr & Misc Elec HVAC B	B
			Battery Rooms Exhaust B	B
			Control Room HVAC B	B
			Control Room Emergency Recirculation B	B
			Electric Power for Method B	B
1DG1a	1DG1a		D/G Bldg Ventilation B	B
			Emergency Service Water B	B
			Electric Power for Method B	B
			Division 2 Diesel Generator	B
1DG1b	1DG1b		None	

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
1DG1c	1DG1c		D/G Bldg Ventilation A	A
			Emergency Service Water A	A
			Electric Power for Method A	A
			Division 1 Diesel Generator	A
2DG1a	2DG1a		None	
2DG1b	2DG1b		None	
2DG1c	2DG1c		None	
DG1d	DG1d		D/G Bldg Ventilation A	A
			D/G Bldg Ventilation B	B
			Emergency Service Water A	A
			Emergency Service Water B	B
			Electric Power for Method B	B
			Division 1 Diesel Generator	A
			Division 2 Diesel Generator	B
DG1e	DG1e		Residual Heat Removal B - Shutdown Cooling Mode	B
			D/G Bldg Ventilation B	B
			Emergency Service Water B	B
DGDB1	DGDB1		Electric Power for Method A	A
DGDB2	DGDB2		Electric Power for Method B	B
ESW1a	ESW1a		ESW Screen Wash A	A
			ESW Screen Wash B	B
			ESW Pumphouse Ventilation A	A
			ESW Pumphouse Ventilation B	B
			Emergency Service Water A	A
			Emergency Service Water B	B
			Electric Power for Method A	A
			Electric Power for Method B	B
ESW1b	ESW1b		None	
ESWDB1	ESWDB1		ESW Screen Wash A	A
			ESW Pumphouse Ventilation A	A
			Emergency Service Water A	A
			Electric Power for Method A	A
ESWDB2	ESWDB2		ESW Screen Wash B	B
			ESW Pumphouse Ventilation B	B
			Emergency Service Water B	B
			Electric Power for Method B	B
FH	FH1	X	None	
FH	FH2a		ESW Screen Wash A	A
			ESW Pumphouse Ventilation A	A
			Emergency Service Water A	A
			Electric Power for Method A	A

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
FH	FH2b		None	
FH	FH3		ESW Screen Wash A	A
			ESW Pumphouse Ventilation A	A
			Emergency Service Water A	A
			Electric Power for Method A	A
IB	IB1	X	None	
IB	IB2	X	Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Residual Heat Removal A - Steam Condensing Mode	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal A - Suppression Pool Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Low Pressure Core Spray	A2
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
			RHR A Pump Room Cooler	A
			RCIC Pump Room Cooler	A1
			Emergency Service Water A	A
			Safety Related Instrument Air A	A
			Instrumentation for Method A	A
IB	IB3	X	ESW Screen Wash A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Reactivity Control for Method A	A
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Residual Heat Removal B - Suppression Pool Cooling Mode	B
			Residual Heat Removal C - Low Pressure Coolant Injection	B
			Reactor Core Isolation Cooling	A1
			RHR B Pump Room Cooler	B
			RHR C Pump Room Cooler	B
			Emergency Service Water A	A
			Emergency Service Water B	B
			Safety Related Instrument Air B	B
			Electric Power for Method A	A
			Instrumentation for Method A	A
			Instrumentation for Method B	B
IB	IB4	X	Residual Heat Removal A - Shutdown Cooling Mode	A
IB	IB5		None	
1TB	1TB	X	Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B
			Residual Heat Removal A - Shutdown Cooling Mode	A
			Residual Heat Removal B - Shutdown Cooling Mode	B
			Reactor Core Isolation Cooling	A1
			Rx Water Clean-Up High/Low Press Interface	A
1TPC	1TPC/1	X	Rx Press Vessel Press Control (MSIVs & ADS) for Method A	A
			Rx Press Vessel Press Control (MSIVs & ADS) for Method B	B

Fire Area	Fire Compt	Plant Trip Initiator	Safe Shutdown System	Method
			Reactivity Control for Method A	A
1TPC	1TPC/2		None	
RWB	RWB		None	
Yard (LOOP)	Yard	X	None	
UNIT2	UNIT2		None	

Table 4-3 - Fire Area/Compartment Screening Results through Phase II Step 2

Fire Area	Fire Comp	Fire Induced Plant Initiator	Safe Shutdown Equipment	Fire Frequency (F1)	Conditional CDP (P2)	F2 = F1 x P2	Screened
1AB1a			X	2.312E-03	1.83E-05	4.231E-08	Phase II Step 2
1Ab1d			X	2.311E-03	1.75E-05	4.044E-08	Phase II Step 2
1AB1f		X	X	2.751E-03	5.19E-04	1.428E-06	No
1AB1g			X	5.349E-03	5.39E-05	2.883E-07	Phase II Step 2
1AB2		X	X	8.255E-03	7.49E-04	6.183E-06	No
1ABE		X	X	3.802E-03	4.87E-04	1.852E-06	No
1ABSTWN				-	-	-	Phase I
1ABSTWS				-	-	-	Phase I
1ABW		X	X	5.439E-03	4.44E-05	2.415E-07	Phase II Step 2
1CC3a		X	X	7.994E-03	7.01E-03	5.604E-05	No
1CC3b				-	-	-	Phase I
1CC3c		X	X	7.670E-03	4.94E-04	3.789E-06	No
1CC3d			X	9.906E-04	4.87E-04	4.824E-07	Phase II Step 2
1CC3e			X	1.089E-03	2.61E-04	2.842E-07	Phase II Step 2
1CC4 638/654			X	6.910E-04	8.28E-05	5.721E-08	Phase II Step 2
1CC4a		X	X	3.131E-03	2.92E-04	9.143E-07	Phase II Step 2
1CC4b		X	X	2.375E-04	8.55E-03	2.031E-05	No
1CC4c			X	3.126E-03	2.52E-04	7.878E-07	Phase II Step 2
1CC4d			X	2.247E-03	2.52E-04	5.662E-07	Phase II Step 2
1CC4e		X	X	2.498E-03	5.28E-05	1.319E-07	Phase II Step 2
1CC4f		X	X	3.058E-03	1.21E-04	3.700E-07	Phase II Step 2
1CC4g			X	2.482E-03	2.63E-04	6.528E-07	Phase II Step 2
1CC4h			X	2.247E-03	2.63E-04	5.910E-07	Phase II Step 2
1CC4i				-	-	-	Phase I
1CC5a		X	X	1.033E-02	1.21E-02	1.250E-04	No
1CC5b			X	9.480E-04	8.28E-05	7.849E-08	Phase II Step 2
1CC5c			X	1.814E-03	8.28E-05	1.502E-07	Phase II Step 2
1CC6			X	5.810E-03	8.28E-05	4.811E-07	Phase II Step 2
1CCSTW			X	6.468E-04	5.86E-04	3.790E-07	Phase II Step 2
1DG1a			X	3.021E-02	4.56E-06	1.378E-07	Phase II Step 2

Table 4-3 - Fire Area/Compartment Screening Results through Phase II Step 2

Fire Area	Fire Comp	Fire Induced Plant Initiator	Safe Shutdown Equipment	Fire Frequency (F1)	Conditional CDP (P2)	F2 = F1 x P2	Screened
1DG1b				-	-	-	Phase I
1DG1c			X	3.021E-02	1.27E-06	3.837E-08	Phase II Step 2
1TB		X	X	1.391E-01	7.44E-04	1.035E-04	No
1TPC		X	X	7.446E-03	4.38E-04	3.261E-06	No
2CC3a				-	-	-	Phase I
2CC3b				-	-	-	Phase I
2CC3d				-	-	-	Phase I
2CC3e				-	-	-	Phase I
2CC4 638/654				-	-	-	Phase I
2CC4a			X	2.509E-03	8.28E-05	2.077E-07	Phase II Step 2
2CC4b			X	8.792E-04	5.39E-04	4.739E-07	Phase II Step 2
2CC4c			X	4.803E-03	8.28E-05	3.977E-07	Phase II Step 2
2CC4d			X	2.289E-03	8.28E-05	1.895E-07	Phase II Step 2
2CC4e			X	2.735E-03	8.28E-05	2.265E-07	Phase II Step 2
2CC4f				-	-	-	Phase I
2CC4g			X	4.803E-03	8.28E-05	3.977E-07	Phase II Step 2
2CC4h			X	1.489E-03	8.28E-05	1.233E-07	Phase II step 2
2CC4i				-	-	-	Phase I
2CC5a				-	-	-	Phase I
2CC5b			X	6.710E-03	8.28E-05	5.556E-07	Phase II Step 2
2CC6			X	5.274E-03	8.28E-05	4.367E-07	Phase II Step 2
2CCSTW				-	-	-	Phase I
2DG1a				-	-	-	Phase I
2DG1b				-	-	-	Phase I
2DG1c				-	-	-	Phase I
CC1			X	8.698E-03	1.35E-02	1.174E-04	No
CC2z	CC2/1	X	X	7.755E-04	3.51E-03	2.722E-06	No
	CC2/2	X	X	1.017E-03	3.51E-03	3.570E-06	No
	CC2/3			-	-	-	Phase I
	CC2/4	X	X	2.912E-03	3.62E-01	1.054E-03	No

Table 4-3 - Fire Area/Compartment Screening Results through Phase II Step 2

Fire Area	Fire Comp	Fire Induced Plant Initiator	Safe Shutdown Equipment	Fire Frequency (F1)	Conditional CDP (P2)	F2 = F1 x P2	Screened
	CC2/5	X	X	1.897E-03	5.28E-03	1.002E-05	No
CC6			X	1.186E-03	8.28E-05	9.820E-08	Phase II Step 2
DG1d			X	1.604E-03	6.20E-05	9.945E-08	Phase II Step 2
DG1e			X	7.002E-04	7.26E-03	5.083E-06	Phase II Step 2
DGDB1			X	1.995E-06	3.66E-06	7.302E-12	Phase II Step 2
DGDB2			X	1.995E-06	4.56E-06	9.097E-12	Phase II Step 2
ESW1	ESW1a		X	6.381E-03	2.61E-04	1.791E-06	No
	ESW1b			-	-	-	Phase I
ESWDB1			X	1.995E-06	1.22E-06	2.434E-12	Phase II Step 2
ESWDB2			X	1.995E-06	2.76E-06	5.506E-12	Phase II Step 2
FHz	FH1	X		5.151E-03	8.28E-05	4.265E-07	Phase II Step 2
	FH2a		X	1.321E-03	6.46E-04	8.534E-07	Phase II Step 2
	FH2b			-	-	-	Phase I
	FH3		X	2.529E-03	6.46E-04	1.634E-06	No
IBz	IB1	X		8.856E-03	8.28E-05	7.333E-07	Phase II Step 2
	IB2	X	X	7.028E-03	3.63E-02	2.551E-04	No
	IB3	X	X	9.802E-03	8.66E-05	8.489E-07	Phase II Step 2
	IB4	X	X	4.268E-03	8.59E-05	3.666E-07	Phase II Step 2
	IB5			-	-	-	Phase I
RWB				-	-	-	Phase I
Yard (LOOP)		X		1.600E-03	4.38E-04	7.000E-07	Phase II Step 2
UNIT 2				-	-	-	Phase I

Table 4-4 - F2 Calculations for Fire Compartment 1CC3a Fire Scenarios

Fire Compartment 1CC3a Fire Damage Scenarios								
Scen #	Source Type	Source Identification (Section if SWGR, Bus or MCC)	Supp (Y/N)	Safe Shutdown Equipment Damaged	SSD Trays and/or Conduit Damaged	F1'	P2'	F3'
1	4.16 kV SWGR	Any	N	All in Fire Compartment	All	8.434E-05	FDS 0 7.01E-03	5.91E-07
2	4.16 kV SWGR	1) 4,160/480 V XFMR XFH-1-A	Y	4.16 kV SWGR	Tray 260/271 Tray 1316/1321 Tray 1320	4.217E-05	FDS 0 7.01E-03	2.96E-07
3	4.16 kV SWGR	2) SW Pump B P41-C001B 3) NCC Pump B P43-C001B	Y	4.16 kV SWGR	Tray 260 Tray 1316	8.434-05	FDS 0 7.01E-03	5.91E-07
4	4.16 kV SWGR	4) CRD Pump B C11-C001B 5) Supply to Bus XH12 6) Alt. Pref. Source - LH-2-A 7) Pref. Source - LH-1-A 8) Spare 9) RHR Pump C E12-C002C 10) 4160/480 V XFMR EHF-1-D	Y	4.16 kV SWGR	Tray 260 Tray 1316 1R33D30B 1R33D54B	2.952E-04	FDS 0 7.01E-03	2.07E-07
5	4.16 kV SWGR	11) RHR Pump B E12-C002B	Y	4.16 kV SWGR	Tray 260 Tray 1316 1R33D54B	4.217E-05	FDS 0 7.01E-03	2.96E-07
6	4.16 kV SWGR	12) Spare 13) CCCW Chiller B P47-B001B 14) ESW Pump B P45-C001B 15) 4160/480 V XFMR EHF-1-C 16) Spare	Y	4.16 kV SWGR	Tray 260 Tray 1316 1R33D29B	2.109E-05	FDS 0 7.01E-03	1.48E-06
7	4.16 kV SWGR	17) Aux. Compt. EH-1202 18) From D/G R43-S001B	Y	4.16 kV SWGR	Tray 260 Tray 1316	8.434E-05	FDS 0 7.01E-03	5.91E-07
8	480 V Bus	Any	N	All in Fire Compartment	All	3.300E-05	FDS 0 7.01E-03	2.31E-07
9	480 V Bus	1) Inst. Compt. EF1C1, Main supply breaker from EH-12, and Inst. Compt. EF1C2 2) FPCC Pump B G41-C003B, M23 Fan B M23-C001B, Reserve Battery ChargerEFD-12-B (R42-S009), and MCC EF1C07 (R24-S023)	Y	480 V Bus EF-1-C	Tray 272 Tray 1317 Tray 1379 Tray 1608	8.434E-05	FDS 2 3.69E-04	3.11E-08
10	480 V Bus	3) MCC EF1C08 (R24-S024) MCC EF1C09 (R24-S025)	Y	480 V Bus EF-1-C 480 V Bus EF-1-D	Tray 272 Tray 1317 Tray 1379 Tray 1608	4.217E-05	FDS 0 7.01E-03	2.96E-07
11	480 V Bus	4) MCC EF1C12 (R24-S032), EF-1-C to EF-1-D X-tie Breaker, and H2 Recombiner M51-D001B	Y	480 V Bus	Tray 272 Tray 1317 Tray 1381	8.434E-05	7.01E-03	5.91E-07

Fire Compartment 1CC3a Fire Damage Scenarios								
Scen #	**Source Type**	**Source Identification (Section if SWGR, Bus or MCC)**	**Supp (Y/N)**	**Safe Shutdown Equipment Damaged**	**SSD Trays and/or Conduit Damaged**	**F1'**	**P2'**	**F3'**
		5) MCC EF1D08 (R24-S028), MCC EF1D09 (R24-S036), and Alternate Power Supply to EF2D11 (R24-S048)			Tray 1608			
12	480 V Bus	6) ECC Pump B P42-C001B, CCCW Pump B P47-C001B, Normal Battery Charger EFD-1-B (R42-S008), and MCC EF1D07 (R24-S026)	Y	480 V Bus EF-1-D	Tray 272 Tray 1317 Tray 1381 Tray 1608	4.217E-05	FDS 0 7.01E-03	2.96E-07
13	480 V Bus	7) Instr. Compt. EF1D01, Main supply breaker from EH-12, and Instr. Compt. EF1D02	Y	480 V Bus EF-1-D	Tray 272 Tray 1317 Tray 1608	4.217E-05	FDS 0 7.01E-03	2.96E-07
14	480 V MCC	EF1C07 (R24-S023)	N	EF1C07	None	6.53E-04	FDS 3 NEG	NEG
15	480 V MCC	EF1C08 (R24-S024)	N	EF1C08	None	1.875E-04	FDS 3 NEG	NEG
16	480 V MCC	EF1D09 (R24-S036)	N	EF1D09	None	2.363E-05	FDS 3 NEG	NEG
17	480 V MCC	EF1C09 (R24-S025)	N	EF1C09	Tray 273 Tray 1356	3.3E-05	FDS 0 7.01E-03	2.31E-07
18	480 V MCC	EF1D07 (R24-S026)	N	EF1D07	Tray 271 Tray 1321 Tray 1320 Tray 1371 Tray 1372 Tray 1373 Tray 1374 Tray 1375 Tray 1376 Tray 1377 Tray 1378	2.97E-04	FDS 1 2.59E-04	7.69E-08
19	480 V MCC	EF1D08 (R24-S028)	N	EF1D08	Tray 271 Tray 1321 Tray 1320 Tray 1384 Tray 1385	8.032-04	FDS 1 2.59E-04	5.49E-08
20	Post LOCA Power Supply	M51-S002	N	M51-S002	Tray 1317 Tray 272 1R33C5423B	4.875E-05	FDS 0 7.01E-03	3.42E-07
21	Junct. Boxes	N/A	N	None	None	0.000E-00	N/A	0.000E-00

Fire Compartment 1CC3a Fire Damage Scenarios								
Scen #	Source Type	Source Identification (Section if SWGR, Bus or MCC)	Supp (Y/N)	Safe Shutdown Equipment Damaged	SSD Trays and/or Conduit Damaged	F1'	P2'	F3'
22	RPS MG Set	N/A	N	None	None	2.753E-03	FDS 3 NEG	NEG
23	XFMR	R25-S034	N	None	None	9.753E-05	FDS 3 NEG	NEG
24	XFMR	R25-S5002	N	None	None	9.753E-05	FDS 3 NEG	NEG
25	XFMR	EHF-1-C (R23-S011)	N	All	All	9.753E-06	FDS 0 7.01E-03	6.82E-08
26	XFMR	EHF-1-C (R23-S011)	Y	480 V Bus EF-1-C	Tray 272 Tray 1317 Tray 1608	8.777E-05	FDS 0 7.01E-03	6.15E-07
27	XFMR	EHF-1-D (R23-S012)	N	All	All	9.753E-06	FDS 0 7.01E-03	6.82E-08
28	XFMR	EHF-1-D (R23-S012)	Y	480 V Bus EF-1-D	Tray 272 Tray 597 Tray 1317 Tray 1608 Tray 1649	8.777E-05	FDS 0 7.01E-03	6.15E-07
29	XFMR	R25-S027	N	None	None	9.753E-05	FDS 3 NEG	NEG
30	XFMR	R25-S035	N	None	None	9.753E-05	FDS 3 NEG	NEG
31	XFMR	R25-S128	N	None	None	9.753E-05	FDS 3 NEG	NEG
32	XFMR	R71-S076	N	None	None	9.753E-05	FDS 3 NEG	NEG
33	--	Transient and Welding/Cutting	N	All	All	6.876E-07	7.01E-03	4.829E-09

Table 4-5 - Grouping of Control Room Cabinets

Fire Damage (Loss of Control from Control Room)							
Cabinet Group	**Division I/II Systems except RCIC, FPCC Vent and ADS Valves**	**RCIC**	**SRVs**	**FPCC Vent Path**	**MSIV Vent Path**	**HPCS**	**Offsite Power**
1a 1b 1c 1d	Division I	**Group 1a** No damage **Group 1b** No damage **Group 1c** Div I **Group 1d** Damaged	Division I	**Group 1a** No damage **Group 1b** No damage **Group 1c** Div I **Group 1d** No damage	No damage	**Group 1a** No damage **Group 1b** Damaged **Group 1c** No damage **Group 1d** No damage	No damage
2a 2b 2c	Division II	No damage	Division II	**Group 2a** No damage **Group 2b** Div II **Croup 2c** Div II	No damage	No damage	**Group 2a** No damage **Group 2b** Damaged **Group 2c** No damage
3a 3b 3c	Division I & Division II	**Group 3a** No damage **Group 3b** Damaged **Group 3c** No damage	**Group 3a** No damage **Group 3b** Damaged **Group 3c** No damage	**Group 3a** No damage **Group 3b** Damaged **Group 3c** No damage	**Group 3a** No damage **Group 3b** No damage **Group 3c** Damaged	**Group 3a** No damage **Group 3b** Damaged **Group 3c** No damage	No damage
4a 4b	No damage	No damage	No damage	No damage		No damage	**Group 4a** No damage **Group 4b** Damaged

Table 4-6 - Cabinet Equipment Status and Fire Frequency Evaluation

Cabinet ID	Function (based on walkdown)	Cabinet Division (GEZ-7132)		RCIC	HPCS	FPCC Vent Path CR Cntrl	MSIV Vent Path	ADS/SRV CR Cntrl	Offsite Power Status	Cabinet Floor Area	Cabinet Fire Freq
		D1/D3	D2/D4								
CABINET GROUP 1A											
P628	ADS I	Div 1						Div 1		2	1.04E-04
P632	Leak Detection	Div 1								1	5.19E-05
P651	Control Rod Position Panel	Div 1								1	5.19E-05
P655	Aux Relay Div 1	Div 1								2	1.04E-04
P669	Neutron Monitoring	Div 1								3	1.56E-04
P881	Div 1 Cont. and DW Isolation Valve	Div 1,3								1	5.19E-05
P884	Post Accident Monitoring	Div 1								1	5.19E-05
										Total	5.71E-04
CABINET GROUP 1B											
P625	Division 3 Aux Relays	Div 3			Failed					1	5.19E-05
P873	Division 3 Aux Relays	Div 3			Failed					1	5.19E-05
										Total	1.04E-04
CABINET GROUP 1C											
P872	Aux Relays Division 1	Div 1				Div 1				3	1.56E-04
										Total	1.56E-04
CABINET GROUP 1D											
P621	Div 1 Aux Relays/RCIC	Div 1		Failed				Div 1		1	5.19E-05
P629	Div 1 Aux Relay LPCS, RHR A	Div 1		Failed				Div 1		4	2.08E-04
										Total	2.60E-04
CABINET GROUP 2A											
P610	Test		Div 2							1	5.19E-05
P618	Div 2 Aux Relay - RHR B&C		Div 2					Div 2		4	2.08E-04
P622	Div 2 Aux Relays		Div 2							2	1.04E-04
P631	ADS II		Div 2					Div 2		1	5.19E-05
P640	Test		Div 2							2	1.04E-04
P642	Leak Detection		Div 2							2	1.04E-04

Cabinet ID	Function (based on walkdown)	Cabinet Division (GEZ-7132) D1/D3	Cabinet Division (GEZ-7132) D2/D4	RCIC	HPCS	FPCC Vent Path CR Cntrl	MSIV Vent Path	ADS/SRV CR Cntrl	Offsite Power Status	Cabinet Floor Area	Cabinet Fire Freq
P652	1H13-P0652		Div 2							1	5.19E-05
P653	CRD Control Instrumentation		Div 2							1	5.19E-05
P654	Aux Relay Div 2		Div 2							1	5.19E-05
P670	Process Rad Monitor SRM & IRM Chnl B/F		Div 2							3	1.56E-04
P882	Containment Isolation		Div 2							1	5.19E-05
P885	Post Accident Monitoring		Div 2							1	5.19E-05
										Total	1.04E-03
CABINET GROUP 2B											
P870	Offsite Power/ BOP		Div 2			Div 2 Failed			Failed	9	4.67E-04
										Total	4.67E-04
CABINET GROUP 2C											
P871	B Relay Division 2		Div 2			Div 2 Failed				3	1.56E-04
										Total	1.56E-04
CABINET GROUP 3A											
P623	Auxiliary Relay	Div 1	Div 4							1	5.19E-05
P671	Neutron Monitoring	Div 3	Div 2							3	1.56E-04
P672	Rad Monitoring	Div 1	Div 4							3	1.56E-04
P680	Unit 1 Control Console	Div 1,3	Div 2,4							8	4.15E-04
P691	Div 1 RPS and Inst & Auxiliary Relay Panel	Div 1,3	Div 2,4							2	1.04E-04
P692	Div 2 RPS and Inst & Auxiliary Relay Panel	Div 1,3	Div 2,4							2	1.04E-04
P693	Div 3 RPS and Inst & Auxiliary Relay Panel	Div 1,3	Div 2,4							2	1.04E-04
P694	Div 4 RPS and Inst & Auxiliary Relay Panel	Div 1,3	Div 2,4							2	1.04E-04
P800	HVAC Control Panel	Div 1,3	Div 2							5	2.60E-04
P868	Analog Loop Division 2 Instrumentation Panel.	Div 1,3	Div 2							3	1.56E-04

Cabinet ID	Function (based on walkdown)	Cabinet Division (GEZ-7132) D1/D3	D2/D4	RCIC	HPCS	FPCC Vent Path CR Cntrl	MSIV Vent Path	ADS/SRV CR Cntrl	Offsite Power Status	Cabinet Floor Area	Cabinet Fire Freq
P869	Analog Loop Division 1 Instrumentation Panel.	Div 1	Div 2,4							3	1.56E-04
P877	Diesel Generator Bench Board	Div 1	Div 2							2	1.04E-04
P883	Post Accident Monitoring	Div 1	Div 2							1	5.19E-05
P904	Common HVAC	Div 1	Div 2							3	1.56E-04
P969	Common Long Response Control Panel	Div 1	Div 2							3	1.56E-04
P970	Common Analog Loop Instr. and Aux Relay Panel	Div 1	Div 2							2	1.04E-04
										Total	2.34E-03
CABINET GROUP 3B											
P601	Emergency Core Cooling Bench Board	Div 1, 3	Div 2, 4	Failed	Failed	Div 1/2		Div 1/2		7	3.63E-04
										Total	3.63E-04
CABINET GROUP 3C											
C22-P001	Redundant Reactivity Control	Div 1	Div 2				Failed			3	1.56E-04
C22-P002	Redundant Reactivity Control	Div 1,3	Div 2,4				Failed			3	1.56E-04
										Total	3.11E-04
CABINET GROUP 4A											
P5001	Control Room Communications Panel									1	5.19E-05
P600	Rad Monitoring									2	1.04E-04
P604	Rad Monitoring									1	5.19E-05
P612	FW/ Recirculation									3	1.56E-04
P613	NSSS Inst.									1	5.19E-05
P614	NSS Recorder									2	1.04E-04
P619	Jet Pump									1	5.19E-05
P630	Annunciator Logic Panel									8	4.15E-04
P634	Recirc/Flow Ctl.									2	1.04E-04
P637	Steam Bypass									2	1.04E-04

Cabinet ID	Function (based on walkdown)	Cabinet Division (GEZ-7132) D1/D3	D2/D4	RCIC	HPCS	FPCC Vent Path CR Cntrl	MSIV Vent Path	ADS/SRV CR Cntrl	Offsite Power Status	Cabinet Floor Area	Cabinet Fire Freq
P643	SAS									1	5.19E-05
P703	Not shown on GEZ 7132										0.00E+00
P802	Fire/Security									3	1.56E-04
P803	Rad Monitoring									1	5.19E-05
P804	Rad Monitoring									3	1.56E-04
P805	Cabinet at center of horseshoe									1	5.19E-05
P807	Main Transformer Relay									1	5.19E-05
P821	EHC									4	2.08E-04
P822	Turbine									1	5.19E-05
P823	Balance of Plant									1	5.19E-05
P840	RFPT Governor									1	5.19E-05
P842	Balance of Plant									1	5.19E-05
P845	Off Gas									3	1.56E-04
P864	Analog Loop B Intrument Panel									2	1.04E-04
P865	Analog Loop A Instrument Panel									4	2.08E-04
P866	Auxiliary Relay A									5	2.60E-04
P867	Auxiliary Relay B									5	2.60E-04
P874	SAS									3	1.56E-04
P902	Rad Monitoring									1	5.19E-05
P906	Rad Monitoring									1	5.19E-05
P907	Rad Monitoring									1	5.19E-05
										Total	3.43E-03
CABINET GROUP 4B											
P808	Generator Relay								Failed	3	1.56E-04
P809	Startup Transformer Relay								Failed	1	5.19E-05
P810	Auxiliary Transformer Relay								Failed	1	5.19E-05
P811	Electrical System								Failed	1	5.19E-05
										Total	3.11E-04

Cabinet ID	Function (based on walkdown)	Cabinet Division (GEZ-7132) D1/D3	D2/D4	RCIC	HPCS	FPCC Vent Path CR Cntrl	MSIV Vent Path	ADS/SRV CR Cntrl	Offsite Power Status	Cabinet Floor Area	Cabinet Fire Freq
									GRAND	TOTAL	9.50E-03

Table 4-7 - Screening of Internal Cabinet Fires

Cabinet Group	Summary of CR Controls Damaged	Frequency of fire in Cabinet Group (F1')	Conditional Core Damage Frequency (P_2')	Core Damage Frequency (F3)'	Fire Damage State (see Table 4-9)
1a	MFW/PCS Division I(1) ADS Valve Div I	5.71E-04	1.08E-04	6.17E-08	FDS 12
1b	MFW/PCS Division I(1) HPCS	1.04E-04	4.47E-03	4.65E-07	FDS 13
1c	MFW/PCS Division I(1) FPCC Vent Div I	1.56E-04	1.08E-04	1.68E-08	FDS 12
1d	MFW/PCS Division I(1) RCIC	2.60E-04	5.26E-04	1.36E-07	FDS 8
2a	MFW/PCS Division II(1) ADS Valve Div II	1.04E-03	4.89E-04	5.09E-07	FDS 14
2b	MFW/PCS Division II(1) Offsite Power FPCC Vent Div II	4.67E-04	7.59E-02	3.54E-05	FDS 16
2c	MFW/PCS Division II(1) FPCC Vent Div II	1.56E-04	3.00E-03	4.68E-07	FDS 15
3a	MFW/PCS Division I & II(1)	2.65E-03	5.33E-04	1.25E-06	FDS 11
3b	MFW/PCS Division I & II(1) ADS Vlvs Div I & II FPCC Vent Div I& II RCIC	3.63E-04	1.91E-02	6.93E-05	FDS 7
3c	MFW/PCS Division I & II(1) MSIV vent path	3.12E-04	2.79E-05	8.70E-09	FDS 19

Cabinet Group	Summary of CR Controls Damaged	Frequency of fire in Cabinet Group (F1')	Conditional Core Damage Frequency (P_2')	Core Damage Frequency (F3)'	Fire Damage State (see Table 4-9)
4a	MFW /PCS	3.43E-03	2.75E-05	9.43E-08	FDS 5
4b	MFW/PCS Offsite Power	3.11E-04	1.03E-03	3.20E-07	FDS 1

(1) Excludes RCIC, ADS and FPCC and MSIV containment vent paths. These are identified separately.

Table 4-8 - Control Room Fire Damage States

Fire Damage State Definition	Fire Damage Designation
Loss of Offsite Power	FDS 1
Loss of Offsite Power and Div II (including FPCC/ADS), w/ no RCIC damage	FDS 16
Loss of Offsite Power and Div II (including FPCC/ADS), w/ RCIC damage	FDS 2
CR evacuation w/ LOOP, w/ Div II damage	FDS 17
CR evacuation w/ LOOP, w/ no Div II damage	FDS 3
Loss of Div II (including FPCC / ADS) w/ no RCIC damage	FDS 15
Loss of Div II (including FPCC / ADS) w/ RCIC damage	FDS 4
None	FDS 5
Loss of MSIV vent path	FDS 19
CR evacuation w/ no Div II damage	FDS 18
CR evacuation w/ Div II damage	FDS 6
Loss of Div I/II (including FPCC/ADS) w/ RCIC damage	FDS 7
Loss of Div I (including FPCC/ADS) w/ RCIC damage	FDS 8
Loss of Div I and HPCS (including FPCC/ADS) w/ no RCIC damage	FDS 13
Loss of Div I and HPCS (including FPCC/ADS) w/ RCIC damage	FDS 9
Loss Div II (including ADS) w/ no RCIC damage	FDS 14
Loss Div II (including ADS) w/ RCIC damage	FDS 10
Loss of Div I/II (not FPCC/ADS) w/ no damage to RCIC	FDS 11
Loss of Div I (including FPCC/ADS) w/ no damage to RCIC	FDS 12

Table 4-9 - Results of Control Room Analysis

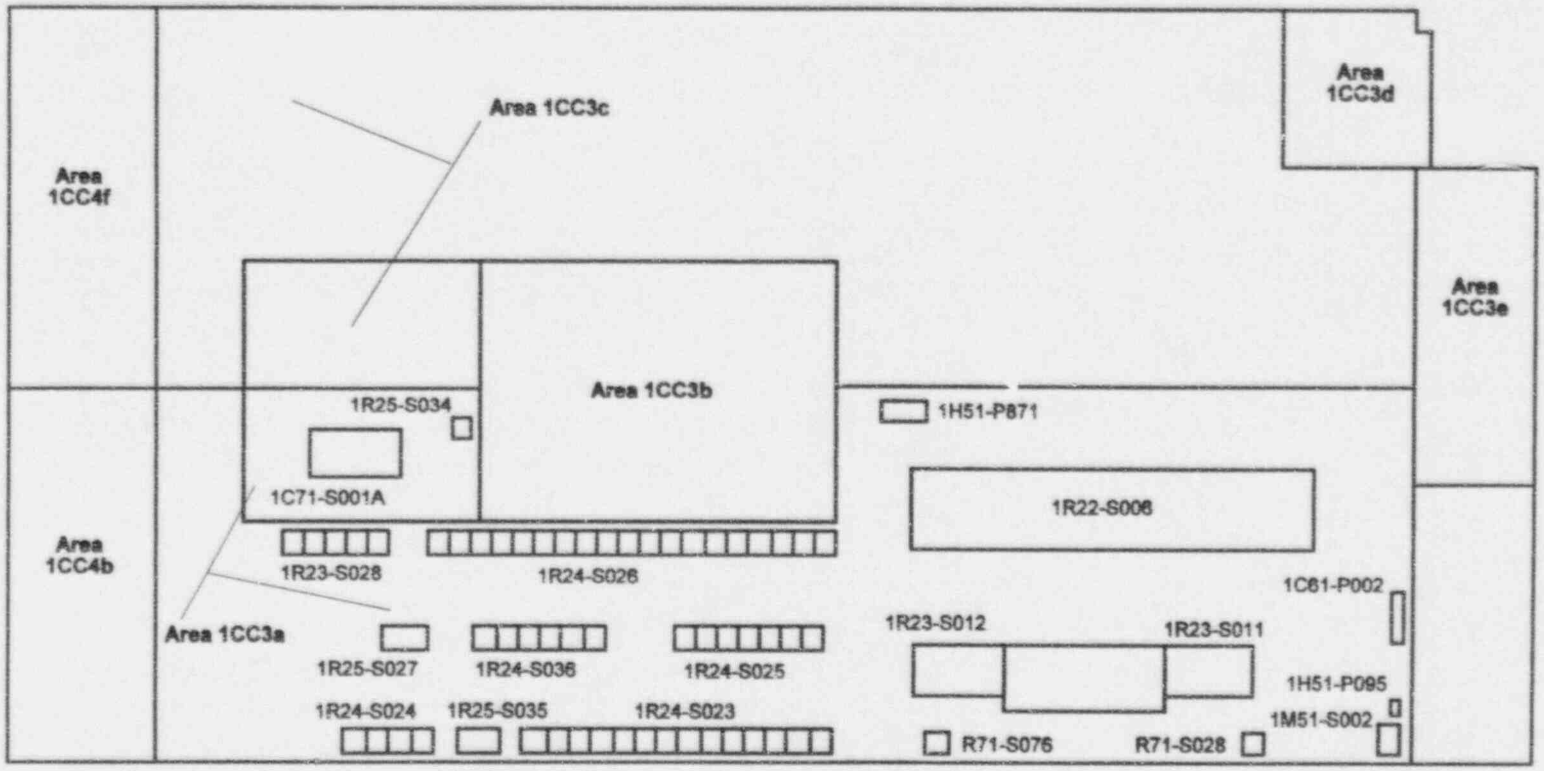
Cabinet	FIRE DAMAGE STATE CONTRIBUTIONS																			Ignition
Group	FDS1	FDS2	FDS3	FDS4	FDS5	FDS6	FDS7	FDS8	FDS9	FDS10	FDS11	FDS12	FDS13	FDS14	FDS15	FDS16	FDS17	FDS18	FDS 19	Freq
Group 1A																				
-All Cabinets						1.14e-6						5.70e-4								5.71e-4
Group 1B																				
-All Cabinets						2.08e-7							1.03e-4							1.03e-4
Group 1C																				
-All Cabinets						3.12e-7						1.56e-4								1.56e-4
Group 1D																				
-All Cabinets						5.20e-7		2.59e-4												2.60e-4
Group 2A																				
-All Cabinets														1.04e-3				2.08e-6		1.04e-3
Group 2B																				
-P870	4.95e-5				3.61e-4										5.19e-5	4.88e-6	1.04e-7	7.27e-7		4.68e-4
Group 2C																				
-P871															1.56e-4			3.12e-7		1.56e-4
Group 3A																				
-P680					4.14e-4													8.15e-7		4.15e-4
-P691											1.03e-4							2.08e-7		1.03e-4
-P692											1.03e-4							2.08e-7		1.03e-4
-P693											1.03e-4							2.08e-7		1.03e-4
-P694											1.03e-4							2.08e-7		1.03e-4
-P800					9.90e-5						5.99e-5	4.95e-5		5.19e-5				5.20e-7		2.61e-4
-P868											1.06e-4			4.95e-5				3.12e-7		1.56e-4
-P877											4.88e-6	4.95e-5		4.95e-5				2.08e-7		1.04e-4
-P904					4.95e-5						4.88e-6	5.19e-5		4.95e-5				3.12e-7		1.56e-4

Cabinet	FIRE DAMAGE STATE CONTRIBUTIONS																			Ignition
Group	FDS1	FDS2	FDS3	FDS4	FDS5	FDS6	FDS7	FDS8	FDS9	FDS10	FDS11	FDS12	FDS13	FDS14	FDS15	FDS16	FDS17	FDS18	FDS 19	Freq
-P623											5.18e-5							1.04e-7		5.19e-5
-P869											1.06e-4	4.95e-5						3.12e-7		1.56e-4
-All Other Cbnts				6.25e-4	1.25e-6															6.26e-4
Group 3B																				
-P601				9.90e-5	4.95e-5		6.16e-5	9.90e-5					5.44e-5					7.28e-7		3.64e-4
Group 3C																				
All cabinets						6.24e-7													3.12e-4	3.13e-4
Group 4A																				
-All Cabinets					3.42e-3	6.84e-6														3.43e-3
Group 4B																				
-All Cabinets	3.10e-4		6.20e-7																	3.11e-4
Total FDS Freq	3.60e-4	0.00E+00	6.20e-7	9.90e-5	5.02e-3	1.09e-5	6.16e-5	3.58e-4	0.00E+00	0.00E+00	7.45e-4	9.26e-4	1.57e-4	1.24e-3	2.08e-4	4.88e-6	1.04e-7	7.26e-6	3.12e-4	9.20e-3
FDS CCDP	1.03e-3	7.62e-2	3.83e-1	3.06e-3	2.75e-5	2.79e-1	1.91e-2	5.26e-4	2.04e-2	6.18e-4	5.33e-4	1.08e-4	4.47e-3	4.89e-4	3.00e-3	7.59e-2	3.83e-1	3.23e-1	2.79e-5	
Total Fire CDF	3.70e-7	0.00E+00	2.37e-7	3.03e-7	1.38e-7	3.04e-6	1.18e-6	1.88e-7	0.00E+00	0.00E+00	3.97e-7	1.00e-7	7.04e-7	6.07e-7	6.24e-7	3.70e-7	3.98e-8	2.35e-6	8.70e-9	1.06e-5

Table 4-10 - Summary of Overall Fire Analysis Results for Areas which Did Not Initially Screen

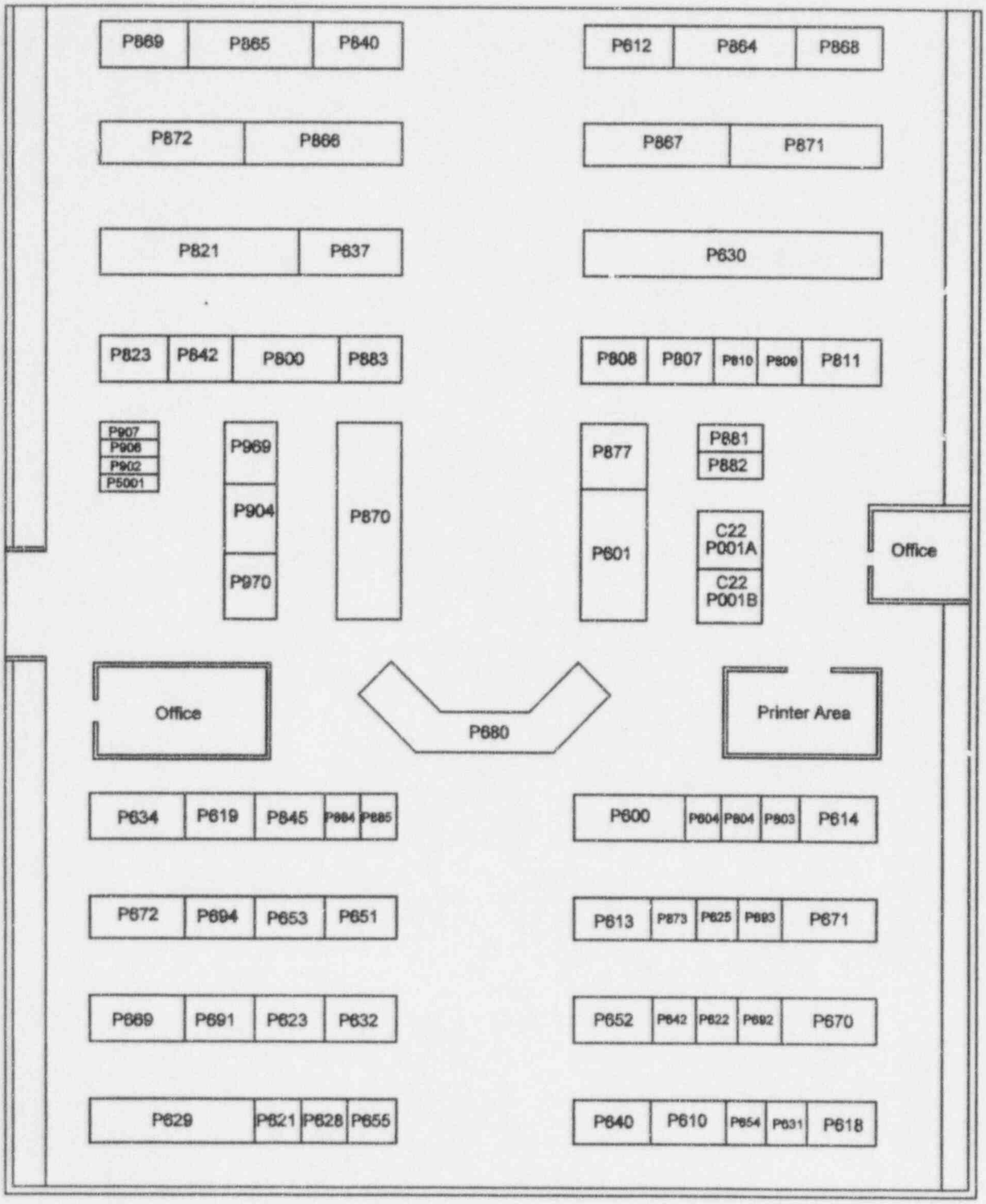
Compartment	Description	Fire Induced CDF Based on Screening	Final Fire Induced CDF
AB1f	HPCS Pump Room	1.43E-06	8.56E-07
1AB2	Unit 1 RCIC and HVAC Room	6.18E-06	2.40E-07
1CC-3a	Unit 1, Division 2 Switchgear Room	5.60E-05	1.05E-05
1CC-3c	Unit 1, Division 1 Switchgear Room	3.79E-06	1.98E-06
1CC-4b	Unit 1 Div 2 Cable Chase	2.03E-05	8.55E-07
CC1	Control Complex El. 574'-0"	1.17E-04	2.03E-06
CC2/1	Northwest Cntl. Cmplx El. 599'-0"	2.71E-04	3.92E-07
CC2/2	Corridor Control Cmplx. El. 599'-0"	3.57E-04	1.64E-07
CC2/4	Cntrl Cmplx El. 599'-0" - NCC Pump Room	1.05E-03	5.82E-07
CC2/5	Cntrl Cmplx El. 599'-0" - NCC HX Room	1.00E-05	3.27E-07
DG1e	D/G Bldg Low Level Rad Waste Room	5.08E-06	2.79E-07
ESW1a	ESW Pumphouse - ESW Pump Room	1.67E-05	7.92E-07
IB-2	Intermediate Building, El. 599'-0"	2.55E-04	8.04E-07
1TB	Unit 1 Turbine Building	7.87E-05	1.30E-06
1CC-5a	Unit 1 Control Room	9.97E-05	1.06E-05
1TPC/1	Unit 1 Turbine Power Complex Switchgear Room	3.28E-06	3.28E-06
FH3	Fuel Handling Building El. 620'	1.634E-06	1.63E-06

Figure 4-1 - Layout of Unit 1, Division 2 Switchgear Room



Unit 1 Control Complex El. 620

Figure 4-2 - Layout of Control Room



Unit 1 Control Room

Figure 4-3 - Example of Control Room Cabinet Fire Growth Event Tree

Fire in Cabinet 870	Fire Location Probability	Fire Extinguished Prior to Inter-Bay Propagation	Fire Extinguished Prior to MCR Evacuation	FDS	Frequency
				1	4.95E-05
	Bay A			16	2.44E-06
	0.111				
		4.9E-02	2.0E-03/4.9E-02	17	1.04E-07
			4.1E-02	15	4.95E-05
	Bay B			16	2.44E-06
	0.111				
		4.9E-02	2.0E-03/4.9E-02	17	1.04E-07
			4.1E-02	5	4.95E-05
4.67E-04/yr	Bay C			15	2.44E-06
	0.111				
		4.9E-02	2.0E-03/4.9E-02	18	1.04E-07
			4.1E-02	5	3.11E-04
	Bays D-J				
	0.667			18	6.23E-07
			2.0E-03		

FDS = Fire Damage State (See Table 3-1 for definitions)

Fire Growth Event Tree For Cabinet 1H13-P870

5 HIGH WINDS, FLOODS, AND OTHERS

The individual plant examination of external events (IPEEE) requires an evaluation of the impact on the plant of hazards that are external to it. The hazards are classified into seismic, fire, and other. In NUREG-1407,[5-1] the conclusion was reached that, of the other external events, only the following need be considered on a plant specific basis; high winds, external floods, and transportation and nearby facility accidents. However, there is also a requirement that there be a review performed to confirm that there are no external hazards unique to the site, that would invalidate the conclusions of NUREG-1407. This section of the report documents the analysis of other external hazards for the Perry Nuclear Power Plant.

The approach used follows the method described in NUREG/CR-4839[5-2] by performing an initial screening analysis, followed by bounding or detailed analyses as necessary.

Section 5.1 provides a brief description of PNPP, and Section 5.2 describes the initial screening analysis approach, and presents the conclusions of that screening, which confirms the conclusions of NUREG-1407. Sections 5.3, 5.4, and 5.5 discuss high winds, floods, and transportation and nearby facility accidents respectively. Conclusions are given in Section 5.6, and references in Section 5.7.

5.1 Generic Plant Description

5.1.1 Site Description

The Perry Nuclear Power Plant (PNPP) consists of a single operating unit located in Lake County Ohio approximately 7 miles northeast of Painesville and 35 miles northeast of Cleveland. The plant site is located along the southeastern shoreline of Lake Erie on an ancient lake plain approximately 50 feet above the lake low water elevation. The site and its environs consist mainly of woodland and former nursery lands. The total area of the site is approximately 1,100 acres and is relatively flat. The land has a gentle slope toward the lake and is crossed by small streams which drain into the lake. The main plant building are located about 800 feet from the toe of a 45 foot high steep bluff that forms the shoreline.

PNPP is a 3,579 MW_{th} General Electric BWR/6 with a Mark III containment. The balance of plant systems were engineered by Gilbert Commonwealth, Inc. The unit started commercial operation in November 1987.

5.1.2 Identification of Structures, Systems and Components Potentially Susceptible to External Events

The objective of performing an external events analysis is the identification of structures, systems or components which are susceptible to damage, and which, if damaged, could lead to a loss of capability to safely shut down the reactor. The majority of the safety related equipment is protected by being enclosed in the safety-related Class I structures which are:

- Reactor Building
- Auxiliary Building
- Control Complex
- Fuel Handling and Intermediate Buildings
- Emergency Service Water Pumphouse
- Diesel Generator Building
- Intake Structure
- Radwaste Building
- Off-Gas Building

5.2 Screening or External hazards

5.2.1 Description of Approach

The objective of the screening analysis is to provide confirmation of the NUREG-1407 conclusion that there are no hazards unique to the site that require evaluation, other than those posed by high winds, external floods, and transportation and nearby facility accidents.

The PRA Procedures Guide[5-3] provides an exhaustive list of potential external hazards which provides the starting point for the analysis. The screening is performed by reviewing the information on the site region and plant design to identify external events that are applicable using the screening criteria below. The data in the Updated Safety Analysis Report (USAR) on the geologic, seismologic, hydrologic, and meteorological characteristics of the site region as well as present and projected industrial activities, i.e., the building of a reservoir, increases in the number of flights at an airport, construction of a road that carries explosive materials, etc., in the vicinity of the plant are reviewed for this purpose. The set of screening criteria has been formulated to minimize the possibility of omitting significant risk contributors while reducing the amount of detailed analyses to manageable proportions. The following screening criteria have been adopted from those given in the PRA Procedures Guide.

An external event is excluded if:

- The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate the resistance of plant structures and systems to a particular external event. For example, it is shown by Kennedy, Blejwas, and Bennett[5-4] that safety-related structures designed for earthquake and tornado loadings in Uniform Building Code Zone 1 can safety withstand a 3.0 psi static pressure from explosions. Hence, if it is demonstrated that the overpressure resulting from explosions at a source (e.g., railroad, highway or industrial facility) cannot exceed 3 psi, these postulated explosions need not be considered.

- The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events. For example, an event may be excluded whose mean frequency of occurrence is less than some small fraction of those for other events. In this case, the uncertainty in the frequency estimate for the excluded event is judged as not significantly influencing the total risk.

- The event cannot occur close enough to the plant to affect it. This is also a function of the magnitude of the event. Examples of such events are landslides, volcanic eruptions and earthquake fault ruptures.

- The event is included in the definition of another event. For example, storm surges and seiches are included in external flooding; the release of toxic gases from sources external to the plant is included in the effects of either pipeline accidents, industrial or military facility accidents, or transportation accidents.

In addition to these, another criterion is added.

- The event is slow in developing and there is sufficient time to eliminate the source of the threat, or to take precautionary measures to minimize the consequences.

Each of the potential other external hazards listed in the PRA Procedures Guide was reviewed with respect to the above criteria and determined to meet one or more of these screening criteria as summarized in Table 5-1.

In accordance with the recommendation of NUREG-1407, a plant walkdown was performed to confirm the conclusions of the paper study, i.e., that only high winds, external floods, and nearby facility and transportation accidents need be investigated in detail. Some of the general observations made during the walkdown include: the plant site is generally very clean with no significant accumulation of objects that could become airborne during the occurrence of a tornado or other extreme wind such as construction steel, pipes, etc.; there are large open areas with no possibility for ponding during heavy downpours; plant structures containing safety related structures have relatively few openings and penetrations that would be susceptible to external hazards; the majority of penetrations, such as doors, air intakes and exhaust vents, of safety related structures are protected by concrete shields which prevent a direct hit from an airborne missile, or by gratings; sandbags are provided outside doorways into safety related structures for use in the event that severe flooding occurs. Therefore, the plant is generally well protected against the external events analyzed in detail.

5.2.2 Results of Screening Analysis

Each of the external hazards listed in the PRA Procedures Guide was reviewed. Based on information in the USAR, and on the basis of the walkdown, it was determined that the conclusions of NUREG-1407 were valid for PNPP, namely that there are no known plant-unique other external events that pose a significant threat of severe accidents within the context of the NUREG-1407 screening approach. The next three sections discuss the potential for significant impact of the three hazards identified by NUREG-1407 for plant-specific evaluation, namely, high winds, external floods, and transportation and nearby facility accidents.

5.3 High Winds

5.3.1 Straight Winds

The evaluation of high wind loading for the PNPP was performed by comparing the PNPP design basis, obtained from Section 3.3.1 of the USAR against the acceptance criteria of the USNRC Standard Review Plan (SRP)[5-5] Section 3.3.1, "Wind Loadings."

The first acceptance criteria requires that the wind used in the design should be the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data has been accumulated. The basic wind speed used in the design of safety class structures is 90 mph at 30 feet above grade, and is in agreement with Figure 2 of ANSI A58.1. Increased wind velocities were considered in the case of wind being channeled through the cooling towers and impinging on safety class structures.

Based on the USAR[5-6] information (see Section 2.3.1.2.3), the annual fastest mile wind (defined as the one-mile passage of wind with the fastest speed including all meteorological phenomena except tornadoes) data at Cleveland for the 30 year period from 1948 to 1977 were used to determine predicted extreme wind speeds for the PNPP site for recurrence intervals of 50 (70 mph) and 100 (74 mph) years. These wind speeds are in excess of any winds actually observed in Cleveland or in the Perry region, as noted in the Nineteenth Annual Report of the Meteorological Program at the Perry Nuclear Power Plant,[5-7] which concludes that the 90 mph wind is still valid as a design basis.

The second acceptance criteria states that the approved parameter values must be in accordance with the SRP Sections 2.3.1 and 2.3.2. The specific criteria necessary to meet the requirements are:

SRP Section 2.3.1

a) The description of the general climate of the region should be based on standard climatic summaries compiled by NOAA. Consideration of the relationships between regional synoptic-scale atmospheric processes and local meteorological conditions should be based on appropriate meteorological data.

   The values provided in USAR Section 2.3.1.2.3 for 50 and 100 year recurrence maximum wind speeds are consistent with those published by the U.S. Department of Commerce and the NOAA for the Perry region.

b) Data on severe weather phenomena should be based on standard meteorological records from nearby representative National Weather Service (NWS), military or other stations recognized as standard installations which have long periods on record. The applicability of these data to represent site conditions during the expected period of reactor operation must be substantiated.

As noted in USAR Section 2.3.1.2.3, the data on peak wind gust at various heights are based on the NWS station at Cleveland and other recognized stations.

c) Operating basis velocity (fastest mile of wind) should be based on the standard published by the ANSI with suitable corrections for local conditions.

The basic wind speed of 90 mph at 30 feet above grade used in the design of safety class structures is in agreement with Figure 2 of ANSI A58.1 which gives 80-90 mph as the fastest mile wind speed expected in a 100 year interval. The 90 mph is higher than the predicted fastest mile wind from meteorological data.

SRP Section 2.3.2

The acceptance criteria require the information regarding meteorological conditions and phenomena meet the requirement of the 10 CFR Part 50, Appendix A, General Design Criterion 2 and 10 CFR Part 100, 100.10(c).

The information provided in USAR meets the requirement of GDC 2 and 10 CFR Part 100.

The third acceptance criterion indicates that the use of procedures delineated in ANSI A58.1, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures" to transform the wind velocity into an effective pressure are acceptable.

The USAR Section 3.3.1.2 provides a discussion on the determination of applied forces. According to the USAR information, the wind pressures corresponding to the basic wind speed were based on Tables 5 and 6 of ANSI A58.1. The pressure values indicated consider the effects of height, terrain and gusts. In particular, the following observations can be made:

- The maximum velocity pressure, based upon the maximum wind velocity, was determined to be consistent with the formula provided in the SRP Section 3.3.1 (USAR Section 3.3.1.2).

- The velocity pressure, in accordance with the SRP acceptance criteria was determined at various heights based on Table 5 and Table 6 of ANSI A58.1 (USAR 3.3.1.2).

- The effective pressures for structures depending on geometry and physical configuration were determined based on Sections 6.4, 6.5 and 6.6 of ANSI A58.1 in accordance with SRP Section 3.3.1.

Since the SRP acceptance criteria are satisfied, the Perry Nuclear Power Plant has been designed in accordance with the intent of the Standard Review Plan with respect to the impact of high winds.

5.3.2 Tornadoes

The plant site lies within Region I for determining design basis tornadoes according to Regulatory Guide 1.76. Region I has the following tornado characteristics:

Maximum wind speed	360 mph
Rotational wind speed	290 mph
Translational speed	70 mph max., 5 mph min.
Radius of maximum rotational speed	150 ft
Pressure drop	3.0 psi
Rate of pressure drop	2.0 psi/sec

The tornado strike probability is 0.00106/year (USAR Section 2.3.1.2.1). Structures, systems, and components whose failure, due to design wind loading, tornado wind loading, or associated missiles, could prevent safe shutdown of the reactor are protected from such failure by one of the following methods:

- The structure or the component is designed to withstand design wind, tornado wind and tornado generated missiles.

or

- The system or components are housed within a structure which is designed to withstand design wind, tornado wind and tornado generated missiles. (USAR Section 3.3)

The plant design conforms with the standard review plan criteria and hence tornadoes are not considered a significant hazard. The most likely damage by a tornado would be a loss of offsite power with a long recovery time. This is already included in the PRA model.

5.3.3 Walkdown

A walkdown was performed to identify if there was any potential for missiles to be generated that could impact safety related equipment. The principal structures are designed against tornado missile impact, but there are doorways and penetrations that could provide weak points in the barriers. Generally for the safety related structures, the doorways are recessed so that a direct missile impact is highly improbable. Similarly, the air intakes and exhausts are protected, either by concrete shields, or gratings. The doorways into the auxiliary building and the fuel handling building are, however, not protected against missile strike. In both structures there is an instrument air tank located within the structures and in the vicinity of a personnel door. However, the simultaneous loss of both air tanks would require a simultaneous missile strike in two locations on small targets, and this is considered to be an event of extremely low probability. Thus it was concluded that the likelihood of missiles penetrating safety related structures, and damaging safety related equipment such that there could be significant consequences, is low enough to be insignificant. The walkdown also confirmed that the potential concerns raised in IEN 93-53, Supplement 1[5-8] about the potential for failures of non-safety equipment leading to failures of safety-related equipment, were not an issue at PNPP.

5.4 External Floods

5.4.1 General Plant Description

The plant site is located along the southeastern shoreline of Lake Erie approximately 50 feet above lake low water datum. The site is approximately 1,100 acres in size and relatively flat. The land has a very gentle slope toward the lake and is incised by small streams which drain into the lake.

The loss of switchyard and/or transformers by flooding would cause a loss of offsite power to the plant. The loss of offsite power is covered by the Perry PRA. Therefore, the effects of external flooding on the switchyard and various transformer sites is not considered here.

5.4.2 Sources of External Flooding

There are four potential flooding sources at the Perry site: Lake Erie, intense local precipitation, and two small streams which border the site to the east and south. In addition, the impact of flooding due to Service Water System (SWS) or Circulating Water System (CWS) fiberglass pipe rupture is also investigated.

Lake Erie is located to the north of the plant site. There are no safety-related structures within 380 feet of the lake shoreline. The lake will not influence the surface water characteristics of the site since the mean lake level is in excess of 40 feet below plant grade.

Flooding from Lake Erie is extremely improbable. No records of flooding in the plant area of the site from Lake Erie exist. Final grade elevations in the immediate plant area vary from 617 to 620 feet (USGS). This is about 45 feet above the maximum monthly mean lake level of 575.4 feet (USGS). Runup occurring coincidentally with the probable maximum setup would extend to about Elevation 607.9 feet on the bluff at the lake shore. This runup would still be about 12 feet below the nominal 620 foot (USGS) plant grade elevation.

The site soil is relatively permeable and contains sand layers in the upper reaches of the drainage area. The ground surface in much of the catchment area is forested with a heavy mulch ground cover. Due to the flat terrain, permeable upper soil layers, and the small catchment areas, it is unlikely that this location has ever been subjected to flooding or is likely to experience severe flooding from lake water in the future.

Two nameless, parallel steams run close to the plant area. The larger has a drainage basin of 7.16 square miles and runs northwestward within 1,000 feet of the southwest corner of the plant.[5-6] The smaller stream, which as a drainage area of only 0.76 square mile, borders the plant area to the east and northeast. The safety-related structures of the plant are located within the drainage basin of the small stream.

No records of flooding in the plant area of the site exist from the two streams draining the coastal watershed. However, as a result of probable maximum flooding (PMF) the level in the two streams can increase and if not contained can cause flooding hazard to the plant.

A new PMF value has been estimated using the latest probable maximum precipitation (PMP) estimates[5-9] where a 30.5% increase in rainfall over one square mile was calculated to correspond to about 4.26% increase in PMF. The calculation furthe showed that for the case of major and minor streams, the flood flow caused by PMF will be contained.

Therefore, it can be concluded that the two streams adjacent to the plant site would not cause flooding hazard to the plant. The potential for extreme precipitation to result in standing water is discussed in the next section.

5.4.3 Flooding Caused by Probable Maximum Precipitation (PMP)

There are two concerns associated with extreme precipitation: water ingress into critical structures as a result of standing water on the site, and increased loading due to roof ponding.

The PMP is defined as "the theoretically greatest depth of precipitation for a given duration that is physically possible over a particular drainage area." Thus, if it can be demonstrated that the PMP will not cause problems, it can be assumed that extreme precipitation is not a cause for concern. The evaluation includes an assessment of the National Weather Service's (NWS) new PMP criteria.[5-9] The revised NWS PMP criteria result in higher precipitation intensities over shorter time intervals and smaller areas. As a result of this PMP, greater roof ponding loads and higher site flooding levels may be expected.

Based on Reference 5-9, Figure 24, the 1-hour, 1-square mile PMP for the Perry site is about 17.1 inches. From Figure 23, Reference 5-9, the 1 to 6 hour ratio of precipitation for the Perry site is about 0.68. This ratio corresponds to 68 percent of precipitation occurring during the first hour and the remaining 32 percent occurring during the 2nd, 3rd up to the 6th hour. The total 6 hour PMP rate for the Perry site is about 25.1 inches.

Evaluations of safety related structures' roof ponding and plant surface flooding are provided in Sections 5.4.3.1 and 5.4.3.2 respectively.

5.4.3.1 Roof Ponding

All of the roofs of the critical plant structures are enclosed by parapets that are generally on the order of 3'-6" in height. The exception is the ESW Pumphouse which is the only safety-related structure with a parapet height of only 9". To prevent ponding, reliance is made on roof drains and scupper overflows.

The design live load of safety related structures is about 84.5 psf, which corresponds to about 16.4 inches (84.5 psf / 62 psf x 12 in) of water. The roof drainage capacity for these structures is about 4.1 in/hr.[5-10] Thus, even if the scuppers are not taken into consideration the load caused by the water on the roofs is considerably less than the designed roof live load. In the event that the roof drains are blocked completely, the roof scuppers (4" by 8"), the lower edges of which are located about 9" above the surface of the roofs, will be sufficient to prevent the overloading of the roofs.

The only section of roof which was been identified as a potential outlier is that between the diesel generator building and the control complex. This section of roof is boxe' in with no scuppers. Drainage is via two large drains. In addition to direct precipitation, two scuppers e diesel generator building roof over the ventilation rooms, drain onto this enclosed section. If water accumulates on the enclosed section of roof to a height of 2 ft, it will begin to enter the three diesel generator building ventilation rooms via the air intakes. In addition, water could also enter the ventilation rooms through the gap under the 3 ft wide access doors. Within the ventilation rooms, if the flood depth reaches about 18", water will get into the horizontal dampers which would allow the water to drop into the diesel generator rooms at the 620 ft elevation below. There are no floor drains in the three rooms above the diesel generator rooms. The water from the ventilation ducts would fall onto control panels which can be assumed to fail the diesel generators. If the PMP is also assumed to fail the offsite power, this could lead to loss of all a.c. power increasing the potential for core damage.

However, the design rating of the roof drains in the enclosed section of roof, the flooding through the ventilation rooms' air intakes will not occur even with the PMP profile. Although some flooding could occur through the door gaps, this would be limited, and is unlikely to result in sufficient depth to cause a significant amount of water intrusion into the diesel generator rooms.

5.4.3.2 Surface Flooding

The major concern is the potential for water ingress into critical structures due to flooding above grade level caused by PMP. However, since the site is generally flat (the exception is the eastern side of the plant site, where it dips to lower elevation toward the site boundary), water level would be controlled by the runoff into this lower area.

The water level on the site was modeled as a flow over a weir, and is believed to be conservative. A conservative estimate of the standing water level caused by the PMP was determined by equating the discharge over the "weirs" equal to the rate of water falling onto the catchment area, and assuming the worst case scenario of complete blockage of the site storm drainage system and using peak discharge from the most intense hour of the PMP. This was used to determine the rate of water intrusion into the safety-related buildings through openings to the yard area. It is assumed that the water intrusion into the safety-related buildings can occur only through personnel/rollup doors and no water can be leaked in through the electrical or mechanical conduits (see Section 5.4.3.3). It should be noted that the buildings have a 6 inch curb at the entrances. For conservatism, the PMP was assumed to fall on fully saturated terrain, with 100 percent runoff. The head of water was calculated as being on the order of 3" during the first hour of intense precipitation, but falls off considerably in subsequent hours.

In order to examine the impact of the PMP on safety related structures, water ingress into the Auxiliary Building, Control Complex, Intermediate Building, and Diesel Generator Building was evaluated, using the above head of water. The Containment, Intake Structure, Radwaste Building, Fuel Handling Building and Off-Gas Building were not analyzed further since no critical equipment susceptible to external flooding are included in these buildings. The ESW structure was not evaluated since flooding of the building can not happen because the open floor in that structure prevents accumulation (floor gratings to the ESW forebay). In addition, the Intermediate Building was also screened out from further evaluation since, according to the internal flood analysis, no significant equipment at the grade level or at the lower level susceptible to the flooding was identified. Flooding is also not a concern in the Unit 1 Diesel Generator Rooms. An 8" barrier, for the purpose of containing fuel oil spills within the room, is installed in each of these rooms just inside the door leading to the corridor adjacent to the Control Complex and just inside the door to the outside.

The analysis of the flow into the structures, given the estimated head of water showed that the time to reach the critical flood height in each of the structures would be well in excess of an hour, by which time the precipitation rate is considerably reduced. The time required to reach the critical flood heights, coupled with the fact that there are provisions for sandbagging entry ways into the critical structures during periods of extreme precipitation, leads to the conclusion that this source of flooding is not a concern at PNPP.

5.4.3.3 Flooding Caused By Inleakage Through Conduits

Since Perry went into operation in 1987, there have been several instances in which water migration from outside the plant through electrical manholes has caused varying degrees of internal flooding. In response to these events, automatic sump pumps were provided in several manholes. In addition, conduits were sealed at the inboard end using Bisco SF60 rubberized sealant. Testing was performed to establish the water tight effectiveness of the seals. Similarly, water intrusion through the annular space between yard piping and embedded pipe sleeves in plant building walls situated below grade has been prevented by the use of link seals and Bisco rubberized sealant.

Thus, it can be concluded that the water leakage through conduits into buildings would not be a safety concern at Perry.

5.4.3.4 Flooding Caused By Inleakage Through Underdrains

The Perry plant site is situated in an area where the ground water table is within a few feet of finished grade for the plant. The plant underdrain system was designed and installed in a manner which alleviates significant hydrostatic head from building up on the plant structure walls that are situated below grade. The main objective of the pressure relief underdrain system is to ensure that the groundwater level around the primary plant structures do not exceed elevation 590.0 feet. Safety-related structures serviced by the underdrain system are designed to withstand all loading conditions at this maximum level.

The underdrain system consists of a porous concrete blanket, one foot thick, which underlies all of the structures of the nuclear island. The blanket is increased in thickness in some areas to incorporate a one foot diameter, porous concrete pipe. The pipe carries the collected water to nine individual pumps located in manholes. The pumps discharge into pipelines that vary from three to six inches in diameter, connect to the gravity discharge system, and drain to Lake Erie via the emergency service water pumphouse.

The system includes two discharge systems (pumping and gravity drain) which are designed to operate independently of each other. The seven underdrain pumps are set to maintain a water surface elevation between 566.0' and 568.0'. If the pumps fail to start or cannot keep up with the rising water level, then when the level reaches 568.5', a high water level alarm will sound in the control room and the two backup pumps will start to provide enough additional pumping discharge capacity. In the event both the underdrain service pumps and the backup pumps fail, the ground water level would rise until it reaches the gravity discharge system which is provided to ensure that the groundwater level around the nuclear island never exceeds elevation 590.0'. The gravity discharge system is designed to provide a redundant discharge system which incorporates a gravity outfall, having no active components and enough discharge capacity to prevent flooding caused by rising groundwater of main plant buildings. Therefore, it can be concluded that the flooding hazard due to rising groundwater level at the Perry plant site is insignificant.

5.4.4 Flooding Caused by Fiberglass Pipe Rupture

The Service Water System (P41) and the Circulating Water System (N71) underground piping at PNPP is fabricated from fiberglass reinforced plastic (FRP). Since Perry has experienced two failures of fiberglass piping in the yard area, the potential impact of these pipe failures to the safety of the plant while in operation is analyzed in this section. The fiberglass pipe breaks at PNPP occurred in the Service Water System (P41) and in the Circulating Water System (N71). In order to provide background on each system, the history of each system is briefly summarized below.

5.4.4.1 Background Information

5.4.4.1.1 Service Water System (P41)

The Service Water System is constructed of various sizes of hoop wound FRP pipe, ranging from 6 inches in diameter to 54 inches in diameter, fabricated by Owens-Corning Fiberglass Corporation. Although the plant went on-line in 1985, the Service Water System went into service in 1981 and operated almost continuously until the failure of a pipe on the supply side piping in March of 1993. The P41 piping failure occurred underground. The flooding emanated from a 30" pipe approximately 50 feet south of the Water Treatment Building. However, no area of the plant at or below the grade level adjacent to the flood containing safety related equipment was significantly affected by the flooding. Flood levels in these areas and were well within the existing USAR limits. Areas adjacent to the flood point with large access doors (rollup doors) received more water than other areas, however, these are areas with no major equipment needed for plant operation.

5.4.4.1.2 Circulating Water System (N71)

The circulating water system was fabricated of helically wound FRP pipe made by CorBan Industries. The line break occurred above ground. A 36" diameter pipe elbow on the supply line leading to the Unit 1 auxiliary condenser ruptured north of the Heater Bay in December 1991. Water flooded the ground surface and also entered manhole #20 of the plant underdrain system which had been left open. In addition, some water also entered the electrical manholes in the flood area and small amounts of water entered buildings in the vicinity of the break through doorways and hatch plugs in the Heater Bay. Some flooding occurred in the basements of the Intermediate Building and the Unit 1 Auxiliary Building. Discharge from the break ceased when the three Circulating Water pumps were shutoff within 34 minutes of the break.

Both circulating and service water pipe failures were determined to have been caused by manufacturing and installation defects.

As a result of these two events, administrative controls were strengthened, improvements in regard to leak tightness of the electrical manholes were performed as explained in the following section, and measures were taken for improved inspection of the FRP piping in each of the systems.

5.4.4.2 The Likelihood of Fiberglass Pipe Rupture at PNPP

A practical method to determine the likelihood of fiberglass pipe ruptures during the plant operation is to use past CEI failures and experiences as carefully as possible to confirm plausibility. In addition, the industry's experience with these type of failures would also help in bracketing the issue. The probabilistic assessment approach can not be used, because a statistically sound database does not exist across the industry.

An industrial survey on FRP piping, a review of the historical service performance of the FRP piping in both the Service Water System and the Circulating Water System was performed for CEI. This undertaking was originally proposed as a means of collecting the data to establish a database and to conduct statistically-based estimate of the future reliability of the FRP piping systems at the Perry Plant. However, after it became evident that the FRP piping systems would not likely last for the remaining life span of the plant, the focus was shifted from numerically determining the future reliability of the piping systems to defining the actions required to enhance the piping so that it would achieve the desired life.

The industrial survey identified 12 CorBan Industries and 7 Owens-Corning Fiberglass Corporation FRP piping systems located in 8 states, with an average age of 16 years. Nine of the 19 systems have been repaired. One of these 9 systems was subsequently completely replaced with an above-ground steel piping system. Based on the survey, those systems that were correctly installed with proper backfill and in firm native soil had only minor problems. The majority of problems have occurred in systems with marginal soil and high water tables. Few of the plants had inspection programs dating from installation and the majority began at least intermittent inspections only after some type of failure had occurred. This is probably because FRP is a very good corrosion resisting material in very benign service and, therefore, should be almost trouble-free. Most of the problems reported were addressed during planned shutdowns. The comparison of extent of cracking and other defects in earlier outages with the RFO-4 and RFO-5 inspection results at PNPP indicates that at least some degradation is occurring.

It is generally believed that the fracture in the circulating water pipes will only have helical orientation in the same angle as the glass windings. These fractures/failures would be most likely to occur in unrepaired areas of past observed damage or cracking. It is further believed that the initial fracture mechanics would probably take years to slowly develop to a helical crack that would extend through the pipe wall.

The fracture type in the service water piping generally will have circumferential orientation, which would be most likely to occur in unrepaired areas of the past observed damage or cracking. The fracture propagation rate would increase exponentially. The initial fracture mechanics would probably take years to slowly develop to a circular crack that would extend through the pipe wall.

While the potential for the FRP pipe cracks cannot be absolutely excluded, their likelihood is minimized by the aggressive inspection and repair program carried out at each refueling outage for the piping in the P41 and N71 systems. When all relevant defects are repaired, it is the opinion of CEI's contracted fiberglass specialist, Fiberglass Structural Engineering, Inc., that the rapid catastrophic failure of either FRP piping system is unlikely to occur between refueling outages. Ultimately, hardware changes will be made to eliminate this potential in the Service Water system. All of the Service Water FRP pipe will be replaced with steel or refurbished with a cured-in-place-pipe (CIPP) in RFO-6. Inspection, trending and monitoring of the Circulating Water system will continue throughout plant life to detect any structural defects prior to a failure condition.

In addition, it should be noted that the impact of any pipe break leading to a large over ground flood should be bounded by the PMP considerations discussed in Section 5.4.3.

5.4.5 Flooding Caused by Miscellaneous Events

The only other event during which water intrusion has occurred at the plant was due to melting of snow piled on one side of the emergency service water pump house structure. However, since this instance the snow cleaning practice at the plant site has been changed, and currently no more snow piling close to the plant building structures takes place. Therefore, it is concluded that the flooding hazard due to sources other than those discussed in the previous sections is insignificant.

5.4.6 Conclusion and Results

In this analysis, an evaluation of the impact on the Perry Nuclear Power Plant of external flooding was performed in response to Generic Letter 88-20, Supplement 4. The following are summaries of the analysis results.

- The "new" PMP criteria will not cause any significant roof overloading of the safety-related structures at the Perry Plant. The 84.5 psf roof design live load for these structures, which corresponds to about 16.4 inches of water, is sufficient to handle the PMP rate of 17.1 in/hour less the capacity of the scupper overflows (5 to 9 in/hr). Additionally, the plant structure roofs are designed for an extra 25 psf of loads for a total design load of 109.5 psf.

- The "new" PMP criteria will not lead to a serious surface flooding and thus will not render the critical equipment/systems inside the safety-related structures from performing their intended operation. The runoff depth has been estimated not to exceed a depth of more than 2.9 inches over the plant entry elevation of 620'-6". The time duration to fail critical equipment inside the safety-related structures, such as the Auxiliary Building and Control Complex, is between 5 to 7 hours. This a very conservative estimate since this is well in excess of the 1 hour period of most intense precipitation. The PMP rate during the 2nd, 3rd and the subsequent hours drops down significantly, with a subsequent lowering of the site water level below the plant entry elevation.

  In addition, the exterior doors have 6 inch curbs and are controlled by a severe weather conditions instruction, where all doors are required to be sandbagged during an intense precipitation condition. This would further reduce the likelihood of a serious flooding of the safety-related structures during a PMP situation.

- The water leakage through conduits into the safety-related structures should not be a safety concern at Perry. Automatic sump pumps are provided to prevent flooding of manholes. In addition, conduits were sealed at the inboard end using Bisco SF60 rubberized sealant. Similarly, water intrusion through the annular space between yard piping and embedded pipe sleeves in plant building walls situated below grade has been protected against by the use of link seals and Bisco rubberized sealant.

- Flooding hazard due to rising groundwater level at the Perry plant site is insignificant to risk. The gravity discharge system is designed to provide a redundant discharge system which incorporates a gravity outfall, having no active components and enough discharge capacity to prevent flooding caused by rising groundwater of main plant buildings.

- Even though the potential for FRP pipe cracks exist, the likelihood of a failure between refueling outages is considered to be low given that the pipes are inspected and defects repaired at each outage, and will be until such time that the pipes are replaced or otherwise enhanced.

5.5 Transportation and Nearby Facility Accidents

5.5.1 Aircraft Impact

Regulatory Guide 1.70 (Rev 3, Section 3.5.1.6, Aircraft Hazards),(5-11) provides the following guidelines and criteria for aircraft hazard analyses:

"An aircraft hazard analysis should be provided for each of the following:

P_{FA} - Federal airways or airport approaches passing within 2 miles of the nuclear facility.

P_A - All airports located within 5 miles of the site.

P_O - Airports with projected operations greater than 500 d^2 movements per year located within 10 miles of the site and greater than 1,000 d^2 outside 10 miles, where d is the distance in miles from the site.

P_M - Military installations or any airspace usage that might present a hazard to the site. For some uses such as practice bombing ranges, it may be necessary to evaluate uses as far as 20 miles from the site."

The total probability of an aircraft crash therefore is the sum:

$$P_{FA} + P_A + P_O + P_M$$

A calculation was performed to establish the probability of an aircraft crash onto the Perry Nuclear Power Plant. The methods used were those outlined in Standard Review Plan 3.5.1.6, Revision 2.[5-5]

5.5.1.1 Federal Airways/Airport Approaches

The calculation of the probability of an aircraft crashing onto the plant (P_{FA}) for situations where federal airways or aviation corridors pass through the vicinity of the plant (vicinity = 2 mi, from Regulatory Guide 1.70, Rev 3), was made using the guidelines of SRP 3.5.1.6, III.2. For each airway or corridor, this probability is:

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

where: C = in-flight crash rate per mile for aircraft using the airway

W = width of airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in miles

N = number of flights per year along the airway

A = effective area of plant in square miles

Thus, for the set of T corridors meeting the analysis criteria in the vicinity of the plant, the total probability is given by:

$$P_{FA} = \sum_{i=1}^{T} C_i N_i A / W_i$$

5.5.1.1.1 Federal Airways

Five airways are situated such that the plant site either lies within the airway or is located less than two miles from one of the airway's outer borders (Figure 5-1). The five cases corresponding to these air traffic routes were investigated. They are V10, V10-188, V522, J190-584 and J29-82 (Ref. 11). V10, V10-188 and V522 are low altitude airways used by general and commercial aviation. J190-584 and J29-82 are high altitude airways designed for use by commercial aviation. All of the airways except V10 are oriented most closely to an east-west direction with respect to the plant axis. Airway V10 is oriented more closely to north-south with respect to the plant axis.

5.5.1.1.2 In-Flight Crash Rates

In-flight crash rates (C_i) were needed for commercial and general aviation aircraft. Summaries of aircraft accident rates are published yearly in a press release from the National Transportation Safety Board (NTSB); the most recent summary available is SB 95-03, which includes data from the years 1982 - 1994.[5-12] More detailed breakdowns of this data are provided in the NTSB's "Annual Review of Aircraft Accident Data," the most recent available version of which is NTSB/ARG-94/02, covering data for calendar years 1983-1992.[5-13] Note that both documents use the same data set for years covered by both surveys, i.e., 1983-1992.

Commercial Aviation Accident Rate (C_{com})

It was assumed for this analysis that fatal accidents correlate to crashes that could impact the plant site, resulting in a value of C_{com} of 1.97×10^{-9} crash/na-mi.

General Aviation Accident Rate (C_{gen})

Again it was assumed that fatal accidents correlate to crashes that could impact the plant site. For general aviation, a rate of 1.76×10^{-5} crash/hr is obtained, and assuming a flight speed of 100 nautical miles per hour (it is conservative to assume a slow flight speed to determine C_{gen}), C_{gen} can be determined by multiplying the accident rate (crashes/hr) by the percent of in-flight accidents and dividing by the air speed as follows:

$$C_{gen} = 1.76 \times 10^{-5} \text{ cr/hr/100 na-mi/hr}$$
$$= 1.76 \times 10^{-7} \text{ cr/na-mi}$$

Fraction of Aircraft Crashes That Occur During Flight Operations

For airway operations, only in-flight crashes need be considered, i.e., takeoff, taxiing, landing can be excluded. Table 21, page 22 of Reference 5-13 provides the percentage based on phase of operation. The cruise, climb, and descent phases result in about 23 percent of accidents. Both C_{com} and C_{gen} need to be adjusted by this ratio to properly reflect the nature of crashes that are postulated for the Perry site. Thus, the rates effective for these calculations will be:

$$C_{com} = 1.97 \times 10^{-9} \text{ cr/na-mi} \times 0.23$$
$$= 4.53 \times 10^{-10} \text{ cr/na-mi}$$

and

$$C_{gen} = 1.76 \times 10^{-7} \text{ cr/na-mi} \times 0.23$$
$$= 4.05 \times 10^{-8} \text{ cr/na-mi}$$

5.5.1.1.3 Effective Plant Areas

The following areas were combined in the calculation of effective plant areas per aircraft type.

- Shadow area (A_{Shadow}) is the area of the vital building elevations projected on the horizontal.
- Skid area (A_{Skid}) is the area in the direction of approach to the plant through which an aircraft could skid into vital buildings
- Roof top area (A_{Roof}) is the simple roof top area of the vital buildings minus the area shadowed by other vital buildings.

Vital buildings and structures are those housing equipment or systems vital to safe shutdown or susceptible to allowing radiological release in excess of 10 CFR 100 limits. Effective plant areas for both general aviation crashes and commercial aircraft crashes were developed for this analysis using worst case assumptions for each.

A_{gen} and A_{com}, the effective crash target areas for general aviation and commercial aircraft, respectively, were developed using an assumed crash angle of 10° with the ground. For each of the east-west airways, an approach direction of 90° to the plant axis was used because it presents the largest area seen by aircraft traveling in this general direction. An approach direction parallel to the plant axis was used for the one north-south oriented airway (V10); this will present the largest representative target area for travel in this direction.

5.5.1.1.4 Shadow Effect of Cooling Towers

For all westbound flight paths in east-west airways, two large concrete cooling towers (approximately 500 feet in height and 300 feet in diameter) provide a significant barrier to impact from crashing or skidding planes on vital structures. Conservatively, only these barriers are considered; no credit is taken for shielding from other structures as would be present on any other approach angle. Each cooling tower (assuming a crash angle of 10°) provides a shadow of length sufficient to protect all vital structures lying directly to the west of the towers. There is, however, a gap in the barrier (the area between the two cooling towers) which leaves some vulnerability to incoming aircraft. The protective effect of these towers, then, is incorporated by modifying the crash rate on East-West airways by a factor P_{gap} which was computed to be equal to 0.667.

5.5.1.1.5 Numbers of Flights

The number of flights assumed is given below:

Airway	Flights per year Commercial[1]	Flights per year General Aviation	Flights per year Total
V10-188	7,300	10,950[2]	18,250
V10	1,460	2,190[2]	3,650
V522	13,870	20,805[2]	34,675
J190-584	19,345	0[3]	19,345
J29-82	56,210	0[3]	56,210

Notes:

1) From FAA estimates
2) General Aviation flights are assumed to represent 60% of total flights
3) All flights assumed to be commercial for high altitude airways.

5.5.1.1.6 Results

The total frequency of aircraft crash into the site from the airways is 8.8×10^{-7}. The main contributions are from airway V10-188 at 3.0×10^{-7}, V522 at 4.1×10^{-7}, and V10 at 1.1×10^{-7}.

5.5.1.2 Airports/Heliports

Three airports and one airstrip are located close to the plant site (Figure 5-1):

Lost Nation Airport
Concord Air Park
Casement Airport
Woodworth (sod airstrip)

There is also a heliport at the site operated by CEI .

The first three airports can be screened on the basis of Criterion 3 of Regulatory Guide 1.70, Revision 3.[5-11]

Woodworth Airstrip

Woodworth is a private sod airstrip located at the Woodworth Farm approximately 4.5 miles east-southeast of the site. Operational airplanes are no longer based there. The approach pattern is 3.5 miles east of the plant center. There is one 2,300 ft runway oriented 90° to 270°. The annual operations are estimated at less than one flight per week; conservatively, 52 flights per year will be assumed.

The annual aircraft crash frequency contribution from the Woodworth Airstrip can be developed as follows.

$$P_A = \sum_i \sum_j C_i N_{ij} A_j$$

where: C_j = probability per square mile of a crash per aircraft movement for the j^{th} aircraft

N_{ij} = number (per year) of movements by the j^{th} aircraft along the i^{th} flight path

A_j = effective plant area (in square miles) for the j^{th} aircraft

The distance from the plant to end of runway is 4 mi + 1,490 ft. From the table on page 3.5.1.6-4 of SRP 3.5.1.6, C = 1.2×10^{-8} crashes/na-mi^2-fl. Nautical miles are assumed.

N = 52 fl/yr

A_{gen} = effective area for general aviation aircraft or 3.5×10^{-3}mi^2

P_{AIV} = C x N x A

= 4.05×10^{-8}cr/na-mi^2-fl x 52 fl/yr x 3.5×10^{-3}mi^2 / 1.32mi/na-mi^2

= 5.58×10^{-9}cr/yr

Perry Heliport

CEI has a private heliport located on the Perry site. This heliport is located approximately 2,000 feet (0.38 mi) southwest of the reactor building and 170 feet southeast of the Emergency Operations Facility (EOF). An analysis was previously developed to calculate the predicted frequency with which a helicopter taking off from or terminating at this heliport might impact vital structures at the Perry site. This analysis was developed in support of revisions made to the USAR that included consideration of the heliport and its impact on plant safety. The use of the heliport has not changed appreciably since the time of that analysis. This prior analysis demonstrated that the predicted annual

frequency of helicopter crashes which impact vital structures is 4.0×10^{-8}. Improvements in helicopter safety over the past ten to twenty years render this figure conservative; however, it will be used as the figure of merit for this analysis.

Combined Airport/Heliport Cases

P_A has contributions only from the Woodworth airstrip and the heliport, totaling 4.56×10^{-8}cr/yr.

5.5.1.3 Military/Other Airspace Usage

There are no military airports or usage in the site vicinity. There is only one holding pattern near the plant site. It is associated with airway J190-584 and is located approximately 17 miles east-northeast of the plant over the shore of Lake Erie. No edge of the holding pattern approaches closer than 10 miles to the site.

Because the holding pattern lies outside the criteria specified by the SRP, it is not considered further.

5.5.1.4 Summary

The total frequency of an aircraft crash from the cases described above is:

$$P = 8.84 \times 10^{-7} + 4.56 \times 10^{-8}$$

$$= 9.3 \times 10^{-7} \text{cr/yr}$$

This frequency does not comply with the acceptance criteria 10^{-7}/yr on page 3.5.1.6-2 of Ref. 5-6. However, Section 2.2.3 of Ref. 5-5 states that "..., the expected rate of occurrence of potential exposures in excess of 10 CFR Part 100 guidelines of approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower." The situation represented in this analysis can easily be shown to meet this criterion.

There are several factors which would lower the calculated probability substantially if accounted for, but were not included in the above calculation because sufficient quantitative information was not available. These factors include:

- The probability referred to in the SRP is for the occurrences of exposures in excess of 10 CFR 100 guidelines. A significant portion of the probability value determined in the above calculation is due to general aviation. The type of aircraft included in the general aviation category is not likely to be able to damage a vital structure to the extent that would cause exposures in excess of 10 CFR 100. Table 6.4.2 of NUREG/CR-5042[5-15] estimates the probability of penetration of concrete structures by incoming aircraft as a function of plane size and distance from the airport at which the plane originated or will terminate. This table shows that, for concrete thickness of two feet or more (as is the case with the vital structures of concern for the Perry plant), there is a probability approaching zero of structure penetration by planes under 12,500 pounds (which bounds the vast majority of general aviation craft, though the FAA could not provide an exact percentage of general aviation craft bounded by this weight).

Thus, if this were accounted for, the above probability would be substantially lower. Assuming, very conservatively, that only half of the general aviation planes in the three low altitude airways of interest weigh less than 12,500 pounds (effectively eliminating their contribution to C_{gen} and thus halving the overall C_{gen} used in the calculations), the overall frequency of exposure in excess of 10 CFR 100 guidelines would drop below $8x10^{-7}$.

- Turbine buildings 95 feet in height and over 450 feet in length lie to the north and the south of the auxiliary/reactor/control/diesel building complex. These buildings provide barriers that would, in reality, prevent a significant amount of air crashes from impacting vital structures. Because they are not concrete structures, they are conservatively not credited. However, keeping in mind the first qualifying statement above regarding general aviation craft, it would be very likely that such a plane would be fully stopped by such a barrier building.

  At an approach angle of 10°, the shadow of each Turbine Building is approximately 550 feet. This would provide adequate protection from northbound or southbound planes for all vital structures except the Emergency Service Water Pump House. This would reduce the overall P for aircraft accidents as stated above by about 12 percent.

While the lack of specific data on general aviation aircraft size precludes quantification of these two factors, it is clear that even a conservative accounting of their effects on the probability of aircraft accidents resulting in offsite exposures in excess of 10 CFR 100 guidelines will result in a frequency below 10^{-7}. Thus, the frequency of aircraft accidents which could result in offsite exposure level exceeding 10 CFR 100 limits is considered to be below the level of acceptability for conservative analyses, and no additional design or procedural protection to mitigate their consequences are required.

5.5.2 Transportation Accidents

There are two aspects of concern for transportation accidents; explosions that could lead to damage to plant structures, and the release of hazardous chemicals which could lead to the incapacitation of the control room operators. Either of these occurrences could initiate a series of events that could lead to core damage. Hazardous materials and explosions will be dealt with separately. The proximity of transportation corridors to the Perry site requires that three modes of transportation be investigated; lake traffic, road, and rail.

5.5.3 Hazardous Materials

5.5.3.1.1 Methodology for the Assessment of Risk from Hazardous Materials

The analysis was composed of a number of steps. First, any chemicals which could be qualitatively judged as of no concern to the control room operators at Perry were excluded from further evaluation. Then, adjusted allowable weights were calculated for each chemical not screened by any of the above factors, using the methodology of Regulatory Guide 1.78.[5-17] Those transported (or stored) in quantities less than the adjusted allowable weight were eliminated from further consideration. The toxic effects of those not screened in either of these ways must be evaluated. This was done using the DOT Emergency Response Guidebook (ERG).[5-24] Chemicals for which the control room is within the ERG impact zone for toxicity of that material were evaluated probabilistically, as it was assumed that these chemicals are capable of incapacitating the control room operators.

<u>Qualitative Screening Criteria</u>

The criteria used to qualitatively screen the lists of hazardous commodities shipped in the vicinity of PNPP to exclude those that pose no threat to the Control Room are described below.

- Section C.2 of Regulatory Guide 1.78 states that "[I]f hazardous chemicals... are known or projected to be frequently shipped by rail, water, or road routes within a five-mile radius of a nuclear power plant, estimates of these shipments should be considered in the evaluation of control room habitability...Shipments are defined as being frequent if there are... 30 per year for rail traffic..." Thus, all chemicals that are shipped in quantities of less than 30 cars per year will be eliminated from further consideration.

- Certain gases, such as argon and nitrogen, have no specific toxicity effect, but act by excluding O_2 from the lungs. Asphyxiants 3 or more miles from the plant are not considered to be a threat to the Control Room.

- Per Ref. 5-16, "when an explosive or flammable material is accidentally released, it is highly likely to explode or burn before its vapor reaches the control room. For this reason, only those substances whose toxicity limits are less than their lower limits of explosion and flammability were considered for control room habitability here." The effects of explosions are addressed in Section 5.5.3.2 of this document.

- Solid and liquid hazardous materials are local hazards and will be assumed to be incapable of traveling distances of three miles or more in sufficient quantities to be of concern (unless they can form a vapor after spillage that can be transported to the plant).

- Liquids with vapor pressures less than 10 torr @ 100°F (38°C) and normal atmospheric pressure are excluded per Regulatory Guide 1.78.

Adjusted Allowable Weights

Regulatory Guide 1.78 introduces the concept of maximum adjusted allowable weight. It is noted that this weight is for that quantity of the chemical which is released immediately in vapor form upon release. This would be the total contents of a gas tank or that portion of a liquefied gas that flashes upon release. Commodities were immediately excluded if the adjusted allowable weight exceeded the capacity of the container in which the chemical was stored or transported. Further consideration (evaporation or flashing) was not given here to those commodities which are not transported as gases to determine whether or not they may be excluded; they were assumed to convert directly and immediately to vapor, i.e., the most conservative assumption of 100% flashing is used.

To perform the required calculations, the distance between the source of the chemical release and the plant must be known. The adjusted allowable weight per Appendix A of Ref. 5-21 was determined for chemicals not screened qualitatively using parameters developed in the following manner:

Reference distance ≡ X miles
Control Room Type ≡ C
Unadjusted allowable weight ≡ 33,000 lb = 16.5 tons
Adjusted allowable weight ≡ 16.5 tons (toxicity limit [mg/m^3] / 50) x 0.44

Adjusted allowable weights, as defined in Regulatory Guide 1.78, were calculated for each chemical not screened by any of the above factors. Those with a shipment or container size less than the adjusted allowable weight were eliminated from further consideration.

Conservatism

Evaporation (boiloff) and flashing models for spilled liquid hazardous chemicals that can be used to calculate concentration at the Control Room based on the evaporative release rate and atmospheric dispersion were not included at this time. There are insufficient data readily available for some of the chemicals in this analysis to properly use the models provided in, for example, NUREG-0570.[5-16] Instead, the entire spilled quantity was assumed to convert directly and immediately into vapor form and was assumed to be transported to the control room intake with "instantaneous puff release" characteristics. This introduced a large factor of conservatism in the analysis.

5.5.3.1.2 Railroads

There are two railroads that carry hazardous chemicals in the vicinity of PNPP; Conrail, and the Norfolk Southern Railway Company (NSRC), see Figure 5-2. Because of the great distance (3 miles or more) between the rail lines and the plant site, only those hazardous materials capable of being transported by air are considered a hazard to control room operators. Both Conrail and Norfolk Southern provided their most recent annual summaries of their hazardous materials shipments for use in this analysis. A simplified screening process was used to eliminate all chemicals that can be specifically identified as non-threatening to the plant and its operators. All other chemicals were included in a calculation of the frequency of a transportation accident involving release of toxic chemicals which could result in offsite exposure levels exceeding 10 CFR 100 limits. In this calculation, unscreened chemicals were not assessed individually; the probabilities derived were based on the aggregate shipment by rail of all commodities whose release as a result of an accident have the

potential to impact control room habitability. This is consistent with the USNRC position described in Reference 5-17.

Method for Estimation of Frequency

The frequency of exceeding 10 CFR 100 limits due to the transport of hazardous materials via rail in the vicinity of the plant is calculated using the methodology of NUREG/CR-2650.[5-18]

$$F_{100} = P_{100/OI} \times F_{OI}$$

$$F_{100} = P_{100/OI} \times F_s \times P_{ak} \times L \times P_R \times P_I$$

where:

F_{100} = probability of exceeding 10 CFR 100 limits per year

$P_{100/OI}$ = probability of exceeding 10 CFR 100 limits given incapacitation of the operators

F_{OI} = probability of incapacitation of the operators as result of hazardous materials accident

F_s = frequency of shipments per year

P_{ak} = probability of accident per km traveled

L = km rail within 5 miles of the plant

P_R = probability of a large release to occur from a given accident

P_I = probability of operator incapacitation given that a large release occurs along a specified stretch of track ($P_{OI/release}$)

Input Parameters

Probability of exceeding 10 CFR 100 limits given incapacitation of the operators:

$$P_{100/OI} = 0.1 \quad \text{(NUREG/CR-2650, page 6)}$$

Frequency of hazardous shipments per year:

$$F_s = 5{,}945$$

Probability of accident resulting in release per km traveled:

$$P_{ak} = 4.18\text{x}10^{-8}$$

This number was calculated using actual U.S. railroad hazardous materials accident and traffic data for the years 1985-1989 (the latest years for which complete data was available).

Rail length exposed per shipment (L):

The approximate rail length exposed per shipment in each 22.5° sector within 5 miles of PNPP is obtained by scaling track lengths from USAR Fig. 2.2-2. The Conrail routing was used as being representative of NSRC routing. Although NSRC tracks are slightly longer in the southern sector, the overall Conrail length within 5 miles is longer. This is acceptable given the conservatism in determining F_s.

Probability of a large release to occur from a given accident:

$$P_R = 0.45$$

The P_{ak} provided above gives a probability of a rail accident which results in any release of toxic chemicals. P_R here, then, is the fraction of all accidents which result in a release that result in large releases, and is derived from Department of Transportation statistics.

Probability of operator incapacitation given that a large release occurs along a specified rail stretch:

$$P_I = P_{OI/release}$$

This probability is the percent of time which meteorological conditions are such that the release of hazardous materials would impact the Control Room. It is assumed that stability classes A, B, C, D, and E sufficiently disperse the cloud that the toxicity level is not reached at the Control Room (Regulatory Guide 1.78 Appendix A). Thus, P_I is the probability that the meteorological condition of stability class F or G is present with the wind blowing in the direction of the Control Room air intake. A distribution of wind direction (hours) versus stability was generated utilizing the PNPP 10-m wind and delta-T (60m-10m) joint frequency distribution reports for the last 5 years.

The probability factors are constant except for L and P_I. Thus, the total probability is given as:

$$P_{100} = P_{100/OI} \times F_s \times P_{ak} \times P_R \times \Sigma (L \times P_I)$$

where the summation is over the sectors which have rail tracks within 5 miles. This computation is shown below:

Summation of Probability Factors: Rail

Sector	L (miles)	L (km)	P_I	P_{100}/sector
SW	0.5	0.80	0.00392	3.51e-08
SSW	1.8	2.90	0.00999	3.22e-07
S	1.3	2.09	0.01513	3.52e-07
SSE	1.3	2.09	0.01886	4.39e-07
SE	1.5	2.41	0.02042	5.48e-07
ESE	1.0	1.61	0.02532	4.53e-07
Total	7.4	11.91		2.15e-06

5.5.3.1.3 Road Transportation

Unlike the case of railroads, specific information on road shipment sizes and frequencies for specific hazardous chemicals is neither required by state or federal agencies nor generally kept by facilities. Information regarding shipment frequency and sizes can only be estimated from information on actual shipments to and from the local facilities hazardous materials.

Three classes of roadway are of interest in this analysis. The first consists of roadways that handle local traffic that lie within five miles of the plant site. The two major roads in the vicinity of Perry that meet this categorization are U.S. Route 20 (1 mile SSE of the plant), and State Route 84 (3.5 mi SSE) (Figure 5-2). There are several facilities within a five mile radius of the Perry site which have some road transport of hazardous materials affiliated with their operations; these will be considered the source of local roadway hazardous materials traffic. Five facilities handle significant quantities of hazardous materials and are within the five-mile radial area of the plant site. Four (Zeneca Chemicals, PET Processors, the Lake County Water East facility, and the Madison Village Sewer Plant) lie more than three miles from the plant site; only the Neff-Perkins facility lies within one mile of the plant site. The list of chemicals assumed transported is based on data received from the Lake County Emergency Planning Committee. Chemicals that are not required to be reported to county officials for emergency planning purposes are assumed to be exempt from further evaluation.

The hazardous materials assumed to be transported on local roads within five miles of the plant site were screened to exclude those commodities that do not pose a threat to the Control Room. A probabilistic evaluation was performed considering the probability of a hazardous materials truck accident that could impact the Control Room and wind direction and stability class joint distribution frequencies to estimate the probability that the offsite exposure levels would exceed 10 CFR 100 limits.

The second class evaluated consists of one stretch of road - the small portion of Interstate 90 that lies within a five-mile radius of the plant. Unlike the local roadways, it is impossible to speculate on the nature of shipments over this section of roadway due to the widely varying composition of the shipments and the lack of available data. Local emergency response procedures become the element in assessing the level of the risk to the plant from accidents on this small stretch of road.

Finally, those roadways lying more than five miles from the plant site were handled using the guidelines of Regulatory Guide 1.78. Several major limited access highway routes (including the rest of Interstate 90, State Route 44, State Route 2 west of Painesville, and the highway extension into Fairport Harbor) and one major local roadway (State Route 528) are outside this radius. The limited access highway routes are the preferred route for hazardous materials traffic into the industrial facilities in the Painesville and Fairport Harbor areas to and from the east. Also, it is likely that Route 528 will carry some local hazardous materials traffic. However, per the instructions of Regulatory Guide 1.78, the traffic on these roads will be excluded from this calculation. To wit, Section C states that "[c]hemicals stored or situated at distances greater than five miles from the facility need not be considered because, if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the incoming plume to such a degree that there should be sufficient time for control room operators to take appropriate action. In addition, the probability of a plume remaining within a given sector for a long period of time is quite small."

Screening of I-90

Most of I-90 lies outside the five-mile radius of the plant. One small section less than two miles in length does, however, lie within the radius, according to the most recent surveys of the area surrounding the plant (Figure 5-2). Interstate routes cannot easily be treated quantitatively, as federal, state, and local governmental agencies do not track hazardous materials shipment frequencies on roadways. The only realistic means of counting truck traffic would be a long-term traffic survey, which is extremely difficult and time-consuming. For that reason, it was desirable to first attempt to determine qualitatively if there is any reason to believe truck traffic on this stretch of roadway presents a realistic risk to plant operators.

Emergency Response to a Truck Accident Involving Hazardous Materials

In the event of an accident, the Lake County Emergency Planning Committee (LEPC) for Lake County would direct response operations. A representative of the LEPC with extensive emergency response procedure experience provided the following information relevant to an accident situation involving hazardous materials accidents on I-90 south of the plant site.

If the first responders determined that the accident posed a toxic threat to the local citizenry, one of the first formal actions that would be taken would be to initiate pluming projections using a computer

program set known as CAMEO/ALOHA to determine the characteristics of the potential spread of the chemical plume. This program projects plume travel and concentrations, given material properties, spill quantities, and weather conditions. Should these projections indicate that evacuation or sheltering in the local area is necessary, an existing county Emergency Operations Center (EOC) facility will be activated to centralize the communications for emergency response.

From this point, the response is very dynamic and situation-driven. Updating the CAMEO/ALOHA runs with shifting weather conditions, the EOC can track the plume and its projected concentrations (or "dose curves"), and determine what actions (evacuation, shelter-in-place, etc.) are required. In most situations, field verifications of the computer projections are made; any discrepancies between the projected and actual concentration/location information are entered into the program, which is constantly updated and rerun. Since the first projections are usually available within 30-40 minutes, this gives the EOC a time window of sufficient size to notify Perry before toxic concentrations will be seen at the plant.

The fact that a highway accident centralizes communications at an EOC significantly enhances communication with the plant control room. To handle nuclear emergency situations, direct telephone lines between the LEPC and the Perry control room are maintained. While these are intended to serve as a resource in the event of a nuclear accident, these direct lines also give the EOC an immediate, direct line that can be used to notify Perry operators of any impending chemical hazard.

As a practical matter, the existence of populated areas between I-90 and the plant narrows the time window in which the EOC must act to one smaller than that required for timely plant notification. Since an evacuate/shelter-in-place/no action decision will have to be made based on populations much closer to the roadway than the plant, a decision on the direction and toxicity of the cloud will be made in time to use the direct communication lines with the plant to notify them to take action. Emergency responders will act very conservatively based on their instincts and experience rather than wait for the results of computer runs and projections, and, in the event of an accident involving an extremely toxic chemical, they will begin appropriate action in all areas surrounding the accident immediately, using the computer projections to confirm the need for such actions. However, in the event of an accident of the level of severity that would be a hazard to the plant, immediate and general action will be taken to ensure public safety. These actions would include notifying the Perry control room, since the Perry plant, because of its concentration of workers and its operational nature and public safety significance, is one of the first priorities for notification in case of a significant accident, according to the LEPC.

Physical Considerations

There are also certain physical realities that would likely result in such an accident being basically a local concern (within a radius of 1-2 miles), and thus have no significant impact on the plant. First, unlike rail or fixed facility accidents, the quantities involved are quite limited (under 9,500 gallons). Thus, for all but extremely toxic chemicals, dispersion would, in most cases, reduce concentrations effectively. More importantly, there are very significant geologic features of the area that would serve to almost completely contain a heavier-than-air plume. (This would include nearly all severely toxic chemicals that would be transported by truck; lighter-than-air chemicals would likely rise well above the control room intake by the time they had traveled five miles.) Specifically, the Grand River runs in a deep ravine basically parallel to I-90 along that stretch of road. There are dramatic elevation changes and small Grand River tributaries associated with that ravine both north and south of I-90. A cloud of

toxic, heavier-than-air vapor would collect in the tributaries and Grand River basin, and, particularly for the limited quantities associated with truck transport, remain there. This makes it extremely unlikely that any of the vapor would reach the plant.

Disposition of Truck Accidents on I-90

Based on these factors, the event of a release from a truck accident on I-90 incapacitating control room operators at Perry appears highly unrealistic. The precautions in place and the physical features of the accident development and the surrounding terrain all act to strongly counter that possibility.

Screening Criteria

The procedure used to screen chemicals transported on local roadways is essentially the same as that for railroads. The major differences between the cases of transportation by road and rail are:

- The closest roadway, U.S. Route 20, is one mile away at its closest approach point to the plant (as opposed to three miles for rail travel, as discussed in the previous section). This forces more restrictive criteria on the chemical screening in terms of air transport after an accident; specifically, the unadjusted allowable weight is reduced to 2,500 pounds for all chemicals transported by truck except perchloromethyl mercaptan (PMM), which is not transported on Route 20.

- The maximum truck tank size assumed is 9,500 gallons. While no formal guidelines dictating tank size for trucks exists, informal discussions with USDOT personnel and review of past hazardous materials truck incidents reported to DOT and the National Transportation Safety Board demonstrate that this is an acceptable upper limit for tank size. Maximum cargo quantities usually run no greater than 90-95% of this capacity; however, in this analysis, a full 9,500 gallon capacity will be assumed.

Method for Estimation of Frequency

The frequency of exceeding 10 CFR 100 limits due to truck transport of hazardous materials within a five-mile radius of the plant is calculated using the methodology of NUREG/CR 2650[5-16]

$$F_{100} = P_{100/OI} \times F_{OI}$$

$$F_{100} = P_{100/OI} \times F_s \times P_{ak} \times L \times P_R \times P_I$$

where:

F_{100} = frequency of exceeding 10 CFR 100 limits per year

$P_{100/OI}$ = probability of exceeding 10 CFR 100 limits given incapacitation of the operators

F_{OI} = frequency of incapacitation of the operators as result of hazardous materials accident

F_s = frequency of shipments per year

P_{ak} = probability of accident per km traveled

L = length of roadway used for hazardous materials transport lying within 5 miles of the plant

P_R = probability of a large release to occur from a given accident

P_I = probability of operator incapacitation given that a large release occurs along a specified stretch of roadway ($P_{OI/release}$)

Input Parameters

Probability of exceeding 10 CFR 100 limits given incapacitation of the operators:

$P_{100/OI} = 0.1$ (NUREG/CR-2650[5-18], page 6)

Frequency of hazardous shipments per year:

There is assumed to be one ammonia shipment and three sulfur monochloride shipments per week on Route 20. All other chemicals have been screened from consideration. Therefore, on an annual basis, $F_s = 4 \times 52 = 208$.

Probability of accident per km traveled:

$P_{ak} = 1.6\times10^{-6}$ accidents/truck-km (NUREG/CR-2650[5-18], page 9)

Road length exposed per shipment (L):

The approximate road length exposed per shipment in each 22.5° sector within 5 miles of PNPP is obtained by scaling road segment lengths from USAR Fig. 2.2-2.

Probability of a large release to occur from a given accident:

$$P_R = 0.005 \text{ (NUREG/CR-2650,[5-18] page 10)}$$

Probability of operator incapacitation given that a large release occurs along a specified stretch of roadway:

$$P_I = P_{OI/release}$$

This probability is the percent of time which meteorological conditions are such that the release of hazardous materials would impact the Control Room. It is assumed that stability classes A, B, C, D, and E sufficiently disperse the cloud that the toxicity level is not reached at the Control Room (Regulatory Guide 1.78 Appendix A). Thus, P_I is the probability that the meteorological condition of stability class F or G is present with the wind blowing in the direction of the Control Room air intake.

The probability factors are constant except for L and P_I. Thus, the total probability is given as:

$$P_{100} = P_{100/OI} \times F_s \times P_{ak} \times P_R \times \Sigma (L \times P_I)$$

where the summation is over the sectors which have road segments within 5 miles. This computation is shown below:

Summation of Probability Factors: Roadway

Sector	L (miles)	L (km)	P_I	P_{100}/sector
SW	2.5	4.02	0.00392	2.62e-09
SSW	1.3	2.09	0.00999	3.48e-09
S	0.8	1.29	0.01513	3.24e-09
SSE	0.7	1.13	0.01886	3.55e-09
SE	0.7	1.13	0.02042	3.83e-09
ESE	1.5	2.41	0.02532	1.02e-08
E	2.8	4.51	0.01705	1.28e-08
Total	10.3	16.58		3.97e-08

5.5.3.1.4 Waterway Traffic

The Great Lakes Region Freight Traffic Tables for the most recent year available provide a comprehensive review of the substances transported over the waterways near the plant. They demonstrate that no hazardous materials capable of being transported great distances in the air are transported over these routes. Mineral and metal products (primarily iron ore, iron products, and limestone) dominate the freight traffic in the area. For this reason, waterway traffic is judged to be incapable of contributing to the toxic chemical hazard risk for the plant and will not be analyzed further.

5.5.3.1.5 Summary

The aggregate probability of a rail, road, or barge accident involving the release of toxic chemicals which could result in offsite exposure level exceeding 10 CFR 100 limits as derived in this calculation is 2.2×10^{-6} per year. This exceeds the SRP 2.2.3 criterion of 1×10^{-6} per year. However, this figure must be viewed in light of the extraordinary conservatism inherent in the simplified analytical method used, particularly for the rail accident analysis, which represents 99 percent of the value stated above. As examples of the conservatism, the actual carload count may be exaggerated by a factor of two, and the probability of operator incapacitation by an order of magnitude. These primary factors, plus other less significant considerations would be expected to bring the actual probability to or below 10^{-7}, in line with SRP criteria. Thus, transportation accidents involving the release of toxic chemicals which could result in offsite exposure level exceeding 10 CFR 100 limits are not considered credible and do not require additional design or procedural protection to mitigate their consequences.

5.5.3.2 Explosions

The potential impact of accidents involving explosives was analyzed by establishing whether the closest point to the plant at which an accident could occur is far enough that the overpressure at the plant created by such an explosion would be less than the design basis. In all cases, (i.e., road, rail, and lake traffic) this was shown to be the case. No explosives, flammable gases, or other combustible materials have a credible potential to explode and cause design overpressures at the plant to be exceeded, taking into account the quantities and specific materials transported.

5.5.4 Fixed Facility Accidents

5.5.4.1 Industrial Facilities

As is the case for transportation accidents, the potential for both the release of toxic chemicals and explosions is of concern. The installations of concern are indicated on Figure 5-2.

5.5.4.1.1 Toxic Chemicals

The methodology used was the same as that for the case of transportation, discussed in Section 5.5.2. Design input and major assumptions used in the analysis are provided below.

- Data received from the LEPC is the basis for identifying offsite hazardous chemicals. Individual companies were contacted only if additional information was required.

- Only those hazardous materials capable of being transported by air, e.g., gases or vapors, will be considered a hazard to control room operators. Solid and liquid hazardous materials are assumed to be local hazards incapable of traveling distances of one mile or more in sufficient quantities to be of concern, unless they can form a vapor after spillage that can be transported to the plant.

- Chemical and toxic properties of chemicals are obtained from standard references, typically References 5-19, 5-20, or 5-21, and are noted where used.

- Ambient temperature is assumed to be 104°F

- If a toxicity limit is provided in Regulatory Guide 1.78[5-23] for the chemical under analysis, that value is used. However, only a limited number of chemicals are listed. Where other chemicals are assessed, Ref. 5-19 and/or Ref 5-21 are used to determine the incapacitating level. These levels and their effects are described in this reference, and are reviewed to ensure conformity with the Regulatory Guide 1.78 definition and intent of "toxicity limit."

- Where no discussion is provided in Ref. 5-17 or Ref. 5-19 describing an incapacitating level, the OSHA Standard Air time-weighted-average concentration (TWA) value is typically used. This value is extremely conservative, since it is a concentration of a substance to which most workers can be exposed for an eight hours per day, five days per work week without adverse effect. In some instances, ceiling or peak concentrations or Short Term Exposure Limit (STEL) values may alternately be used.

- It is assumed that container, i.e., rail car, tank, failures are independent, and that multiple container failures in one incident are not credible, i.e., are below the threshold probability of concern for this analysis.

Adjusted Allowable Weights

To perform the required calculations, the distance between the source of the chemical release and the plant must be known. In this analysis, a reference distance of 3 miles was used. Nearly all of the major chemical storage facilities in the area are more than three miles away from the control room intake, so assuming a 3 mile reference distance is conservative. Storage facilities located more than three miles will be bounded by these weights, as the adjusted allowable weights increase with distance between the source and site. If a facility is less than three miles from the plant, a separate calculation was done where necessary to determine the appropriate adjusted weight for that case.

Two chemicals were screened on the basis of adjusted allowable weights; carbon tetrachloride, and methyl ethyl ketone.

Isolation and Protective Action Distances

Following the above screening, only chlorine, trichloromethane sulfenyl chloride or PMM, sulfur monochloride, and ammonia were left to be analyzed with the next screening approach. This evaluation was performed using the guidelines of the DOT Emergency Response Guidebook (ERG).[5-24] This guidebook was developed by the U.S. Department of Transportation for use by firefighters, police, and other emergency services personnel who may be the first to arrive at the scene of a serious incident involving a hazardous material. It is primarily a guide to aid first responders in quickly identifying the specific or generic classification of the material(s) involved in the incident, and protecting themselves and the general public during this initial response phase of the incident.

Initial isolation and protective action distances for this guidebook were determined for small and large spills occurring day or night. Analysis used state-of-the-art source term and vapor cloud dispersion modeling, probabilistic application of actual atmospheric data, and the latest toxicological exposure guidelines available for each material. Dispersion models calculated downwind vapor concentrations based on actual, 24-hour, groundlevel and upper-air meteorological data from 61 cities (including one each in Alaska and Hawaii) over a 5 year period. Data showed nighttime atmospheric conditions transported vapor plumes much greater distances than daytime conditions, therefore, daytime and nighttime protective action guidance is provided to more accurately describe risk.

Source term modeling considered three factors; tank sizes, spill rates from damage to each tank, and release of vapors by evaporation from a liquid pool, direct release of gaseous vapors from a package into the atmosphere, or a combination of both.

Toxicological short-term exposure guidelines for the materials were applied to vapor concentrations to determine how far downwind the public is in danger. An independent panel of toxicological experts from industry and academia recommended that toxicological exposure guidelines be chosen from emergency response guidelines, occupational health guidelines, and lethal concentrations determined from animal studies. Specific means of application of these health criteria and adjustments based on time-of-exposure were made when recommended by the panel of experts.

The Protective Action Distance (PAD) assigned for each chemical assumes random changes in wind direction confines the vapor plume to an area with 30 degrees on either side of the predominant wind direction, resulting in a crosswind protective action distance equal to the downwind protective action

distance. Within the protective action zone, vapor concentrations present may be capable of incapacitating nearly all unprotected persons. The Initial Isolation Zone is determined as an area, including upwind from the incident, within which a high probability of localized wind reversal may expose nearly all persons without appropriate protection to life threatening concentrations of the material. (The isolation zone represents a much smaller area than that of the PAD; thus, in this analysis the PAD was used exclusively.)

Zeneca Chemicals, the only facility at which very large stores of chemicals are present within a five-mile radius of the plant, uses the ERG to plan and conduct much of its accident response activity. Thus, this was deemed a suitable reference for Protective Action Distances for the chemicals not screened out earlier. To maintain an appropriate level of conservatism, the largest published PAD was used to determine whether or not the chemical in question can incapacitate plant operators. This will maximize the probability that the Perry Control Room falls within the protective action distance. A probabilistic evaluation must be conducted for any chemical for which the control room falls within these boundaries, as it was assumed that the chemical is capable of incapacitating control room operators. Note that, due to the conservatism inherent in the DOT analyses used to derive these distances, these distances themselves are quite conservative. They are intended to provide a large margin of safety to local facilities and citizens, and are thus adequate for this analysis.

The Perry control room is outside the affected zone for three of the four chemicals. However, detailed analysis must be performed for chlorine storage, since the Perry control room lies within its impact radius.

Chlorine Storage Locations and Quantities

The table below lists all of the stores of chlorine that are of concern in this analysis. The adjusted allowable weight of chlorine is 13,068 lb. Any storage tank containing less than this amount of chlorine can be excluded from further analysis, per Regulatory Guide 1.78; any exceeding this must be included in the probabilistic evaluation.

Location	Maximum Tank Quantity *(Note 1)*	Tank Type	Screened on Adj. Allowable Weight?
Madison Village Sewer Plant	150 lb.	Small tanks	Yes
PET Processors	150 lb.	Small tanks	Yes
Lake County Water-East	2,000 lb.	Tanks	Yes (Note 2)
Zeneca, Inc.	999,999 lb.	Rail car	No
Zeneca, Inc	9,999 lb.	Tank (inside)	Yes
Zeneca, Inc.	999 lb.	Tank (outside)	Yes

Notes:

1) From LEPC/SARA reports. Note that sizes are often reported as ranges, e.g., 10-99 lbs, 100-999 lbs, etc. For those cases, maximum inventory in range is shown and used as comparison point for adjusted allowable weights.

2) The Lake County Water - East facility, located four miles from the plant site, was contacted to determine the precise chlorine storage capabilities they have on their site. A representative of the facility stated that 14,000 pounds of chlorine are generally stored on site in seven 2,000 pound tanks. Since the facility is more than 3 miles away from the Perry plant, the adjusted allowable weight of 13,068 lb. can be used to disposition the chemical. Since the maximum storage in any container of 2,000 pounds is well under the adjusted allowable weight of approximately 13,000 pounds and several coincident failures of chlorine tanks are considered not credible, this chemical will be excluded from further consideration.

Estimation of Frequency

Methodology

The frequency of exceeding 10 CFR 100 limits due to failure of one of the Zeneca rail car chlorine storage tanks is derived from the methodology of NUREG/CR-2650.(5-18)

$$F_{100} = P_{100/OI} \times F_{OI}$$

$$P_{100} = P_{100/OI} \times C \times P_{ak} \times P_R \times P_I$$

where:

F_{100} = frequency of exceeding 10 CFR 100 limits per year

$P_{100/OI}$ = probability of exceeding 10 CFR 100 limits given incapacitation of the operators

F_{OI} = frequency of incapacitation of the operators as result of hazardous materials accident

C = number of cars in storage within five miles of the plant

P_{rc} = probability of hazardous materials release per car

P_R = probability that a given car failure is catastrophic and results in a large release

P_I = probability of operator incapacitation given that a catastrophic rail car failure with large release occurs

Input Parameters

Probability of exceeding 10 CFR 100 limits given incapacitation of the operators:

$$P_{100/OI} = 0.1 \text{ (NUREG/CR-2650,[5-18] page 6)}$$

Number of cars in storage within five miles of the plant:

Storage of chlorine at Zeneca cannot be screened; this represents [illegible] ars per year.

Probability of hazardous materials release per car:

Hazardous materials release data for rail cars for non-accident incidents in the United States from 1986-1991 give a frequency of hazardous materials release per car of 7.09×10^{-4}.

Probability that a given car failure is catastrophic and results in a large release:

> U.S. Department of Transportation Hazardous Materials Information System records for 1993 and 1994 were reviewed to determine the fraction of all releases that could be described as "large." Since the lower of the two exclusion weights for the chemicals of interest here is about 13,000 pounds (chlorine), and the liquid density for chlorine is about 13 pounds per gallon, a "large release," for the purposes of this analysis, will be one that exceeds 13,000 pounds or 1,000 gallons. Of the 2,177 releases reported in the DOT data base, 23 exceeded this criterion. Therefore, the probability of a large release given a non-accident rail car release will be approximated as 23 / 2,177, or 1.05×10^{-2}.

Probability of operator incapacitation given that a catastrophic rail car failure with large release occurs:

$$P_I = P_{OI/release}$$

> This probability is the percent of time which meteorological conditions are such that the release of hazardous materials would impact the Control Room. It was assumed that stability classes A, B, C, D, and E sufficiently disperse the cloud that the toxicity level is not reached at the Control Room (Regulatory Guide 1.78 Appendix A). Thus, P_I is the probability that the meteorological condition of stability class F or G is present with the wind blowing in the direction of the Control Room air intake. The factor that applies to the spatial relationship between the Zeneca and Perry facilities is 0.02512.

Therefore, the total probability of a rail car leak or rupture that results in a large chemical release capable of incapacitating the control room operators and resulting in an offsite exposure exceeding 10 CFR 100 limits is:

$$P_{100} = P_{100/OI} \times C \times P_{ak} \times P_R \times P_I$$
$$= 3.74\times10^{-8}$$

Summary

The aggregate probability of a railcar or storage tank failure involving the release of toxic chemicals which could result in offsite exposure level exceeding 10 CFR 100 limits derived in this calculation is 3.74×10^{-8} per year. This is well below the SRP 2.2.3 criterion of 1×10^{-6} per year. Thus, offsite chemical storage container failures involving the release of toxic chemicals which could result in offsite exposure level exceeding 10 CFR 100 limits are not considered credible and do not require additional design or procedural protection to mitigate their consequences.

5.5.4.1.2 Explosives

Only two chemicals that are potentially explosive were identified as being stored in nearby facilities. These chemicals, butadiene and carbon bisulphide, are stored at the Zeneca facility, which is 3.5 miles from PNPP. As with the case of the transportation accidents involving explosives, it was demonstrated in two ways that the impact of explosions from these stored chemicals would not have significant consequences at PNPP. First, it was demonstrated that the plant is at a sufficient distance form the stored chemicals that the overpressure is less than 1 psi. Second, the overpressure expected at the plant was estimated, and shown to be considerably less than the design basis overpressure. Thus, chemical explosions are demonstrated to be of no concern.

5.5.4.2 Military Facilities

There are no military facilities close enough to the plant to be of concern.

5.5.4.3 Pipeline Accidents

There are natural gas pipelines that pass within 2,000 feet of the plant (Figure 5-3). Analyses of various effects of natural gas pipeline leaks in the vicinity of the plant conclude that natural gas leaks cannot cause concentrations in the control room intake high enough to result in explosive conditions. Since natural gas serves as an asphyxiant danger, the concentration must be even higher than the explosive limits to pose a toxic danger. Since explosions have been ruled out for natural gas pipeline leaks, toxic hazards can be eliminated also. Thus the pipelines do not cause a hazard to the plant.

5.6 Conclusions

This review of the impact on PNPP of other external events leads to the conclusion that there are no significant events of concern. A comprehensive screening analysis of those other external hazards identified in the PRA Procedures Guide resulted in supporting the NUREG-1470 conclusion that only high winds, external floods, and transportation and nearby facility accidents required to be reviewed in detail. The plant as designed meets the intent of the criteria of the Standard Review Plan of 1975 and thus it is not surprising that these events do not pose a significant threat to the plant.

5.7 References

5-1 NUREG-1470, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

5-2 Ravindra M.K., and Bannon, H., "Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development," NUREG/CR-4839, July 1992.

5-3 NUREG/CR-2300, "PRA Procedures Guide," January 1983.

5-4 Kennedy, R.P., Blejwas, T.E., Bennett, D.E., "Capacity of Nuclear Power Plant Structures to Resist Blast Loadings," NUREG/CR-2462, September 1983.

5-5 NUREG-0800, "Standard Review Plan," USNRC, Rev.2, July 1981.

5-6 Perry Nuclear Power Plant, Units 1 and 2, USAR.

5-7 Nineteenth Annual Report of the Meteorological Program at the Perry Nuclear Power Plant, January 1, 1994 through December 31, 1994.

5-8 USNRC, Information Notice 93-53, Supplement 1: Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned, April 29, 1993.

5-9 NOAA HMR No. 52, Application of Probable Maximum Precipitation Estimates - United States East of the 105th Median, August 1982.

5-11 CEI Calc. No. CN-1.1, dated 1/8/74.

5-12 USNRC Regulatory Guide 1.70, Rev 3, November 1978.

5-13 SB 95-03, National Transportation Safety Board, January 19, 1995

5-14 NTSB/ARG-94/02, Annual Review of Aircraft Accident Data U.S. General Aviation Calendar Year 1992, Government Accession No. PB94-181054 (Attachment 3).

5-15 NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," U.S. Nuclear Regulatory Commission, 1987.

5-16 NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release," June, 1979.

5-17 USNRC letter to TVA, "TVA backfit claim regarding the NRC's safety evaluation of the potential impact of hazardous chemicals transported by barge upon habitability of the control room at Brown's Ferry," 11/20/90.

5-18 NUREG/CR-2650 "Allowable Shipment Frequencies for the Transport of Toxic Gases near Nuclear Power Plants," October, 1982.

5-19 Lewis, Richard J., Sax's Dangerous Properties of Industrial Materials, Eighth Edition, Von Nostrand Reinhold Co., New York, 1992

5-20 CRC Handbook of Chemistry and Physics, 61st Edition, 1981.

5-21 American Conference of Governmental Industrial Hygienists, "Industrial Ventilation, A Manual of Recommended Practice," 19th Edition, 1986.

5-22 Deleted.

5-23 USNRC Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," June, 1974.

5-24 1993 Emergency Response Guidebook, U.S. Department of Transportation, Research and Special Programs Administration, RSPA 5800.6, October, 1993.

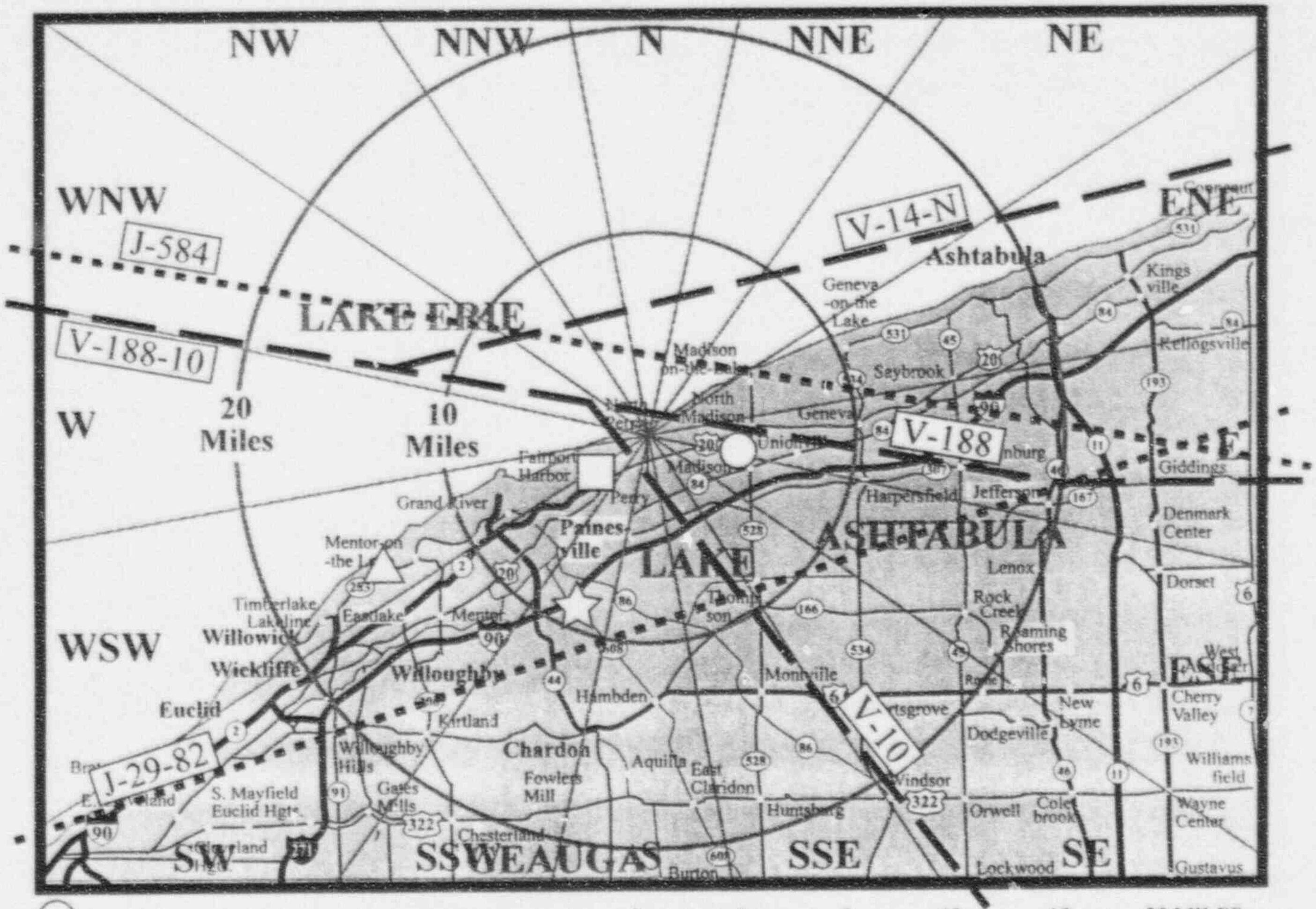
Table 5-1 - Screening of External Events for PNPP

Event	Applicable Screening Criteria
Avalanche	3
Biological Events	5
Coastal Erosion	5
Drought	2,5
Fire	3
Fog	4 (principal impact is on transportation)
Forest Fire	3
Frost	1
Hail	1,4
High Tide, High Lake	4 (external flooding)
High Summer Temperature	1,5
Ice Cover	4, 5
Landslide	3
Lightning	4 (included in loss of offsite power events)
Low Lake or River Water Level	4, 5
Low Winter Temperature	1,5
Meteorite	2
Pipeline Accident	1, 4
Intense Precipitation	4 (treated under external flooding)
River Diversion	3
Sandstorm	3

Table 5-1 - Screening of External Events for PNPP (Continued)

Event	Applicable Screening Criteria
Seiche	4 (included in external flooding)
Snow	1
Soil Shrink-Swell Consolidation	1
Storm Surge	4 (included in external flooding)
Tsunami	3
Toxic Gas	(included in transportation and nearby facility hazards)
Turbine Generated Missiles	2
Volcanic Activity	3
Waves	3, 4

Figure 5-1- Airways and Airports in the Vicinity of PNPP



◯ Woodworth Airstrip

□ Casement Airport

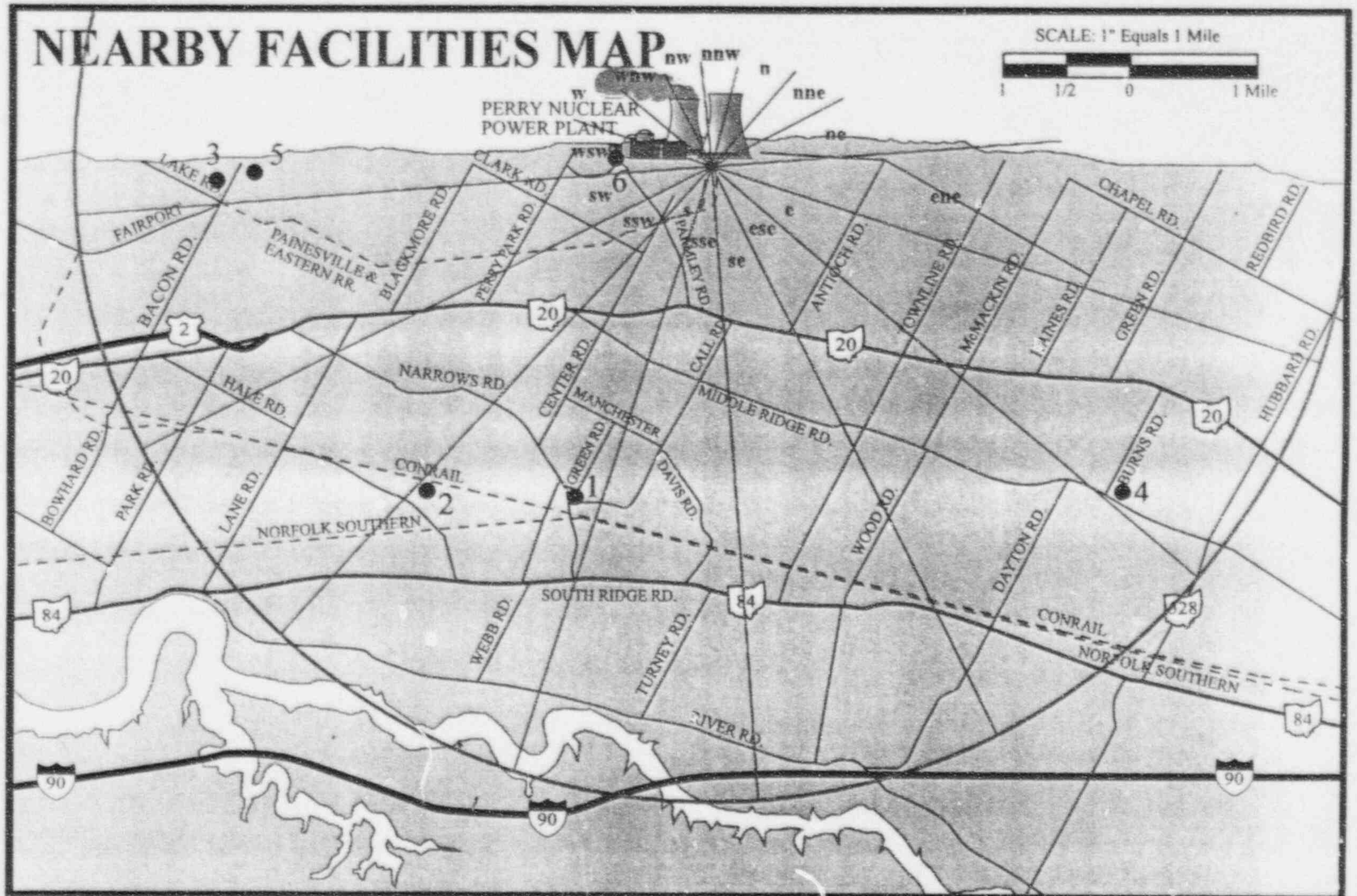
☆ Concord Airport

△ Lost Nation Airport

5 0 5 10 15 20 MILES

Selected Air Facilities and Airways in the Region

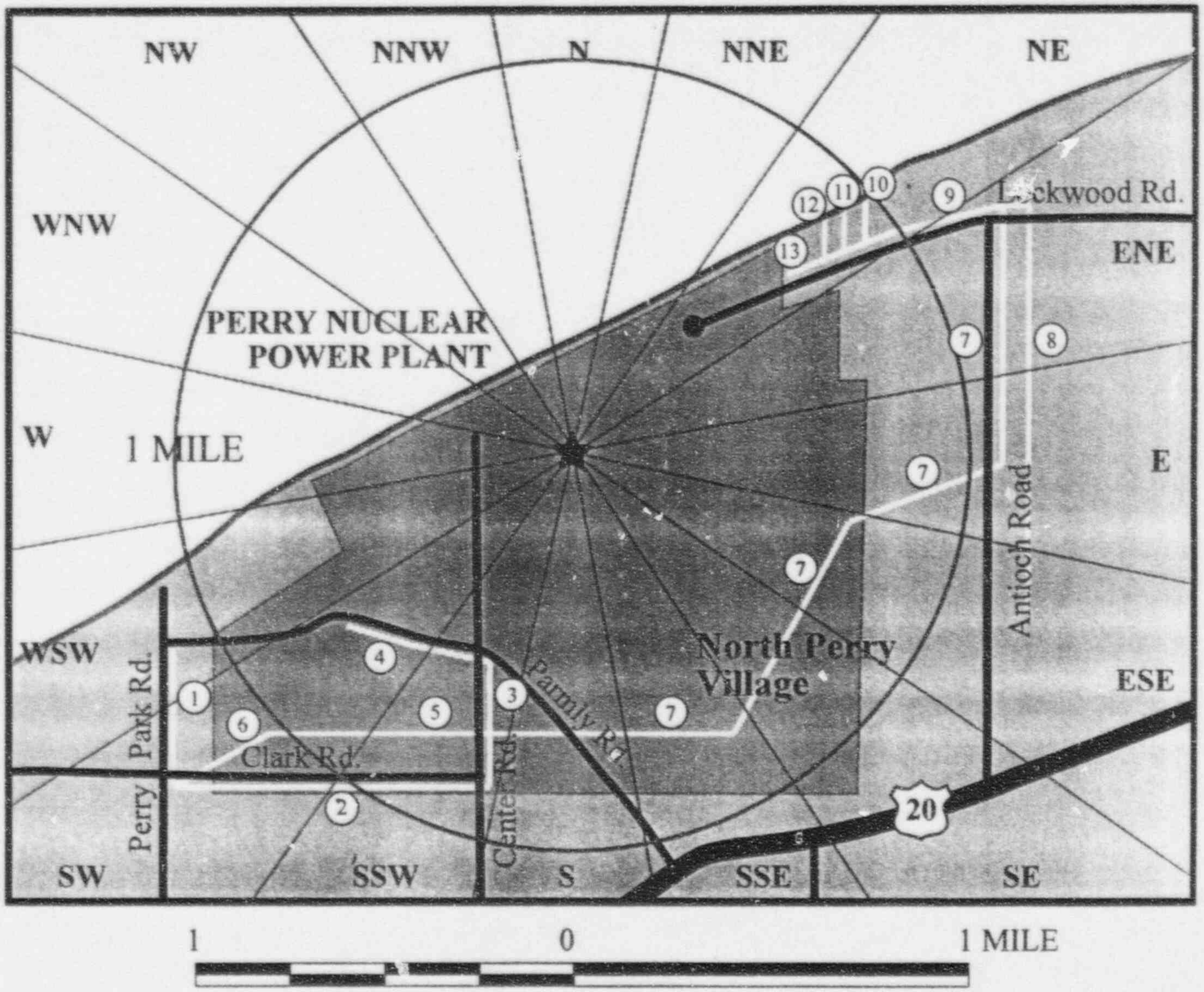
Figure 5-2 - Nearby Industrial Facilities and Transportation Routes



1. Alltell - Perry Exchange
2. Zeneca Chemicals
3. Lake County Water East
4. Madison Village Sewer Plant
5. PET Processors
6. Neff-Perkins

NEARBY FACILITIES MAP

Figure 5-3 - Gas Pipelines in the Vicinity of PNPP



Natural Gas Pipeline Map

6 LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

Cleveland Electric Illuminating (CEI) personnel were involved in both the development of the IPEEE and review of the output documents and submittal. Consultants were retained to provide specialized expertise and an overview of the IPEEE project.

6.1 IPEEE Program Organization

The IPEEE team was put together to optimize CEI resources while meeting the requirements in Generic Letter 88-20, Supplement 4. The organization structure for the IPEEE is shown in Figure 6-1. Overall management of the IPEEE was provided by the CEI Project Engineer. Specific direction for each of the three broad areas of investigation came from the CEI Seismic Engineer, CEI Other External Hazards Engineer and the CEI Fire Protection Engineer. The analyses performed for the IPEEE were generated by both CEI and consultant personnel.

The consultants were retained to provide assistance in the performance of the IPEEE and to provide additional technical expertise in the areas of the EPRI FIVE methodology and the EPRI Seismic Margins Methodology. In addition to the management role of CEI, CEI personnel also participated in the analyses produced during the course of the IPEEE effort. All output documents were reviewed by CEI personnel.

This approach ensured that the level of both technical expertise and plant familiarity was high. The participation of CEI personnel in either the production of analyses or the review of completed products resulted in a good technical transfer in all areas.

6.2 Composition of Independent Review Team

As noted above, CEI personnel performed an in depth review of all output documents to verify both the technical adequacy of the analyses and the applicability to the Perry plant design. In addition to this review, Vectra Technologies was retained to perform an overview of the analyses performed to ensure that all areas were adequately addressed and that the analyses met the requirements embodied in Generic Letter 88-20, Supplement 4 and NUREG-1407. For the seismic walkdown, Dr. Paul Smith of The Readiness Operation and Mr. Harry Johnson of Programmatic Solutions were retained to provide "on the spot" review and advice on the walkdown results. These gentlemen are recognized industry experts in this area.

The organization structure for the IPEEE peer review is shown in Figure 6-2. Overall peer review for the seismic effort was performed by Charbel Abou-Jaoude of Vectra Technologies. Mr. Abou-Jaoude has over 10 years experience in this field. Mr. Abou-Jaoude was also responsible for performing he peer review of high winds, external floods and other external hazards with the assistance of Greg Ashley. The peer review for internal fires was performed by Debby de la Cruz and Staci Thompson of Vectra Technologies. Ms. de la Cruz has 9 years of experience in the field of fire protection. Ms. Thompson has 5 years experience in external events analysis and fire protection.

6.3 Areas of Review and Major Concerns

6.3.1 Seismic Margins Assessment

Each of the output documents such as development of the SSEL, Screening Evaluation Work Sheets (SEWS), HCLPF calculations, etc. were reviewed by Mr. Abou-Jaoude after they were developed. Comments were received from Mr. Abou-Jaoude and addressed by CEI personnel. The SEWS were also reviewed by either Dr. Smith or Mr. Johnson at the time the equipment was walked down. This enabled their comments to be addressed while the walkdown teams were still intact and the equipment in the field was still accessible.

In general, no major deficiencies were identified in the process used in the Perry seismic margins assessment. Additional detail was recommended for some of the SSEL calculations. The reviewer suggested that some refinement of the needed systems and equipment for a seismic event could reduce the number of components on the SSEL.

Additional detail was also recommended on the SEWS generated during the seismic walkdown by both the Vectra reviewer and Dr. Smith and Mr. Johnson.

The methodology used to determine the HCLPFs was deemed appropriate. One comment was received regarding the justification for screening out one of the components. An additional analysis was performed to verify that he component could be screened.

The remainder of the comments were minor.

6.3.2 Internal Fires

The peer review for internal fires was performed by Ms. de la Cruz and Ms. Thompson. The EPRI FIVE results, the fire ignition frequency calculation, the conditional core damage probability (CCDP) calculation, and several detailed fire analyses were reviewed by the peer reviewers. A summary of the review concluded that, "In general, the documentation was found to be consistent with the FIVE guidelines and thorough, particularly in the area of Detailed Fire Analysis. The fault tree modeling approach utilized was effective in isolating fire scenarios and reducing fire frequency values. Suggestions were made to include additional items which may eliminate potential questions and provide clarity. Additional discussion is recommended in the areas of plant trip initiators, manual suppression, and simplified fire modeling..."

The documentation was substantially improved by developing formal calculations. Minor technical issues and corrections were addressed during the development of the formal calculations. No major deficiencies were identified.

6.3.3 High Winds, External Floods, and Other External Hazards

The calculations developed for the other external hazards were reviewed by Mr. Abou-Jaoude and Greg Ashley. A summary of the review stated that, "Based on the review of the [calculations], the Peer Reviewer finds that the "other" events portion of the PNPP IPEEE has been performed in a very thorough and conservative manner. The guidance provided by NUREG-1407 has been adopted for this evaluation and the objectives of the NUREG met."

Four observations were offered to clarify or strengthen the conclusions of the "Other" External Hazards portion of the IPEEE. The observations were incorporated into the calculations.

6.4 Resolution of Comments

Resolution of the review comments is discussed above in Section 6.3.

Figure 6-1 - IPEEE Project Organization Chart

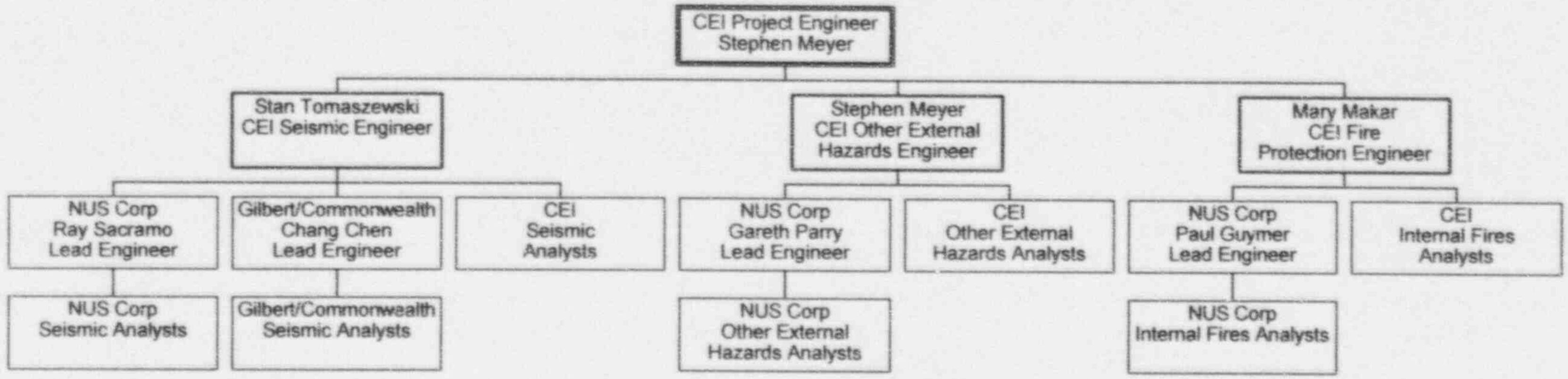
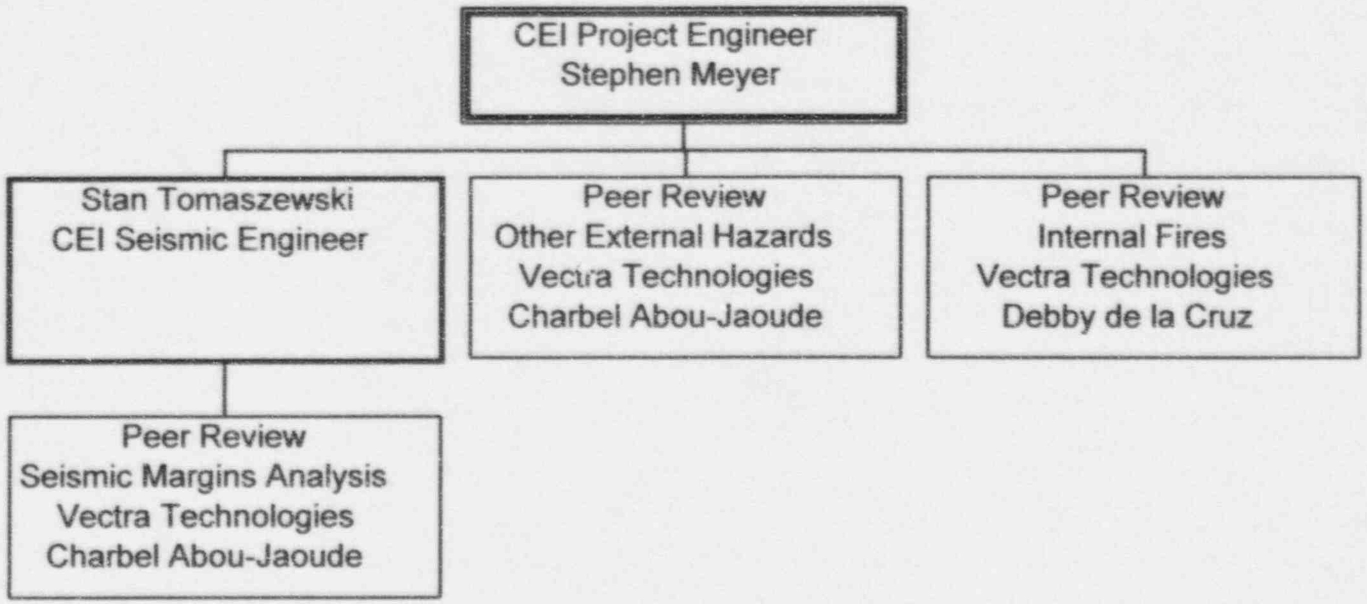


Figure 6-2 - Peer Review Organization



7 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

7.1 Unique Safety Features

There are a number of unique features at Perry which help to safely shut the plant down from certain initiating events.

The diesel generator which supplies the HPCS system is not the same size or design as the two standby diesel generators supplying the Division 1 and 2 emergency buses so that a common cause failure of all three diesel generators is less likely to occur. Similarly the differences in design between the emergency service water train for HPCS and the other two emergency service water trains will lessen the probability of a common cause failure of all three trains.

The ability exists and is proceduralized to cross connect the electric supply from the HPCS diesel generator to the Division 2 emergency bus. This enables the inboard containment isolation valves (which include the valves used to vent the containment) to be powered in the event of a loss of offsite power from an external event and the failure of both the Division 1 and 2 diesel generators.

7.1.1 Seismic Analysis

Sufficient diversity exists at Perry such that Success Path A is nearly completely separate from Success Path B, i.e., there are very few common components for the RPV level control systems and decay heat removal systems between Success Paths A and B. For Success Path A, RPV level control is accomplished with the HPCS system and decay heat removal by venting the containment via the FPCC system. Power may be supplied to the FPCC containment isolation valves by cross-tying the HPCS diesel generator to the Division 2 emergency bus. For Success Path B, low pressure ECCS using any one of the three LPCI trains or LPCS provides RPV level control. The bypass around the RHR heat exchangers can be isolated to provide decay heat removal capability or the containment can be vented via the containment spray headers and the RHR system piping.

The only common links between Success Paths A and B are some Division 2 emergency buses which may be used for containment venting, reactivity control and RPV overpressure protection.

7.1.2 Internal Fire Analysis

The Perry design incorporates good physical separation of the different electrical divisions. Existing housekeeping and hot work procedures minimize the risk from ignition sources. This provides the defense in depth desired for safe plant shutdown and decay heat removal.

7.1.3 High Winds, Floods, and Others

The key to minimum risk from other external hazards is the structures housing the equipment needed to safely shutdown the plant and removal decay heat. In addition to meeting the intent of the Standard Review Plant, the design of the structures at Perry is robust. The ability to withstand the postulated external events was built into the plant to protect the operators and needed equipment.

7.2 Plant Improvements

There were no vulnerabilities identified during the performance of the IPEEE. However, as part of the seismic analysis, several items were identified which could increase the resistance to relay chatter from spatial interactions during a seismic event. No enhancements for internal fires or other external hazards were identified.

7.2.1 Seismic Analysis

The seismic margins analysis resulted in four outliers that did not screen out upon further analysis. These outliers and their resolutions were all from spatial interactions and are considered housekeeping items. No physical plant modifications were found to be needed.

Maintenance Test Bench

A maintenance test bench is located in the Unit 1, Division 2 Switchgear Room. The test bench is mounted on rollers so that it can be easily moved from one location to another for use during testing and maintenance activities. The potential exists that during a seismic event, the test bench could roll and impact motor control center MCC EF1D08. The resolution to this issue is to provide locking wheels for the test bench.

Operations Electrical Locker

An operations electrical locker is located in the Unit 1, Division 2 Switchgear Room near the panel containing Bus EH12 (1R22-S0006). The locker is 84" in height and would come close to impacting 1R22-S0006 if it were to overturn during a seismic event. To provide assurance that there would be no possibility of striking 1R22-S0006 upon overturning, the resolution is to replace the tall narrow locker with two shorter lockers. The shorter lockers would be less likely to overturn and even if they did, the distance to 1R22-S0006 would preclude impact.

Switchgear Trolleys

Trolleys are mounted on top of the 480 VAC switchgear panels to support the hoists used in removal and installation of the breakers. The trolleys are not restrained from side to side movement and potential exists for the lifting hooks to swing and strike the panel face during a seismic event. Based on the evaluation conducted, the trolleys, if parked in the center of the panel, would provide sufficient margin from moving along the rails and impacting the stops. If the lifting hook is restrained by the trolley frame, the potential for swinging into the panel face is eliminated. A caution on parking the trolleys and restraining the lifting hook are being placed in the appropriate instructions.

Control Room Furniture

During the seismic walkdown of the control room numerous potential spatial interactions were identified. During the subsequent evaluation, the majority of these potential interactions were screened out leaving thirteen items. Relocation of these items is being explored to resolve this outlier. This effort may involve relocation of some items by only a few inches while others may need to be relocated to less sensitive places in the control room.

7.2.2 Internal Fire Analysis

Seven fire area/compartments remained above the screening criteria of 10^{-6}/yr following the detailed fire analyses. Five of the fire area/compartments have fire induced core damage frequencies close to the screening criteria. Fire areas 1CC5a (Unit 1 control Room) and 1CC3a (Unit 1 Division 2 Switchgear Room) dominate the fire induced CDF. By its very nature a fire in the control room (1CC5a) has the potential to result in a core damage frequency above 10^{-6}/yr. No additional actions were identified which would further reduce the contribution to the CDF of a control room fire. The fire induced CDF in the Unit 1 Division 2 switchgear room is dominated by the fires in the 4.16 kV switchgear and the 480 V bus sections which damage overhead cable. Conservatisms in the analysis are discussed in Section 4.7.3.2 of this report. No additional actions were identified which would further reduce the contribution to the CDF of a fire in 1CC3a.

7.2.3 High Winds, Floods, and Others

There were no open items at the end of the screening process and subsequent analyses. No plant improvements were made as a result of the high winds, floods, and other external hazards evaluations performed for Generic Letter 88-20, Supplement 4.

8 SUMMARY & CONCLUSIONS (including proposed resolution of USIs, GIs and other issues)

Cleveland Electric Illuminating Co. (CEI) performed an Individual Plant Examination of External Events (IPEEE) for the Perry Nuclear Power Plant (PNPP). The Perry IPEEE meets the intent of Generic Letter 88-20, Supplement 4 and NUREG-1407. Perry was found to be rugged with respect to seismic events, internal fires and high winds, floods and other external hazards. Management of the project was retained within CEI. The analyses themselves were performed by a combination of CEI and consultant personnel.

8.1 Seismic Analyses

A seismic margins assessment was performed for Perry using the EPRI seismic margins methodology. Perry was analyzed as focused scope plant per the guidance contained in Generic Letter 88-20, Supplement 4 and NUREG-1407. The review level earthquake was anchored at 0.3g.

Two safe shutdown paths (both of which can mitigate a small LOCA) were developed and found to have a high confidence of a low probability of failure (HCLPF) of 0.3g. In addition, the containment also was found to have a HCLPF of 0.3g.

Four enhancements to reduce the threat of spatial interactions were identified and are in the process of being implemented. No plant modifications were required.

8.2 Internal Fire Analyses

The internal fire analysis was performed using the EPRI Fire-Induced Vulnerability Evaluation (FIVE). The progressive screening used in the FIVE methodology left nine fire zones which required additional detailed evaluation. Based on the detailed evaluations performed each of the remaining fire zones were found to be acceptable with no enhancements or modifications required.

8.3 High Winds, Floods, and Other External Hazards

The approach used for high winds, floods and other external hazards follows the method described in NUREG/CR-4839 by performing an initial screening analysis followed by bounding or more detailed analyses as necessary. A comprehensive review of external hazards based on those identified in the PRA Procedures Guide supported the NUREG-1407 conclusion that only high winds, external floods and transportation and nearby facility accidents needed a detailed evaluation. The plant as designed was found to meet the intent of the criteria of the Standard Review Plan of 1975. No enhancements or modifications were made as a result of the other external hazards analyses performed for Generic Letter 88-20, Supplement 4.

8.4 Resolution of USIs, GIs and Other Issues

8.4.1 USI A-17 "System Interactions in Nuclear Power Plants"

The external events portion of USI A-17, "System Interactions in Nuclear Power Plants," was subsumed into the IPEEE. Seismic spatial interactions were evaluated as part of the seismic walkdown effort. The walkdowns were performed in accordance with the guidance contained in EPRI document NP-6041-SL. No seismic vulnerabilities were identified due to spatial interactions. This issue is considered complete for Perry.

8.4.2 USI A-45 "Shutdown Decay Heat Removal Requirements"

The Perry Individual Plant Examination (IPE), submitted in June 1992, addressed the impact on shutdown decay heat removal requirements of internal events. The Perry IPEEE covers external events and their impact on decay heat removal. The safe shutdown equipment list (SSEL) generated for the seismic margins analysis contains the equipment necessary to safely shut the plant down and maintain in a safe condition for 72 hours assuming no offsite a.c. power. The equipment needed to remove decay heat is included in the SSEL. No potential vulnerabilities were identified that would prevent decay heat removal processes. This issue is considered complete for Perry.

8.4.3 The Eastern U.S. Seismicity Issue (The Charleston Earthquake Issue)

The Eastern Seismicity Issue, formerly called the Charleston Earthquake Issue, was subsumed into the IPEEE. The performance of the seismic margins assessment as part of the IPEEE provides the resolution to this issue. The work on the seismic hazard estimates by the USNRC, LLNL and EPRI played a key role in developing the review level earthquake bins which set the magnitude of the earthquakes each plant was assigned to evaluate. Perry performed the seismic analyses to a review level earthquake anchored at 0.3g. No vulnerabilities were identified. Based on this, The Eastern U.S. Seismicity Issue is considered complete for Perry.

8.4.4 January 31, 1986 Leroy Earthquake Issues

Following the 1986 earthquake near Leroy, Ohio, PNPP committed to perform an evaluation to determine the level of available seismic margin. The review level earthquake for the PNPP seismic margin assessment performed as part of the IPEEE was anchored at 0.3g. This is exactly twice the safe shutdown earthquake of 0.15g utilized for the design of the plant. A HCLPF of 0.3g for the SSEL, shows that Perry has significant margin above design. This completes the Perry commitment made for the 1986 Leroy earthquake.