H. B. Robinson Steam Electric Plant Unit No. 2

Individual Plant Examination for External Events Submittal

CAROLINA POWER & LIGHT COMPANY

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H. B. ROBINSON STEAM ELECTRIC PLANT
UNIT NO. 2

IPEEE SUBMITTAL PROJECT TEAM

RISK ASSESSMENT MANAGEMENT

Rudy E. Oliver II, Manager - PWR Safety Analysis

PROJECT MANAGEMENT/TECHNICAL LEAD

Issa Zarzar/ Neil Johnson, Project Engineers - PWR Safety Analysis

TECHNICAL CONTRIBUTIONS

SEISMIC MARGIN ANALYSIS

CP&L	EQE	SAIC
Jeffrey H. Bond	Ronald W.Cushing	Michele Laur
Steven R. Bostian	Gregory S. Handy	
Martha C. Cook	Timothy Mason	
Daryl W. Hughes	Thomas R. Roche	
Ronald L. Knott		
Robert N. Panella		
Kevin N. Poythress		

FIRES AND OTHER EXTERNAL EVENTS

CP&L	NUS	
Neil Johnson	Paul Guymer	Yong S. Kim
Steve Laur	Gregory Rozga	Gareth W. Parry
Issa Zarzar	Thinh Q. Dhin	Joey Aldieri
	Kevin Elliott	Kahnu Viradia

HBRSEP PLANT/CORPORATE SUPPORT

Fred A. Emerson	Don Dykstерhouse
Chuck Baucom	Al Garrou
Richard Hightower	Jan Kozyra
Jim Townsend	Ed Roper

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

Section Page

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

TABLE OF CONTENTS (continued)

Number Page

LIST OF TABLES

TABLE OF CONTENTS (continued)

LIST OF FIGURES

LIST OF ACRONYMS

AFSS	Automatic Fire Suppression System
AFW	Auxiliary Feedwater
AOP	Abnormal Operating Procedures
ATWS	Anticipated Transient Without Scram
APCSB	Auxiliary Power Conversion Systems Branch (NRC)
BTP	Branch Technical Position
CAFTA	Computer Aided and Fault Tree Analysis (Software)
CCDF	Conditional Core Damage Frequency
CCW	Component Cooling Water
CDF	Core Damage Frequency
CFA	Critical Floor Area
CFAR	Critical Floor Area Reduction
COMPBRN	(computer code for fire analysis)
CP&L	Carolina Power & Light
CSD	Cold Shutdown
CVCS	Chemical and Volume Control System
DC	Direct Current
DG	Diesel Generator
DS	Dedicated Shutdown
DSP	Dedicated Shutdown Procedure
EDBS	Equipment Data Base System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EQE	Earthquake Engineering International Consulting Company
FDAP	Fire Door Monitoring Relay Panel
FEDB	Fire Event Data Base

LIST OF ACRONYMS (continued)

FRSS	Fire Risk Scoping Study
FS	Fire Suppression
FSAR	Final Safety Analysis Report
GERS	Generic Equipment Ruggedness Spectra
GI	Generic Issue
GIP	Generic Implementation Procedure
GL	Generic Letter
HBRSEP	H. B. Robinson Steam Electric Plant
HCLPF	High Confidence of Low Probability of Failure
HEP	Human Error Probability
HF	Hydrogen Fluoride
HGL	Hot Gas Layer
HSD	Hot Shutdown
HVAC	Heating, Ventilation and Air Conditioning
IEF	Initiating Event Frequency
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISDS	Ignition Source Date Sheets
ISLOCA	Interfacing Systems LOCA
LER	Licensce Event Report
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
MCC	Motor Control Center
MCR	Main Control Room
MOV	Motor Operated Valves
NOAA	National Oceanic and Atmospheric Administration
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center

LIST OF ACRONYMS (continued)

NSSFC	National Severe Storms Forecasting Center
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OST	Operations Surveillance Test
OEF	Operating Experience Feedback
PMP	Probable Maximum Precipitation
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
RAB	Reactor Auxiliary Building
RCP	Reactor Coolant Pump
RCS	Rreactor Coolant System
RESS	Robinson Engineering Support Section
RF	Reduction Factor
RHR	Residual Heat Removal
RLE	Review Level Earthquake
RPS	Reactor Protection System
RTGB	Reactor Turbine Generator Room
RWST	Refuelling Water Storage Tank
PVC/PE	Polyvinyl Chloride/Polyethylene
SAIC	Science Applications International Corporation
SAMG	Severe Accident Management Guidelines
SCBAs	Self-contained breathing apparatus
SCE	Seismic Capability Engineers
SEWS	Screening and evaluation worksheet
SI	Safety injection
SMA	Seismic margin assessment
SNL	Sandia National Laboratory
SPLD	Success Path Logic Diagrms

LIST OF ACRONYMS (continued)

SQUG	Seismic Qualification User's Group
SRP	Standard Review Plan
SRT	Seismic review team
SSD	Safe shutdown
SSEL	Safe shutdown equipment list
UBC	Uniform Building Code
UFSAR	Updated final safety analysis report
USI	Unresolved Safety Issue
WNW	West northwest
WSW	West southwest

SECTION 1

EXECUTIVE SUMMARY

1.0 BACKGROUND AND OBJECTIVES

This report was developed in response to the Nuclear Regulatory Commission (NRC) request for each licensee to perform an Individual Plant Examination of External Events (IPEEE) for each of its nuclear plants, as detailed in Generic Letter (GL) 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" issued in April 1991 (NRC, 1991a). With the performance of the work described in this report Carolina Power and Light (CP&L) Company has fulfilled all the objectives of the Generic Letter for its H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The principal objectives of the IPEEE as outlined in GL 88-20 are, for the case of external initiating events:

- to develop an appreciation of severe accident behavior,
- to understand the most likely severe accident sequences that could occur at the plant under full-power conditions,
- to gain a qualitative understanding of the overall likelihood of core damage and fission product release, and
- if necessary, to reduce the overall likelihood of core damage and fission product release by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

In NUREG 1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" (NRC, 1991b), the NRC specifically identified the following external events as those to be included in the scope of the IPEEE:

- seismic events,
- internal fires,
- high winds and tornados,
- external floods, and
- transportation and nearby facility accidents.

All other previously identified external events are considered to be, in all likelihood, screened out. However, it was requested that a search for unique and plant specific events be made.

This document summarizes the results of the IPEEE for HBRSEP in a manner consistent with the submittal guidance provided in GL 88-20 and in NUREG 1407.

In response to the original Generic Letter 88-20, published in November 1988 (NRC, 1988), CP&L published its Individual Plant Examination (IPE) (CP&L, 1992), which addressed the risk from internal initiating events including internal flooding. This report builds upon the plant models created for that study and the subsequent updates to those models.

The information provided in this submittal is based on the plant configuration that existed at the end of refueling outage 15 (March 1994) and is backed by extensive documentation in the form of analysis notebooks. The organization of the documentation is designed to support a detailed review of the analysis.

This executive summary provides a brief description of the study and its results. Section 2 of this report is a description of the overall scope of the IPEEE and a summary of the methods used for the analysis. Section 3 provides a summary of the analysis of seismic events; a more detailed description of the analysis is provided in Appendix A. Section 4 describes the analysis of the risks from fires, and Section 5 the analysis of the risks from all other external events. Section 6 describes CP&L participation in the project. Section 7 discusses the plant improvements that have been identified as a result of this investigation, and Section 8 provides a summary and overall conclusions. References are provided separately at the end of each section.

1.1 PLANT FAMILIARIZATION

The H. B. Robinson Steam Electric Plant, Unit No. 2 is located near Hartsville, South Carolina, sharing the site with the Unit No. 1 fossil plant. The Unit began commercial operation in 1971. The nuclear steam supply system (NSSS) is a three-loop Westinghouse design and is rated at 2300 megawatts thermal. The NSSS is enclosed by a large, dry, reinforced concrete, steel-lined containment. Ebasco was the architect engineer.

The performance of the IPEEE requires additional knowledge of the plant over and above that which was required for the performance of the IPE. In particular, the physical characteristics of the plant, including detailed knowledge of the location of equipment and details of its anchorage is required. This plant familiarization was brought into the project by involving engineers with detailed specialized knowledge, for example fire protection engineers, structural engineers, and also by performing walkdowns. Several walkdowns with different objectives were performed, some to address seismic issues, others to address fire related issues, and one to confirm the general characteristics of the plant and the site.

1.2 OVERALL METHODOLOGY

The IPEEE for HBRSEP was performed using methods identified in NUREG 1407 (NRC, 1991b). The specific methodologies for analyzing seismic events, fires, and other external events are summarized below.

1.2.1 Seismic Analysis

The analysis of seismic events was achieved by performing a Seismic Margins Assessment (SMA). This method, unlike that used in the IPE and that used for fire events analysis does not result in estimates of core damage frequency. Instead, it is an analysis to assess whether the plant is designed and constructed so that it can be safely shut down following what is known as a Review Level Earthquake (RLE). The methodology for performing the Seismic Margin analysis has been described in "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" (EPRI, 1991), and is briefly summarized here.

The first step is to define a set of equipment that will be sufficient to safely shutdown the plant. A walkdown is performed to identify items on that list that do not appear to be sufficiently seismically rugged. Items which are not screened out by the walkdown are dispositioned in one of several ways, as follows:

- resolved through housekeeping or maintenance,
- resolved through repair or modification,
- resolved through further investigation, and
- resolved through a demonstration that the acceleration corresponding to the High Confidence of Low Probability of Failure (HCLPF) is higher than that of the Review Level Earthquake.

Detailed plant walkdowns are considered the most cost-effective and beneficial aspect of the SMA program. Because resolution of USI A-46 also requires seismic walkdowns, combined A-46 and IPEEE in-plant reviews were performed by teams of CP&L and consultant Seismic Review Teams in accordance with the Seismic Qualification Utility Group Generic Implementation Procedure, with enhancements based on EPRI NP-6041.

1.2.2 Fires

The analysis of the impact of fires was performed using a fire PRA approach. This method provides an estimate of core damage frequency for a set of limiting fire initiated scenarios, identified using a systematic screening approach. The analysis considers the likelihood of fire occurrence in each plant area and its subsequent impact on plant systems. Equipment damage resulting from the thermal effects of fire (conductive, radiative and convective) are considered as well as the degradation of operation reliability. Potential vulnerabilities raised in the Fire Risk Scoping Study (NRC, 1989) related to seismic/fire interactions, effects of suppressants on safety equipment and control system interactions are addressed through specifically tailored walkdowns, as defined in the EPRI FIVE methodology (EPRI, 1992).

The fire evaluation was performed on the basis of fire areas which are plant locations completely enclosed by rated fire barriers. The fire area boundaries were assumed to be effective in preventing a fire from spreading from the originating area to another area based on the implementation of a satisfactory fire barrier surveillance and maintenance program. The fire area boundaries recognized in this study are identical to those identified in the plant's 10 CFR 50 Appendix R report (CP&L, SSD). In some cases these fire areas were further subdivided into compartments. For analysis purposes, for the more significant compartments, fire damage states within those compartments were defined that identified subsets of the equipment within the compartments as being damaged due to the fire.

The analysis was conducted in three main stages as follows:

Stage 1 was a systematic qualitative and quantitative screening analysis of all plant fire areas/zones. The screening analysis was based largely on information already available in the plant's 10 CFR 50 Appendix R report (CP&L, SSD) and the IPE study (CP&L, 1992). At this stage all equipment and cable in an area/compartment were assumed to be damaged. The damage was assessed qualitatively to determine if the effects were significant; that is, would the fire cause an initiating event or lead to loss of accident mitigating equipment. Areas/compartments not screened out qualitatively were then subject to a determination of their associated fire frequency (F_1) and conditional core damage frequency (P_2), given loss of all functions which may be impacted by the fire. If the resulting fire induced core damage frequency (F_1 x P_2) was less than 1.0E-6 per year the area/ compartment was screened out.

Stage 2 was a detailed evaluation of the fire areas/compartments which did not previously screen out, using fire PRA techniques as well as methods and data provided in the FIVE technical report. The principal difference in this stage of the analysis is that the resulting impact of intermediate fire growth stages within each fire area/compartment was assessed (rather than assuming the contents are immediately damaged).

Stage 3 was an evaluation of the Sandia Fire Risk Scoping Study Issues using a tailored walkdown approach.

1.2.3 Other External Events

Since the plant was built prior to the issuance of the 1975 Standard Review Plan (NRC, 1975), the analysis of the remaining group of external events, the Other External Events, was performed by the successive screening approach outlined in NUREG-1407. A detailed evaluation of the impact of extreme winds was performed.

1.3 SUMMARY OF MAJOR FINDINGS

1.3.1 Seismic Events

Results of the seismic margins assessment are grouped in three categories as follows:

- Housekeeping/Maintenance Issues

Thirty-three items were identified as outliers requiring minor maintenance that could be repaired by a work ticket. Items that can be repaired by work tickets are typically those items whose conditions have only slightly degraded from the original design intent and can be fixed by using existing plant drawings, procedures, or guidelines. The repair can be implemented by maintenance or construction without any engineering input or review. It usually involves a replacement of like hardware, torquing of bolts, etc. These items are listed in Tables 3-1 and 3-2.

- Repairs/Modifications

Twenty-two items were identified as outliers and the Seismic Review Team determined that additional calculations would potentially not resolve the outlier issues. They concluded that the twenty-two items would best be resolved by the implementation of physical plant modifications. Modifications provide the vehicle to change components using engineering review and input. These components are listed in Table 3-3.

- Raceway Repairs/Modifications

Sixteen issues involving electrical raceway installations were identified as requiring work ticket/maintenance or modifications in order to restore the reported item to an acceptable condition.

All 789 relays on the HBRSEP essential relay list have been accepted by either capacity screening or system consequence screening.

Twenty items were evaluated using the High Capacity for Low Probability of Failure methodology. These items are identified in Table 5-3 of Appendix A to this report.

Maintenance repairs, modifications, and outlier resolutions will be completed by the end of refueling outage 18 (currently scheduled for 1998).

1.3.2 Fires

In total, twenty-three scenarios that have contributions to core damage frequency greater than 1.0E-6 were identified. They are summarized in Table 8-1. The overall annual core damage frequency due to fires was 2.22E-04. There are five scenarios with contributions to CDF greater than 1.0E-5. They are:

- Scenario 16-1. This is a fire originating in battery room A-16 in MCC-A or MCC-B, with failure of manual suppression, leading to a loss of train A and B DC power. This scenario is significant because the MCCs have several open conduits, and the battery room is small. Therefore, damage to the redundant MCC due to the formation of a hot gas layer is possible in a short time, such that manual supression may not be possible.

- Scenario 20-16. This is a fire originating in the emergency switchgear room that leads to loss of the E2 480 V bus. This fire is significant because offsite power is lost and it has a high initiating event frequency.

- Scenario 22-3. This is a fire in RTGB cabinet D that is suppressed within the cabinet. This is significant because it is a control room fire and requires evacuation.

- Scenario 22-4. This is a fire in RTGB B, C, D or E that propagates to other RTGB cabinets. This is significant because it is a control room fire and requires evacuation.

- Scenario 26-1. This scenario results from an explosive transformer fire in the switchyard that results in a loss of offsite power and the dedicated shutdown diesel generator. The transformers of concern, because of their proximity to a conduit associated with the DS diesel which is routed on the outside of the turbine building, are the auxiliary and start up transformers.

1.3.3 Other External Events

No major vulnerabilities were identified. However, the prolonged operation of the emergency diesel generators, which would in all probability be required following the occurrence of extreme winds in the vicinity of the plant, could be compromised by the fact that the fuel oil transfer pumps are unprotected from missiles. The day tanks for the diesel generators are of limited capacity, allowing only about 90 minutes of operation. The frequency of scenarios leading to the simultaneous loss of offsite power and damage to the fuel oil transfer pumps was estimated to be on the order of 2.0E-06 per year.

1.3.4 Resolution of Unresolved Safety Issues

By performing this IPEEE, CP&L has not only addressed the requirements of the GL 88-20, Supplement 4 (NRC, 1991a), but has also addressed other regulatory requirements.

Three programs are subsumed in the IPEEE: (1) the external event portion of USI A-45 "Shutdown Decay Heat Removal Requirements," (2) GI-131 "Potential Seismic Interaction

Involving the Moveable In-Core Flux Mapping System used in Westinghouse Designed Plants," and (3) the Charleston Earthquake issue.

As discussed in the IPE submittal, loss of decay heat removal is inherently considered in a PRA evaluation of core damage frequency. In the fire analysis, significant fire areas were identified on the basis of their contribution to core damage frequency. The significance of an area is not tied to the decay heat removal issue per se; however, resolution of issues arising from an identification of these areas as significant will resolve any issues related to USI A-45. In the seismic analysis, equipment necessary to achieve long term heat removal is included on the safe shutdown paths. Therefore, an acceptable result of the seismic margin assessment also reflects a favorable outcome for the USI A-45 resolution.

The Eastern U.S. Seismicity Issue is resolved by the seismic part of the IPEEE. Since CP&L exercised the seismic margins option, the resolution was achieved by an appropriate choice of review level earthquake. GI-131 deals with the seismically induced failure of the flux mapping transfer cart that would lead directly to the rupture of instrumentation tubes at the seal table. Since this is applicable to Westinghouse plants, it is applicable to HBRSEP. It has been addressed in the IPEEE. USI A-46 has subsumed USI A-17, "Seismic Interactions in Nuclear Power Plants," USI A-40 "Seismic Capability of Large Safety-Related Above-Ground Tanks," and GI-57 "Effects of Fire Protection System Actuation on Safety Related Equipment." The USI A-46 resolution was performed at the same time as the IPEEE, and is reported on separately as a response to GL 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46."

The Fire Risk Scoping Study (FRSS) Issues, NUREG/CR-5088, were examined through comparison to standardized checklist questions and through specifically tailored plant walkdowns according to the FIVE Methodology. The FRSS issues are discussed in Section 4.8. The issue of seismic-fire interactions raised in the FRSS and in Information Notice 94-12, "Insights Gained from Resolving Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment," has been addressed and is discussed in Sections 3.1.6 and 4.8.1.

The revised "Design Probable Maximum Precipitation (PMP)" criteria were assessed within the Other External Events Task as described in GL 89-22, "Resolution of Generic Safety Issue No. 103, 'Design for Probabale Maximum Precipitation'." The conclusions are presented in Section 5.4.

Information Notice 93-53, Supplement 1, "Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned," (NRC, 1993) requested that the IPEEE address the lessons learned . This was addressed during the performance of a walkdown that was conducted to confirm the conclusions of the review of the plant design with respect to Other External Events, as discussed in Section 5.

1.3.5 Resolution of Issues

As described in Section 6.3, a multi-disciplinary team was established to evaluate the IPE results and suggest areas for potential improvements. The recommendations include some immediate changes and areas for further investigation to see if other changes are cost-effective. The results of the evaluation are summarized in Section 7.

1.4 REFERENCES

(CP&L, 1992), Carolina Power and Light Company, "H. B. Robinson Steam Electric Plant, Unit No. 2, Individual Plant Examination Submittal", August 1992.

(CP&L, SSD), Carolina Power and Light Company, "H. B. Robinson Long Term Compliance, 10 CFR 50, Appendix R, Safe Shutdown Separation Analysis, FPP-RNP-600.

(EPRI, 1991), Electric Power Research Institute, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin, Revision 1", EPRI NP-6041, August 1991.

(EPRI, 1992), Electric Power Research Institute, "Fire-Induced Vulnerability Evaluation (FIVE) Method for Internal Fire", EPRI TR-100370, RP 3000-41, April 1992.

(NRC, 1988), USNRC, Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)", November 23, 1988.

(NRC, 1989), USNRC, "Fire Risk Scoping Issues", NUREG/CR-5088. January 1989.

(NRC, 1991a), USNRC, Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)", April, 1991.

(NRC, 1991b), USNRC, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", NUREG 1407, June 1991.

SECTION 2

EXAMINATION DESCRIPTION

2.0 INTRODUCTION

As part of the implementation of the Severe Accident Policy, the US Nuclear Regulatory Commission issued Generic Letter (GL) 88-20 on November 23, 1988, requesting that each licensee conduct an individual plant examination (IPE) for internally initiated events including internal flooding (NRC, 1988). To comply with the generic letter, the IPE report for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 was submitted in August 1992 (CP&L, 1992). In GL 88-20, Supplement 4 (NRC, 1991a), the NRC requested that the licensee extend its examination to include what have become known as External Initiating Events. This report presents the Individual Plant Examination of External Events (IPEEE) for HBRSEP in response to that request. The general objectives of the IPEEE are similar to that of the IPE, namely (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at the plant under full-power conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage and fission product release, and (4) if necessary, to reduce the overall likelihood of core damage and fission product release by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

This section demonstrates that the analysis conforms with the NRC requirements for a response to Supplement 4 of the generic letter, and contains a brief description of the methodology and the information used in the course of the study.

2.1 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

Carolina Power and Light (CP&L) Company has performed an IPEEE pursuant to 10 CFR 50.54, as invoked by GL 88-20, Supplement 4.

The IPEEE generic letter and report guidance document, NUREG-1407, 0Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities,1 (NRC, 1991b) requires licensees to consider five specific external events in performing their IPEEE: Seismic events, internal fires, high winds, floods (external), and transportation and nearby facility accidents. Licensees are also required to confirm that no other plant unique external events, with potential for severe accidents, are being excluded. The IPEEE subsumes aspects of several ongoing NRC programs, such as A-45 "Shutdown Decay Heat Removal Requirements," GI-131 "Potential Seismic Interaction Involving the Moveable In-Core Flux Mapping System used in Westinghouse Designed Plants," and the Eastern U.S. Seismicity issue, also known as the Charleston Earthquake issue; their resolution is therefore required to be explicitly addressed in the IPEEE response.

Consideration of the specific provisions of the generic letter is provided in the following sections.

2.1.1 Identification of External Hazards

The specific external hazards that should be addressed by the study are identified in the generic letter supplement as:

Seismic Events,
Internal Fires,
High Winds and Tornados,
External Floods, and
Transportation and Nearby Facility Accidents.

However, in addition to addressing these hazards, as required by the generic letter, a review has been conducted to confirm that there are no plant-specific hazards excluded by the IPEEE guidance that might initiate severe accidents.

2.1.2 Methods of Examination

The response to the IPEEE for Fires has been met by performing a PRA. Portions of the EPRI Fire Induced Vulnerability Evaluation (FIVE) (EPRI, 1992) methodology have been adopted, particularly in the areas of location screening and fire frequency evaluation for the fire PRA. The seismic hazard has been addressed utilizing the Seismic Margins approach (EPRI, 1991). All other external events have been analyzed using the approach discussed in NUREG 1407. This approach includes screening and bounding calculations as a substitute for detailed PRA analysis.

Procedures for performing, documenting and reviewing each individual task were developed to assume a technically accurate and complete analysis. The methodology is described in more detail in later sections of the report.

2.1.3 Co-ordination with Other External Event Programs

Three programs are subsumed in the IPEEE: (1) the external event portion of USI A-45, (2) GI-131, and (3) the Eastern U.S. Seismicity issue. Any vulnerabilities associated with decay heat removal (USI-A-45) would be revealed and resolved during this process. The Eastern U.S. Seismicity Issue is resolved by the seismic part of the IPEEE. Since CP&L is exercising the seismic margins option, the resolution is achieved by an appropriate choice of review level earthquake. GI-131 deals with the seismically induced failure of the flux mapping transfer cart that would lead directly to the rupture of instrumentation tubes at the seal table. Since this is applicable to Westinghouse plants, it is applicable to HBRSEP. It is addressed in the IPEEE. USI A-46 has subsumed USI A-17, "Seismic Interactions in Nuclear Power Plants." Since HBRSEP is an A-46 plant, USI A-17 has been addressed in the A-46 submittal.

The Fire Risk Scoping Study (FRSS) Issues, NUREG/CR-5088, "Fire Risk Scoping Study," (NRC, 1989a), were examined through comparison to standardized checklist questions and through specifically tailored plant walkdowns according to the FIVE Methodology. The FRSS issues are discussed in Section 4.8. The issue of seismic-fire interactions has been addressed and is discussed in Section 3.

The revised "Design Probable Maximum Precipitation (PMP)" criteria were assessed within the Other External Events Task as requested in GL 89-22, Supplement 4, 0Resolution of Generic Issue No. 103, 'Design for Probable Maximum Precipitation,1 (NRC, 1989b). The conclusions are presented in Section 5.4.

2.1.4 Containment Performance

In accordance with GL 88-20, Supplement 4, Appendix 2, mechanisms that could lead to containment bypass, failure of containment to isolate, availability of and performance of containment systems under each external hazard have been evaluated to see if:

(i) they are different from those evaluated under the IPE, and

(ii) external events contribute significantly to the likelihood of containment failure.

The seismic margin study included consideration of containment integrity, containment isolation, and other containment functions, and is discussed in Section 3.1.5. Of the remaining external events, only fires are judged to have the capability of damaging individual components that could impact containment integrity, other than through support systems such as electric power or service water. However, in accordance with the FIVE Methodology these issues were only addressed for areas where the core damage frequency is greater than 1E-6/per year. In such cases the following is required: (1) An assessment of the potential for fires to damage equipment or prohibit manual operator action used to accomplish the containment function, and (2) identification of a minimum set of equipment and manual actions necessary to achieve the containment function considering those lost in the fire. This is discussed further in Section 4.7.

2.1.5 Accident Management - Vulnerability Screening

The evaluation of severe accident vulnerabilities was accomplished by reference to the Severe Accident Issue Closure Guidelines NUMARC 91-04, 0Severe Accident Issue Closure Guidelines,1 (NUMARC, 1991). Core damage sequences were grouped primarily on the basis of location (e.g., fire area/compartment) and also on the nature of the sequence, (non-LOCA, LOCA, fire induced containment bypass), and the group frequency compared to the closure guidelines which are provided in Tables 1 and 2 of the NUMARC document. These tables provide guidance as to the nature of appropriate action, ranging from effective hardware fixes (if the CDF of the group is greater than 1.0E-4 or > 50% of the total CDF), to no action required (if the group CDF is less than 1.0E-6). This is discussed further in Section 8.2.1.

2.1.6 Documentation of Examination Results

The documentation of the IPEEE study has three components. The first is this report which summarizes the results and findings of the IPEEE analyses given the plant status as of the end of refueling outage 15, and constitutes the Tier 1 documentation. The second is the PC based HBRSEP PSA plant model which is an updated IPE model, and was used to evaluate conditional core damage probabilities in the critical fire areas. The third is a set of analysis files which contain all the supporting assumptions, plant walkdown records, calculations, and reference information, all of which constitute the Tier 2 documentation.

This report follows the format specified in NUREG-1407, Appendix C as closely as possible. Information retained for audit corresponds to that specified in NUREG-1407, Section 8.2.

2.1.7 Examination Process

The seismic margins study was performed by a team comprised of CP&L engineers and engineers from EQE International. A peer review was conducted by two engineers from Vectra Technologies Inc.

The fire analysis and the analysis of external events was principally performed by NUS at their Gaithersburg office. In order to ensure that CP&L personnel are fully conversant with the IPEEE methods used for the analysis of these hazards and are in a position to fully integrate the knowledge gained from performing the work into operating procedures, training programs and appropriate hardware changes, a cognizant CP&L engineer was appointed to be the point of contact throughout the study. CP&L engineers performed an in-depth review of each of the separate analyses that make up the study.

In addition, CP&L engineers performed the quantification of the conditional core damage probabilities for the various plant damage states that were identified during the course of performing the fire analysis.

2.2 GENERAL METHODOLOGY

The following provides a brief overview of the approach used to evaluate seismic events, internal fires, as well as other and unique external events for HBRSEP. Further details of the methodology and application are provided in Sections 3, 4 and 5 of this report.

2.2.1 Seismic Analysis

The HBRSEP seismic IPEEE was performed using the EPRI seismic margins methodology recommended by NUREG*1407 (NRC, 1991b) for a full scope category plant. The essence of the approach is a demonstration that sufficient equipment needed to safely shut down the reactor is capable of surviving the review level earthquake (RLE). The following discussion provides a summary of the general approach and its philosophy. The application to HBRSEP is described in Section 3.

A seismic margin assessment (SMA) consists of the following essential steps:

1. Selection of the Review Level Earthquake (RLE),
2. Selection of Assessment Team,
3. Preparatory Work Prior to Walkdowns,
4. Systems and Equipment Selection ("Success Paths") Walkdown,
5. Seismic Capability Walkdown,
6. Subsequent Walkdowns (as-needed),
7. SMA Work, and
8. Documentation.

Step 1 involves the specification of the earthquake for which the seismic margin assessment is to be conducted using the guidance provided in EPRI NP-6041 (EPRI, 1991). The Review Level Earthquake as prescribed in NUREG 1407 was selected.

The Seismic Review Team (SRT) for Step 2 consisted of senior seismic capability engineers who were ultimately responsible for the seismic capability walkdowns and screening out components from further seismic margin assessment, and also for defining any required seismic margin assessment scope of work for those components not screened out. The SRT was assisted by other seismic capability engineers in gathering data and conducting SMA calculations.

The qualifications of the members of the SRT included:

1. Knowledge of the failure modes and performance of structures, tanks, piping, process and control equipment, active electrical components, etc., during strong earthquakes,

2. Knowledge of nuclear design standards, seismic design practices and equipment qualification practices for nuclear power plants,

3. Ability to perform fragility/margin-type capability evaluations including structural/mechanical analyses of essential elements of nuclear power plants,

4. Some general understanding of Seismic PRA conclusions and systems analysis, and

5. Some general knowledge of the plant systems functions.

The Step 3 preparatory work prior to the walkdowns consisted of gathering and reviewing information about the plant design and operation. During this step, the system engineers defined the candidate shutdown paths and the associated frontline and support systems and components. The seismic capability engineers gathered information on the seismic design and equipment qualification. Information on locations of relays central to the selected success paths was obtained. Summaries of equipment design features and locations of equipment to be walked down were noted on walkdown data sheets for use by the SRT during the walkdown.

The primary objective of the Step 4 walkdown was to assess the relative seismic ruggedness of the major equipment in the candidate success paths and select a preferred and one alternate success path. If weak links were observed that are not economically feasible to fix, then success paths that rely on the weak link components were avoided. Some support systems, such as emergency AC power, are required for all success paths and were assessed during the Step 4 walkdown or postponed until the Step 5 walkdown. During the Step 5 walkdown, all fluid, electrical power and instrumentation systems that are required for the selected success paths were walked down to identify any potential weak links, including the potential for seismic spatial systems interactions (SI). The Step 6 walkdown may not be necessary if all screening decisions and necessary data gathering is completed during Steps 4 and 5. It is optional if selected components require a revisit to gather further information.

In the case of active electrical and control equipment, it may not be possible or cost effective to demonstrate functionability on the basis of achieved test level or by use of Generic Equipment Response Spectra (GERS) (SQUG, 1992). The system engineers are required to evaluate the electrical circuits and operations procedures to assess the consequences and recovery action for relay chatter, breaker trip, etc.

Guidelines are provided (EPRI, 1991) for the documentation of results. The documentation may be designed to suit the seismic assessment team and the guidelines presented are considered to be suggestions for a content that would completely summarize the study but not require great detail in reporting for each item evaluated.

2.2.2 Internal Plant Fires

The object of this task is to estimate the contribution of accident sequences induced by in-plant fires to overall core damage frequency. The analysis considers the likelihood of fire occurrence in each plant area and its subsequent impact on plant systems. Equipment damage resulting from the thermal effects of fire (conductive, radiative and convective) are considered as well as the degradation of operation reliability. Potential vulnerabilities raised in the Fire Risk Scoping Study related to seismic/fire interactions, effects of suppressants on safety equipment and control system interactions are addressed through specifically tailored walkdowns, as defined in the EPRI FIVE methodology (EPRI, 1992).

The models were developed in a systematic manner which enabled the specific strengths and weaknesses of plant defenses against fire to be clearly identified.

The fire evaluation was performed on the basis of fire areas, which are plant locations completely enclosed by rated fire barriers. The fire area boundaries were assumed to be effective in preventing a fire from spreading from the originating area to another area based on the implementation of a satisfactory fire barrier surveillance and maintenance program. The fire area boundaries recognized in this study are identical to those identified in the plant's 10 CFR 50 Appendix R report. In some cases these fire areas were further subdivided into compartments. For analysis purposes, for the more significant compartments, fire damage states within those compartments were defined that identified subsets of the equipment within the compartments as being damaged due to the fire.

The analysis was conducted in three main stages as follows:

Stage 1 was a systematic qualitative and quantitative screening analysis of all plant fire areas/zones, following the methodology described in FIVE, Phase 1 and Phase 2, steps 1 and 2. The screening analysis was based largely on information already available in the plant's 10 CFR 50 Appendix R report and the IPE study. This resulted in the identification of fire areas and compartments in accordance with the FIVE methodology. At this stage all equipment and cable in an area/compartment was assumed to be damaged by a fire in that room. The damage was assessed qualitatively to determine if the effects are significant; that is, does the fire cause an initiating event or lead to loss of accident mitigating equipment. Areas/compartments not screened out qualitatively were then subject to a determination of their associated fire frequency (F_1) and conditional core damage frequency (P_2), given loss of all functions which may be impacted by the fire. If the resulting fire induced core damage frequency (F_1 x P_2) was less than 1.0E-6 per year the area /compartment was screened out.

Stage 2 was a detailed evaluation of the fire areas/compartments which did not previously screen out, using fire PRA techniques as well as methods and data provided in the FIVE technical report, Phase 2, steps 3 - 5. The principal refinement in this phase of the analysis was that (i) the resulting impact of intermediate fire growth stages within each fire area/compartment was assessed (rather than assuming the contents are immediately damaged), and (ii) the impact of fire on containment systems was evaluated. The initial exposure fires resulting from each potential ignition source were evaluated individually. Through the use of COMPBRN IIIe to predict burning rates, heat fluxes, secondary combustible ignition, hot gas layer temperature and target temperature, the fire duration required to cause initiation of fire detection and damage to specific targets were evaluated. The use of COMPBRN demonstrated that either:

(i) damage to equipment/cable in the immediate vicinity of the source occurred quickly such that initiation of fire detectors or auto suppression would not be effective, or

(ii) no damage was sustained by targets located a few feet from the fire source prior to the fire self extinguishing.

In theory, the probability of achieving a particular fire damage state (i) can be represented as:

$$Q(FDS_i) = \int f_{tg}(t) * (1 - F_{ts}(t))dt$$

Where $F_{ts}(t)$ is the probability of successful suppression before time (t), and $f_{tg}(t)$ is the probability that the fire duration required for achieving fire damage stage (i) is between t and (t+dt).

A simplified, discretized form of this equation was used in the HBRSEP IPEEE.

The second stage required a substantial new data collection effort including identifying and locating sub-components and cables associated with critical PRA components and the compilation of an electronic data base to efficiently record and search this data.

The third stage of the fire evaluation was an evaluation of the Sandia Fire Risk Scoping Study Issues using the tailored walkdown approach provided in FIVE, Section 7. This evaluation is presented in Section 4.8.

2.2.3 Other and Unique External Events

The "other external events" analysis was performed based on the progressive screening approach described in Section 5 of NUREG-1407.

NUREG-1407 describes the screening process used to arrive at the list of potentially significant other external events, namely: high winds, external floods, transportation and nearby facility accidents. Since this process used generic plant information, NUREG-1407 concludes that the first step in the IPE of other external events is to determine that there are no critical features of the plant or its surroundings that might invalidate the generic conclusions regarding potentially significant hazards (i.e., that there are no "unique" external events which should be addressed). For HBRSEP this was achieved by reviewing the information available in the Updated Final Safety Analysis Report (UFSAR) (CP&L, UFSAR) by collecting supplemental information that might have changed since the last revision of the Final Safety Analysis Report (FSAR), and by performing a confirmatory plant walkdown.

For the specific other events called for in NUREG-1407, the analysis was conducted using the procedure recommended in NUREG-1407 as follows:

1) The specific hazard data were reviewed,

2) Significant changes since the operating license was issued were identified, and

3) An evaluation was made to determine if the plant and facilities meet the 1975 Standard Review Plan (SRP) (NRC, 1975) criteria. If the review revealed that the 1975 design criteria has been met with respect to a particular hazard, it was judged that the contribution from the hazard is less than 1E-6/yr and the IPEEE screening criteria is met. For hazards where the SRP is not met, one or both of the following optional steps was taken:

   (a) A hazard frequency determination, and

   (b) A bounding analysis to show either the hazard could not cause core damage or the resulting core damage frequency is below 1E-6/yr.

In the case of HBRSEP all the Other External Events with the exception of high winds were screened at step 3.

2.3 INFORMATION ASSEMBLY

The first step in the performance of the IPEEE tasks was the assembly and review of plant specific and generic data which would form the basis for the study. This consisted primarily of the UFSAR (CP&L, UFSAR), seismic design information, the 10 CFR 50 Appendix R report, fire area layout drawings, Appendix R Cable Block Diagrams, Off-Normal Procedures, the Robinson IPE study, the EPRI FIVE methodology and supporting documents. The Other External Events analysis relied on the UFSAR and the 1975 Standard Review Plan (NRC, 1975) for much of the needed information. These sources were supplemented by specific data collection when considered necessary. A precise description of how the information in each of these documents was used is provided in Sections 4 and 5.

The analysts responsible for the analysis were already familiar with the techniques and methods of external events analysis at the onset of the study, having already completed other IPEEEs, and external event PRAs. The NUS Fire Protection Engineers were very familiar with the plant through other projects performed for CP&L. The CP&L engineer assigned as the main point of contact was very familiar with the plant systems through his involvement with the IPE. Walkdowns were performed at various phases in the study to confirm the validity of information already used as well as to collect detailed data regarding cable raceway configurations and ignition source locations, etc. Walkdowns were performed according to a pre-determined plan and the results formally recorded. In addition, plant operations and training personnel were interviewed to discuss the plant procedures and resolve many "what if" type questions.

2.4 REFERENCES

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SECTION 3

SEISMIC MARGIN ANALYSIS

3.0 INTRODUCTION

The H. B. Robinson Steam Electric Plant (HBRSEP), Unit 2 seismic Individual Plant Examination for External Events (IPEEE) was performed using the EPRI seismic margins methodology recommended by NUREG-1407 (NRC, 1991), Methodology for Assessment of Nuclear Power Plant Seismic Margin, for a full scope plant. The essence of the approach is a demonstration that the minimal set of equipment that is needed to safely shut down the reactor is capable of surviving the review level earthquake (RLE). This minimal set of equipment is called the safe shutdown equipment list (SSEL). For those items, and the structures that either contain them, or whose failure could cause their failure, an assessment is made of their capability. Those items capable of withstanding the RLE can be screened from further consideration. Any items that cannot be screened are identified and must be addressed in some manner.

Civil structures, equipment and subsystems were reviewed using the methodology provided in EPRI NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin", (EPRI, 1991) for a full scope plant. The guidelines were supplemented by Appendix A of EPRI NP-6041, which provides the basis for the seismic capacity screening guidelines. A walkdown of the items on the SSEL was performed to confirm the as-designed characteristics, and to address the seismic-systems interactions issues raised by the USI A-17, GI-131, and the Sandia Fire Scoping Study Issues (NRC, 1989).

The detailed Seismic Margin Study is attached as Appendix A to this report. This section is a summary of the analysis and its conclusions.

3.1 APPLICATION OF SEISMIC MARGIN METHOD TO THE ROBINSON NUCLEAR PLANT

3.1.1 The Review Level Earthquake for the Robinson Plant

3.1.1.1 The Operating Basis and Safe Shutdown Earthquakes

Based on historical seismicity, the maximum potential earthquake which might affect the site would be a recurrence of the Charleston, South Carolina earthquake of 1886 which was probably felt at the site as an intensity VI on the Modified Mercalli Index. Only one earthquake strong enough to crack plaster has occurred since 1850 within seventy-five miles of the plant site. This earthquake occurred in McBee, South Carolina on October 26, 1959 and the estimated intensity at the site was about Modified Mercalli Index V. The descriptions of the earthquake indicate a very mild motion with accelerations not exceeding a few percent of gravity.

A more detailed discussion of geology and seismology for the site is included in the HBRSEP Updated Final Safety Analysis Report (UFSAR), Section 2.5 (CP&L, UFSAR).

On the basis of the seismology of the site area, the operating basis earthquake (OBE) peak accelerations are based on a Richter scale magnitude 4.5 earthquake with an epicentral distance of less than ten miles from the site. The probable ground acceleration from this earthquake would be .07 to .09g. However, the maximum horizontal ground acceleration at foundation level was conservatively selected to be .10g.

The safe shutdown earthquake (SSE) is designated as .20g based on a magnitude 7.0 earthquake comparable to the 1886 Charleston shock occurring 35 miles from the site.

All safety related structures and systems are designed to assure safe plant shutdown for the maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously.

3.1.1.2 Ground Response Spectra

The design response spectra used for all Seismic Category I structures, systems, and components, except dams and dikes, were developed in accordance with recommendations from George Housner based on similarities between the depth and the type of overburden material at the Robinson site and certain areas in California.

The HBRSEP in-structure ground response spectrum for 1/2% equipment damping and a normalized earthquake of 0.2g was provided by the Westinghouse Electric Corporation (Westinghouse, 1970). A copy of this is included as an attachment to Corporate Consulting and Development Company, Limited (CCL) Report A-764-88, "Generation of Floor Response Spectra for HBRSEP". The methodology used to estimate the ground response acceleration at various values of equipment damping is detailed in the CCL report. A synthetic time history which simulates a 0.2g earthquake was used to duplicate the Westinghouse spectra at 1/2% damping. This time history was used as a basis to ratio the Westinghouse 1/2% spectra for other damping values. Spectra for 1%, 2%, and 5% damping as addressed in the UFSAR were generated.

3.1.1.3 Review Level Earthquake

For the seismic IPEEE program, the review level earthquake is prescribed by NUREG 1407 as the NUREG/CR-0098 median response spectrum anchored to 0.30g.

3.1.2 Identification of Components and Structures for Review

A preliminary walkthrough was performed by CP&L and EQE personnel to search for potential low seismic capacity components. The Seismic Qualification Utility Group (SQUG) 0Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 2 (SQUG, GIP) was utilized in choosing the items and identifying boundary conditions and assumptions.

The seismic Safe Shutdown Equipment List (SSEL) identifies the equipment that provides the critical plant functions required to safely shut down the plant and maintain it in hot or cold shutdown for seventy-two hours. The relevant plant functions are:

- Reactivity control,
- Reactor coolant system inventory control,
- Reactor coolant system pressure control, and
- Decay heat removal.

Equipment in the following systems were identified as being essential for ensuring critical plant functions:

- Safety and some Non-Safety Related AC Power,
- Safety Related DC Power,
- HVAC for Emergency Diesel Generator Rooms and Control Room,
- Reactor Coolant System,
- Safety Injection System,
- Residual Heat Removal System,
- Auxiliary Feedwater and Condensate,
- Main Steam System,
- Instrument and Station Air System,
- Nitrogen Supply System,
- Service and Cooling Water System,
- Component Cooling Water System,
- Fuel Oil System,
- Emergency Diesel Generator System, and
- Chemical and Volume Control System.

Success Path Logic Diagrams (SPLD) were constructed based on an understanding of available plant equipment function as well as the plant's normal and emergency operating procedures. The SPLDs were reviewed and agreed upon by Robinson operations personnel. They were used as a basis for the identification of the equipment to be included on the SSEL. Equipment selected for inclusion on the SSEL was evaluated in a manner similar to that described in the SQUG Generic Implementation Procedure (GIP) (SQUG, GIP). Guidance from EPRI NP-6041 (EPRI, 1991) was also used in preparing the format for the list of components. The SSEL used for the

IPEEE walkdown is presented in Appendix C of the full Seismic IPEEE report for HBRSEP (Appendix A of this report).

The assessment of the equipment necessary to maintain the identified functions was made under a set of boundary conditions. Offsite power was assumed to be lost. The success paths must be capable of maintaining the plant in either hot or cold shutdown for a period of 72 hours. Two success paths were considered, one for a transient in which the RCS remains intact, and one success path must be considered for a 1" LOCA event. Non-seismic failures were included in the assessment, and the potential for relay chatter was addressed.

In addition to the components of the systems discussed above, the structures housing the components included in the above systems were reviewed. Seismic Category I structures are cast-in-place reinforced concrete structures. The floors are supported on beams and girders which are in turn supported on interior columns and/or exterior walls. Where interior shear walls are installed, the beams and girders are supported on the shear walls. All interior shielding walls and partitions, other than shear walls, are either reinforced concrete or concrete block, and are not load bearing. The buildings are supported on pipe piles filled with concrete which are founded on suitable soil. For design purposes, the seismic analyses of the Seismic Category I structures were performed by using the normal mode time-history technique. The structures, considered as seismic systems and analyzed in this manner, are the Reactor Containment Building, the Reactor Auxiliary Building, the Class I Bay Turbine Generator Building, and the Service Water Intake Structure.

3.1.3 Seismic Walkdowns

3.1.3.1 Pre-screening

The seismic review team utilized plant design drawings, analyses and test reports to use in conjunction with the screening criteria. A considerable amount of information was reviewed and summarized in the Screening and Evaluation Worksheets (SEWS) during a pre-screening. Pre-screening was enhanced by the use of the software program EHOST. EHOST is a data base program which has been adapted specifically for use in performing Unresolved Safety Issue (USI) A-46 and IPEEE evaluations. The program is set up so that the data is incorporated onto SEWS forms which are consistent with those recommended in EPRI NP-6041. In this manner the walkdown teams, using portable computers with the companion program EWALK were then able to work more efficiently by having access to SEWS that had already been partially completed.

The objective of pre-screening was to ensure efficiency in the walkdowns and evaluations with a goal of completing the maximum amount of data entry in advance of the walkdown. This was accomplished by incorporating existing data onto the seismic A-46/Seismic IPEEE SEWS documentation forms prior to the walkdowns. Data that was reviewed consisted of the Updated Final Safety Analysis Report (UFSAR), design criteria, stress reports, equipment qualification reports (testing and analysis), structures and equipment support drawings, equipment location drawings, anchorage calculations, and records from other related programs previously performed at HBRSEP. The information is not intended as the sole basis for screening, but assists the SRT in their review. An initial walkdown was performed by CP&L and EQE personnel as part of the pre-screening task to review the SSEL and to group items according to the specific host equipment for evaluation, considering the "Rule of the Box" (SQUG, GIP).

Pre-screening was performed with three purposes in mind:

- To identify critical failure modes to be specifically reviewed on the walkdown,

- Assemble qualification and installation data for use as a basis for screening in the margins review, and

- To provide data to be utilized in calculations to establish the acceleration corresponding to the High Confidence of a Low Probability of Failure (HCLPF).

3.1.3.2 Performance of Walkdowns

The HBRSEP seismic margin walkdown was completed in conjunction with the A-46 walkdowns during two phases. The first phase involved the walkdown of all components in the Reactor Containment Building and all electrical components in the Reactor Auxiliary Building that could not safely be inspected during plant operation. This walkdown occurred during the Fall 1993 Refueling Outage 15 (RFO-15). September 14-29, 1993, were the actual dates for these walkdowns. The second phase of the walkdowns captured the balance of plant SSEL components and occurred during various periods during 1994 (March 7-11, April 4-8, May 5-6, and November 3,) and early 1995 (January 17, January 23, and February 14).

The procedure for performing walkdowns is described in the Robinson Project Plan. The walkdowns concentrated on the strength and load path of the equipment to the structure as well as function and integrity. The review of equipment anchorage was a prime objective for the walkdown teams. The walkdown addressed the physical attributes of the anchorage installation. Anchorage capacities were addressed in the pre-screening as much as possible where drawing details and calculations could be found. The walkdown teams verified that the anchorage was generally in accordance with the design configuration. If applicable details could not be found, then the walkdown teams prepared sketches and detailed notes to evaluate the anchorage of the component after the walkdowns. Component anchorage was screened where possible based on the high capacity anchorage and the SRT walkdown information.

Interaction reviews were performed to identify falling, impact, spray and flood issues that could affect success path items. Interaction issues were addressed on the Seismic Evaluation Worksheet (SEWS) forms. No spray or flood issues were noted during the SRT walkdown.

Suspended systems, such as conduit, cable trays and ductwork were evaluated on a sampling basis in the plant. A general survey was performed to obtain an overview of the suspended system construction throughout the plant. This included a review of the variety of system layouts, support configurations, and construction details. The inspection also included known concerns for suspended systems, such as taut cables, sharp edges, or overloading of cable trays and supports, and potential anchor point displacements. The ceiling above the control room was also reviewed to verify if the light fixtures and ceiling grid were adequately supported, and to evaluate the potential for ceiling panels to fall.

Containment piping and electrical penetrations were reviewed on an area basis to identify anomalies that might affect containment performance. Concerns such as falling and differential building displacement were considered. Also reviewed were displacement concerns between the containment shell and internal structure. Containment isolation valves were also reviewed on a sample basis in accordance with the SMA requirements, and on a walk-by basis based on the caveats listed on the valve SEWS.

3.1.3.3 Results of Walkdown

Seismic margin walkdown results are summarized for structures and equipment and subsystems in Table 5-2 and Table 5-3 of Appendix A, respectively.

Table 5-2 lists civil structures, following the format of EPRI NP-6041, Table 2-3, along with screening results for the HBRSEP. All HBRSEP civil structures were screened from further review based on the EPRI NP-6041, Table 2-3, screening criteria and Section 3.8 of the HBRSEP UFSAR (CP&L, UFSAR).

Table 5-3 of Appendix A lists equipment and subsystems following the format of EPRI NP-6041, Table 2-4, along with screening results for the HBRSEP. At the conclusion of plant walkdowns, SRT members, including senior level participants from CP&L and EQE, reviewed the completed SEWS documentation forms and assigned the components into the following resolution categories:

- Those components screened out by meeting all of the GIP caveats,
- Those components identified as outliers requiring only minor repairs and/or work tickets (this group included components with housekeeping issues) to be screened out,
- Those components identified as outliers requiring modifications to be screened out, and
- Those components that were initially screened by review of the caveats or those screened by other methods that were also good candidates for HCLPF evaluation.

An outlier is defined as a condition where the component does not meet the intent of one or more of the GIP caveats. Outliers can be resolved by further review of applicable records and documentation, by preparation of a calculation to evaluate the as-built condition, or by repair and/or modification to place the component into a condition that does meet the intent of the GIP caveat. Outliers do not necessarily constitute an operability issue, although they can. No operability issues resulted from the walkdowns by the Seismic Review Teams.

3.1.3.4 Resolution of Issues Arising from the Walkdown

Following the initial walkdown, SRT members re-walked items not screened out and revisited existing design and qualification documentation in an attempt to screen out more. SRT members that performed the initial walkdowns presented and discussed issues with remaining SRT members. Further categorization was refined by group consensus. The following results were determined based on the above categorization.

- Thirty-three items were identified as outliers requiring minor maintenance that could be repaired by a work ticket. Items that can be repaired by work tickets are typically those items whose condition have only slightly degraded from the original design intent and can be fixed by using existing plant drawings, procedures, or guidelines. The repair can be implemented by maintenance or construction without any engineering input or review. It usually involves a replacement of like hardware, torquing of bolts, etc. These items are listed in Tables 3-1 and 3-2 in this section.

- Seventeen items were identified as outliers that could be resolved by calculations.

- Twenty-two items were identified as outliers and the Seismic Review Team determined that additional calculations would potentially not resolve the outlier issues. They concluded that the twenty-two items would best be resolved by the implementation of physical plant modifications, as summarized in Table 3-3. Modifications provide the vehicle to change components using engineering review and input. These components are listed in Table 5-3 in the SMA main report (Appendix A).

- Twenty items were evaluated for the High Capacity for Low Probability of Failure methodology. These items are identified in Table 5-3 of the SMA main report.

- Sixteen issues involving electrical raceway installations were identified as requiring work ticket/maintenance or modifications attention in order to restore the reported issue to an acceptable condition.

3.1.4 Relay Evaluation

HBRSEP is identified as a full scope plant for the .3g earthquake by NUREG-1407. The NUREG-1407 document requests that full scope plants such as HBRSEP which are also included as a USI A-46 Plant should follow the USI A-46 procedures for the relay review. NUREG-1407 also states that the plant systems should be reviewed within the scope of the IPEEE, including those that are within the scope of USI A-46, using appropriate margins from EPRI NP-6041 or USI A-46 procedures for the RLE. The USI A-46 criteria for relay functionality review are contained in Generic Letter 87-02 (NRC, 1987). Generic Letter 87-02 endorses the review procedure established in the Generic Implementation Procedure (GIP) Revision 2.

The GIP states that the purpose of the relay functionality review is to determine if the plant safe shutdown systems could be adversely affected by relay malfunction in the event of an SSE and to evaluate the seismic adequacy of those relays for which malfunction is unacceptable.

The GIP methodology for evaluation of the seismic functionality of relays is based on a two part screening process. The first part identifies a minimum set of relays whose function is essential for safe shutdown. The second part of the relay evaluation process uses relay GERS and test data to assess the seismic adequacy of the essential relay types.

The identification of a minimum set of relays whose function is essential to the safe shutdown of the plant was prepared by engineers in the CP&L Probabilistic Risk Assessment (PRA) Group and the Robinson Engineering Support Section (RESS) Instrumentation and Control (I&C) Group. There were 789 relays and switches that were identified whose function were required for safe shutdown. Relays required to satisfy A-46 functions and relays required to satisfy IPEEE functions were incorporated to form the composite listing of 789 relays.

These 789 relays and switches were organized into an HBRSEP Unit 2 Essential Relay List. The relays on this list were investigated during the same walkdowns when the safe shutdown equipment was evaluated. This investigation and evaluation were performed by CP&L and EQE engineers who had successfully completed the SQUG Walkdown Screening and Seismic Evaluation Training Course.

The seismic capability engineers (SCEs) verified that the manufacturer, the make, and the model were accurate according to the information provided on the essential relay list for a representative majority of the relays. The SCEs also observed and evaluated the mounting of the relays on or within electrical panels. It should be noted that the seismic adequacy of the panel structures and the anchorage was addressed by the separate evaluation of the panel as an SSEL equipment component.

After the completion of the walkdown and physical determination phase of the relay evaluation, the seismic adequacy of the essential relays was then assessed by EQE using GERS and other test data. The HBRSEP Unit No. 2 Relay Evaluation Report for USI A-46 provides a summary of the results of the relay seismic capacity versus seismic demand study. This report addresses all 789 relays. Details for this assessment are provided in EQE calculation 52212-C-020 entitled "H.B. Robinson Unit 2: Relay Evaluation."

Although there was only one composite list for relays, they were evaluated against the A-46 screening criteria and also against the IPEEE criteria. 745 passed the USI A-46 capacity screening levels 0, 1, and 2. Level 0 screening is associated with switchgear only. Level 1 is associated with high capacity relays, the use of response spectra comparison, the location of relays within the plant, and the identification of no known low-ruggedness relays. Level 2 capacity screening is based on the use of in-cabinet amplification factors, appropriate factors of safety, and the use of generic equipment ruggedness spectra (GERS) or relay-specific seismic test data. Forty-four relays did not meet the screening criteria and were submitted for further evaluation in the form of relay system consequence reviews.

707 of the 789 relays passed the IPEEE capacity screening levels 0, 2, and 3. Eighty-two relays and switches did not meet the IPEEE screening criteria and were submitted at the same time as the USI A-46 relays for further evaluation in the form of relay system consequence review. These eighty-two IPEEE relays submitted for consequence review included the forty-four A-46 relays identified in the previous paragraph.

Eleven of the relays submitted for consequence review were screened out because they were determined not to be associated with equipment on the SSEL. Twenty-six of the relays were screened out because they were determined to provide annunciation input only. Thirty-one of the relays were screened out as being acceptable for relay chatter. This relay chatter would not result in the failure of the associated SSEL equipment or its ability to perform its required function. Finally, fourteen relays were screened out based on relay chatter associated with appropriate operator action. Plant procedures are already in place which stipulate operator action for these relays for certain scenarios. A memorandum was addressed to the plant operations department documenting these relays and the results of the consequence review.

In summary, all 789 relays on the Robinson Essential Relay List have been accepted by either capacity screening or system consequence screening. There were no low-ruggedness relays encountered during the relay evaluation. It should be noted that the relay evaluation was based on adequate and direct mounting of the relays to the electrical panel. Any missing mounting hardware or loose relay connections were addressed on the panel SEWS forms and work tickets and/or maintenance requests have been identified or already issued to correct these issues. It should also be noted that the adequacy of the panel structure and anchorage of the panels or cabinets is addressed by the separate evaluation of the panels as an SSEL equipment component. Relay panels and cabinets that were not anchored properly or had other unacceptable criteria were addressed on the SEWS forms and are being corrected via the applicable plant requirements.

3.1.5 Containment Integrity

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

The guidance provided 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 (e.g., spatial interactions, unique penetration configurations) 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, which is addressed in Section 9 of Appendix A to this report.

A review of seismic capacities for containments of similar design to Robinson indicates that the containment structure is expected to have a seismic capacity far above the review level earthquake (NRC, 1987). In addition to the containment structure, NUREG-1407 suggests that certain considerations could require some additional study. Hatches that employ inflated seals is one potential area for concern. The HBRSEP design does not employ this type of seal. Another concern is the post-operation of penetration cooling that is present in some designs. HBRSEP, however, does not employ this design feature. Finally, air-operated valves used for isolation are also listed as a possible concern. The containment isolation valves at HBRSEP are fail safe, and close on loss of air supply. Thus, failures in containment isolation would not be expected due to containment system failures.

Two normally closed motor operated values, MOV 750 and 751, had HCLPF less than the review level earthquake value. These are discussed in Section 3.1.8.

The containment walkdown consisted of looking at/evaluating unusual conditions/configurations (e.g., spatial interactions, unique penetrations, piping hard spots, items/components bridging the seismic gap between the containment liner and interior structure, and etc.). The containment walkdown was performed by the SRT (see section 5 of Appendix A).

No unusual conditions/configurations were noted.

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 walkdown performed of the containment did not identify any vulnerabilities. Therefore, the HCLPF for the containment is greater than 0.3g, based on the results of this analysis.

3.1.6 Seismic Induced Fire and Flood Evaluation

Seismic/fire interactions, effects of suppressants on safety equipment, and control system interaction are addressed in the IPEEE. As discussed in Section 8 of Appendix A, failures of the fire protection system that lead to a release of water will not affect the capability 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 sources was performed to identify any vulnerabilities. As discussed in Section 8 of Appendix A, no vulnerabilities were found.

3.1.7 Soils Evaluation

The soils evaluation approach is based upon guidelines contained in EPRI NP-6041 (1991) for assessing liquefaction susceptibility utilizing standard penetration test data and previous geotechnical reports generated for the Robinson site. An additional evaluation was made for the Robinson Dam and is based on simplified approaches developed by Newmark (1965).

Blowcount information from the available borehole data logs for the Robinson site were converted to equivalent Standard Penetrometer Test (SPT) values for the conditions of 1 ton per square foot overburden pressure and corrected to account for the effects of fines content and earthquake magnitude. These modified blowcounts were taken to be equivalent to SPT data collected with hammer having an energy efficiency of 60%.

Potentially liquefiable soil deposits for each boring log were identified by comparing the corrected blowcount with threshold values corresponding to the onset of liquefaction. This comparison indicates that nearly all of the data points fall in the non-liquefiable region. The few points that do fall below this threshold are considered statistically insignificant. Accordingly, the acceleration threshold for liquefaction at the Robinson site is considered to be above 0.3g.

An additional evaluation was performed on the Robinson Dam. The Robinson Dam is maintained to provide the cooling water supply for the plant. The dam was originally designed to provide the cooling water supply for the fossil plant located adjacent to the nuclear plant. The dam was not specifically designed for the nuclear plant. Therefore, provisions are in place to provide emergency cooling for the nuclear plant if the dam were to fail. However, dynamic analysis of the dam indicates a factor of safety of 1.02 against slope failure for an earthquake of 0.2 g. To assess the impact of a 0.3 g earthquake, reference was made to the relationship developed for unsymmetrical block sliding (Newmark, 1965). The details for this analysis are included in the calculation 52212-C-067. Considering the excavation plan for the dam and the results of the site evaluation of liquefaction susceptibility, liquefaction is judged not to be a likely concern for the Robinson Dam at 0.3g and the dam is considered acceptable for a 0.3g earthquake.

3.1.8 Results of HCLPF Analysis

With one exception, the HCLPF values for the twenty items selected for further examination were greater than the review level earthquake (RLE) of .3g. However, two MOVs were identified as having low ruggedness due to the presence of ductile iron in their yokes. These MOVs are RHR-750 and RHR-751 which are the RHR shutdown cooling pressure boundary isolation valves at the RCS loop 2 hot leg. Seismic evaluation determined that the HCLPF (High Confidence Probability of Low Failure) value is 0.28g. Statistically, this means there is a 95% confidence level that the component will fail only 5% of the time for this peak ground acceleration. According to EPRI NP-6395-D , Figure 3-241 (EPRI, 1989), the frequency of exceeding a 0.28g earthquake at the Robinson site is 2.0E-5 per year.

These isolation valves provide two functions that may be compromised by a seismic event:

1. The first function is to provide a suction source for the RHR system during shutdown cooling. Since this is the only shutdown cooling suction path, it is postulated that failure of

the yoke may prevent the MOVs from opening and cause the plant difficulty in establishing normal cold shutdown. Only one valve must fail to cause this difficulty.

The frequency of one MOV failing closed due to a seismic event is calculated as follows:

$$F_{(SD\ Cooling)} = F_{(0.28\ g)} * P_{(MOV\ Yoke)}$$

$$F_{(SD\ Cooling)} = (2.0E\text{-}05\ /year) * (0.05) = 1.0E\text{-}06\ /year$$

Shutdown cooling is only credited as an optional path in the event of a 1 inch LOCA. Given this path fails, containment sump recirculation is available. The non-LOCA paths credit either long term secondary side heat removal (greater that 72 hours) or containment sump recirculation following feed and bleed cooling.

Since the shutdown cooling line is not required to maintain the plant in a safe stable state for the first 72 hours following a seismic event, and the failure likelihood is sufficiently low, loss of the shutdown cooling function of these MOVs caused by a seismic event is not considered to be a vulnerability.

2. The second function is to provide a high to low pressure system boundary during normal power operations. Loss of this pressure boundary would result in an interfacing systems LOCA outside of containment, and possibly failure of both low head and high head safety injection. Failure of the yokes could conceivably cause both MOVs to open. (Both MOVs must fail for the event to occur.) Using the standard seismic PRA assumption that seismic failures of like components in the same general location are correlated, the frequency of yoke failure on both MOVs was evaluated to be less than 1.0E-6/year with a 95% confidence. Failure of the yokes could be regarded as a precursor to the ISLOCA event; however, it would be conservative to equate this to the frequency of the occurrence of the ISLOCA. Nevertheless, this scenario will be evaluated further for possible inclusion in the Severe Accident Management Guidelines.

3.2 COORDINATION WITH OTHER PROGRAMS

Four programs are subsumed in the IPEEE: (1) the external event portion of USI A-45, (2) GI-131, (3) the Eastern U.S. Seismicity issue, and (4) A-46 - Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Nuclear Plants. The decay heat removal issue (USI-A-45) is addressed by the fact that the SSEL contains the equipment necessary to maintain decay heat removal for a period of 72 hours. Since CP&L is exercising the seismic margins option, the resolution of the Eastern U.S. Seismicity Issue is achieved by an appropriate choice of review level earthquake. GI-131 deals with the seismically induced failure of the flux mapping transfer cart that would lead directly to the rupture of instrumentation tubes at the seal table. It is addressed in the IPEEE in Section 5.9 of Appendix A. USI A-46 has subsumed USI A-17, "Seismic Interactions in Nuclear Power Plants". Robinson is an A-46 plant and USI A-17 has been addressed through the seismic walkdown that were performed to meet the requirements of the A-46. Issues raised in Information Notice 94-12 are discussed in Section 8 of Appendix A.

3.3 CONCLUSIONS

The results and conclusions of the HBRSEP IPEEE seismic project are discussed in more detail in Appendix A. However, the principal conclusion is that there are no seismic vulnerability concerns at HBRSEP. The scenario involving failure of the cast iron yokes on the RHR valves 750 and 751 will be evaluated further for possible inclusion in the Severe Accident Management Guidelines.

3.4 REFERENCES

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

(Westinghouse, 1970), Docket No. 50-261, Letter dated June 5, 1970 to Peter Morris from J.A. Jones.

Table 3-1
Containment Walkdown Work Ticket Items

Equipment No.	Building	Elev.	Work Ticket	Description	Status
CVC-303A	RCB	251.1	WR 93APRE1	Install longer flex conduit between rigid conduit and limit switch to ensure adequate slack and flexibility and protect interior cables.	Complete
CVC-310A	RCB	226.0	WR 93APQZ1	Install longer flex conduit between rigid conduit and limit switch to ensure adequate slack and flexibility.	Further review revealed that valve fails in desired position. Therefore, work request was withdrawn and SEWS revised appropriately. Reference Memorandum NED-C-0198.
CVC-310B	RCB	226.0	WR 93PARB1	Install longer flex conduit between rigid conduit and limit switch to ensure adequate slack and flexibility.	Further review revealed that valve fails in desired position. Therefore, work request was withdrawn and SEWS revised appropriately. Reference Memorandum NED-C-0198.
Relay on Auxiliary Relay Panel CC	RAB	242.5	WR ACSM1	Replace missing screw to provide proper support for Agastat relay 7022.	Complete
Relay on Auxiliary Relay Panel MC	RAB	242.5	WR 94ACSQ1	Tighten screws to design values to ensure proper support of relay.	On hold
Relay on Auxiliary Relay Panel FC	RAB	242.5	WR 94ACSS1	Replace missing screw to provide proper support for Agastat relay 7012.	Complete

Equipment No.	Building	Elev.	Work Ticket	Description	Status
Relay on Auxiliary Relay Panel FD	RAB	242.5	WR 94ACSW1	Replace non-standard fasteners for relay with manufacturer's recommended hardware.	Scheduled for completion.
Relay on Auxiliary Relay Panel FF	RAB	242.5	WR 94ACSY1	Tighten screws to design values to ensure proper support of relay.	On hold
Relay on Auxiliary Relay Panel FD	RAB	242.5	WR 94ACSZ1	Rework relay model NBFD22S to fully engage the relay to the mounting slots and secure it to the rack with mounting screws.	Complete
Relay on Auxiliary Relay Panel BF	RAB	242.5	None	Calculation required to determine if a portion of the relay that is within one inch of the toe board of the raised floor represents an interaction concern.	Reference memorandum NED-C-0172. *[What's the story here?]*
RPS Cabinet 55	RAB	242.5	WR 94ACTA1	Torque hold down bolts to base of cabinet.	Complete
Miscellaneous Relay Rack 50	RAB	242.5	WR 94ACTB1	Relocate loose cables inside cabinet.	Scheduled for completion
Miscellaneous Relay Rack 50	RAB	242.5	WR 94ACTC1	Repair grout which has deteriorated below west base angle.	Complete
EDG-A Control Switchboard A	RAB	226.0	WR 94AQMW1	Replace missing square nut on door fastener.	Complete
MCC-9	RCB	242.5	WR 04AKKL1	Remove loose anchor bolt.	Complete

Equipment No.	Building	Elev.	Work Ticket	Description	Status
MCC-9	RCB	242.5	None	Tighten anchors to meet GIP requirements.	Work request not initiated at this time.
MCC-16	RAB	242.5	None	Secure electrical bucket which is presently secured only at top of compartment 3M.	Work request not initiated at this time.
MCC-18	RAB	242.5	None	Secure electrical bucket which is presently secured only at top of compartment 3M.	Work request not initiated at this time.

Table 3-2
Balance of Plant Walkdown Work Ticket Items

Equipment No.	Building	Elev.	Description
FY-1425A, FY-1425B, FY1425C	RAB		Install flat plate fittings between enclosure mounting 0ears1 and Unistrut to ensure positive bearing.
FY-1426A, FY-1426B, FY1426C	Turbine Bldg.		Install flat plate fittings between enclosure mounting 0ears1 and Unistrut to ensure positive bearing.
PSL-1476-1			Install missing screw on pressure switch.
LT-1454A	Yard		Reattach loose conduit clamp to secure conduit to CST anchor chair steel.
EV-1711	Pipe Restraint Tower	251.75	Install missing support screw on valve.
PT-117	RAB	246.0	Reinstall loose clamp on conduit running to valve CVC-256 above PR-117.
EDG-A and B Platform	EDG-A and -B Rooms		Move platform step to a location so that it does not represent a seismic interaction concern.
EDG-A Control Switchboard	EDG-A Room		Rework fastener on door to ensure positive restraint of door.
A1-E1/2	Turbine Bldg.	242.5	Remove unrestrained sheet metal cover above pressure switch to alleviate seismic interaction concern.
MCC-9	RAB	246.0	Remove tools that are presently being stored on chain link fence to alleviate seismic interaction concern.

Table 3-3
Recommendations for Plant Modifications

Equipment No.	Building	Elev.	Description	Recommendation for Resolution
HVH-7A & 7B	RAB	226.0	Air handling units have no bracing or lateral supports, resulting in possible unrestrained displacement. Flooding potential also exists due to presence of rigidly attached piping.	Install lateral bracing to increase lateral rigidity and prevent interaction concerns.
TCV-1902A	Turbine Building	226.0	Excessive flexibility in attached conduit and tubing.	Modify current supporting configuration for conduit and tubing near valve body.
ARR A-F and G-M	RAB	242.5	Relay racks share support system with overhead cable trays. Configuration is not similar to equipment in the seismic experience data base.	Locate and eliminate any possible sources of vibration loading to racks.
EDG-A and EDG-B Diesel Control Panels	EDG Rooms		Control panels containing sensitive relays are mounted on diesel generator steel skids and supported on spring isolators.	Change the support configuration to eliminate shock loading impact generated by bottoming out of spring isolators.
ERFIS MUX 1 and 2	RAB	242.5	Cabinets are not bolted together (they contain no sensitive relays) and represent a potential interaction source to nearby sensitive auxiliary relay racks. The existing anchorage for the MUX cabinets is adequate.	Bolt MUX cabinets and adjacent megawatt hour recorder together to form a rigid configuration.
WEST HAGAN RACKS	RAB	254.0	Several cabinets are not bolted together along one edge or along both edges, representing a potential interaction concern.	Bolt adjacent cabinets to one another.

Equipment No.	Building	Elev.	Description	Recommendation for Resolution
PIC-1393	Turbine Building	226.0	Conduit to PIC-1393 has excessive flexibility and is judged to be an interaction concern.	Modify current support configuration to eliminate excessive flexibility.
CVT7.5/INST-1	RAB	242.5	Conduit to transformer has excessive flexibility and is judged to be an interaction concern.	Install independent hanger for conduit adjacent to transformer.
Racks 50, 51-52, 53-57, 58-62, and 63-64	RAB	242.5	Anchorage judged to be marginal.	Add additional anchorage to cabinets.
PCV-456	RCB	288.0	Solenoid valves SV-1 and SV-3, associated with PCV-456, are judged to have insufficient support.	Provide bracket hanger for SV-1 and SV-3 similar to SV-2 and SV-4 which are associated with PCV-455C.
SDAFW Pump	Turbine Building		Several conduits associated with the general configuration of the steam driven auxiliary feedwater pump were judged to have excessive flexibility, creating an interaction concern.	Modify current support configuration to eliminate excessive flexibility.
Charging Pumps B and C	RAB	226.0	Conduit connected to charging pump C was judged to have excessive flexibility, representing an interaction concern. Additionally, conduit, instrument air tubing, and lube oil piping between charging pumps B and C lack support, resulting in excessive flexibility.	Add missing clamp to conduit in ceiling. Install one hanger adjacent to connection lines at each pump.
CC-735	RAB	226.0	Valve operator is in contact with adjacent vertical support member for another pipe.	Reconfigure or move support steel or reposition valve operator.

Equipment No.	Building	Elev.	Description	Recommendation for Resolution
Reactor Trip Breaker Cabinet	RAB	226.0	DB-50, DB-75 and DB-100 breakers have no seismic restraint. Base anchorage and top supports of enclosure judged to be marginal.	Investigate the purchase and installation of seismic retrofit kit from Westinghouse and install to restrain breakers. Install additional base anchorage and enhance existing top support.
EDG-A and EDG-B Air Dryers	RAB	226.0	Dryers are installed atop support pedestal with no positive restraint to pedestal.	Provide positive anchorage of air dryers to pedestal to ensure adequate seismic margin for piping into dryers.
Control Room RTGB, RMS Console, and NIP Cabinets	RAB	254.0	Cabinets are not bolted together and anchorage is marginal.	Bolt cabinets together at one or both edges to resolve interaction concern.
MCC-5	RAB	226.0	MCC is quite long with top supports at each end, but not at interior regions of unit. MCC contains sensitive relays.	Provide additional supports where feasible along top interior part of MCC.
MCC-6	RAB	246.0	MCC floor anchorage judged to be marginal.	Install additional floor anchorage. Bolt adjacent compartments together where required to resolve interaction issue.
E1 Bus and E2 Bus	RAB	246.0	DB-50, DB-75 and DB-100 breakers have no seismic restraint. Trolley used to install and remove breakers is unrestrained and judged to represent a potential interaction concern.	Purchase seismic retrofit kit from Westinghouse and install to restrain breakers. Positively secure the breaker trolley to the cabinet when not in use or relocate off of cabinet.

Equipment No.	Building	Elev.	Description	Recommendation for Resolution
FT-122	RAB	226.0	Nearby unanchored tool cabinet represents potential interaction concern.	Anchor and/or brace the tool cabinet.
PSL-1476-1 and PSL-1476-2	Seismic Class I Turbine Building	226.0	Conduit routed to pressure switch enclosure judged to have excessive flexibility, representing an interaction concern.	Modify current support configuratin to eliminate excessive flexibility.
125V CD MCC-B	RAB	248	Overhead cable tray routed from MCC-B to Station Batteries B is not well supported above the MCC which represents an interaction issue.	Modify current cable tray support configuration above the MCC to resolve the interaction issue.

SECTION 4

INTERNAL FIRES ANALYSIS

4.0 METHODOLOGY SELECTION

Acceptable methodologies for analyzing internal fires are specified in NUREG-1407, Section 4 (NRC, 1991b). Of those methods, a Fire PRA was selected for HBRSEP. Specific fire PRA issues raised in NUREG-1407 were addressed as follows:

Fire compartments of potential risk significance were identified using the initial qualitative and quantitative screening steps defined in the FIVE methodology (EPRI, 1992a) document.

Those fire compartments which did not screen out were subject to detailed modeling described in various procedure guides such as NUREG-2300 (NRC, 1983), "PRA Procedures Guide", NUREG-2815 (NRC, 1985f), "Probabilistic Safety Analysis Procedures Guide", NSAC/181 (EPRI, 1993), "Fire PRA Requantification Studies", and EPRI NP 3385-01 (EPRI 1994b), "Fire Risk Implementation Guide". The COMPBRN IIIe, "An Interactive Computer Code for Fire Risk Analysis", code (EPRI, 1991) together with simplified methods prescribed in the FIVE methodology were used for all deterministic modeling of fire growth and damage. Inter-area and compartment fire propagation analysis was not required based on the review of the fire area and compartment boundaries performed to address NUREG/CR-5088 (NRC, 1989b), "Fire Risk Scoping Study", issues.

Fire frequencies in particular locations accounted for both generic experience (US plant experience obtained from the EPRI Fire Event Data Base) 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.

A qualitative review of the input and modeling uncertainties has been performed. However no formal propagation of those uncertainties through the model was performed or considered of value in terms of providing additional insights.

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

4.1 FIRE HAZARD ANALYSIS OVERVIEW (METHODOLOGY)

4.1.1 Overview

The HBRSEP plant has already undergone an extensive deterministic fire hazards and safe shut down review conducted under the 10 CFR 50 Appendix R program and was demonstrated to be in full 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	Uses 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 reliabilty
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 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 Sections 4.1.3 of this report.

4.1.1.1 Qualitative Screening Analysis of Fire Areas

The major steps are briefly summarized below. Further details of the methodology can be found in Section 5.3 of the FIVE methodology document (EPRI, 1992a).

Essentially, the purpose of this task was to identify the boundaries of the plant fire areas and their respective compartments, 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 plant's Safe Shutdown Analysis and PSA. That information was then initially 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 Fire Areas and Compartments

The plant was initially divided into fire areas which are physically separated from one another by fire rated walls, floors and ceilings which comply with the requirements of the FIVE methodology (definition 2.2). At this stage the FIVE methodology also provides the option of sub-dividing an area into compartments, which are locations within an area separated by non-combustible barriers. Such barriers, although not necessarily fire rated, may provide a significant degree of independence with respect to fire propagation. At the HBRSEP plant, fire areas are in fact sub-divided into zones, which with the exception of those zones associated with the outdoor fire areas, are bounded by three hour fire rated barriers with sealed penetration assemblies having a similar rating. Each fire zone corresponds well to the definition of a FIVE compartment and satisfies the FIVE compartment interaction screening criteria. For the purposes of this IPEEE fire analysis, each zone was therefore treated as being completely independent.

Step 2: Identify Plant Safe Shutdown Systems

The Safe Shutdown Analysis (SSA) and PSA models were reviewed to identify the HBRSEP safe shutdown systems. Both front line and support systems were listed, including balance of plant as well as safety related equipment. In the FIVE methodology, the target shutdown mode of operation selected should be consistent with the plant's PSA (FIVE, Section 2.10). In general, the HBRSEP PSA 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 (CP&L, 1992).

Step 3: Identify Safe Shutdown Equipment in Each Fire Area or Compartment

Safe shutdown components and the associated cabling for the required components are identified in the Safe Shutdown Separation Analysis Report. Based on the above information, a summary of the affected safe shutdown equipment in each fire compartment is documented.

Step 4: Perform Fire Area Safe Shutdown Function Evaluation

Each plant compartment was first evaluated to ascertain whether it contains any susceptible safe shutdown equipment. If so, a demand for shutdown was assumed unless it could be shown with confidence that the fire would not cause an automatic trip or that plant operating conditions or Technical Specifications would not require a shutdown within 8 hours.

The HBRSEP fire analysis incorporates a revision of the FIVE methodology (NUMARC 1993) which requires that fire areas or compartments should not be screened out unless it can be shown that safe shutdown equipment is not damaged <u>and</u> there is no demand for shutdown. (Note: It was not necessary to assume a loss of offsite power as in the case in Appendix R studies, unless there is some potential for the postulated fire inducing such an event as identified in step 3).

4.1.1.2 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 HBRSEP, the fire frequency calculations were performed using the methods provided in the FIVE methodology, Phase 2, step 1, and generic fire data information provided in the Fire Events Data Base (EPRI, 1992b). The approach requires the analyst to weight generic fire data 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 HBRSEP drawings and the EDBS equipment data. The information was modified as necessary as a result of plant walkdowns.

The area/compartment ignition sources and the fire frequency calculations were documented according to the FIVE Attachment 10.2, Table 3, Ignition Source Data Sheet (ISDS). These are included in the tier 2 documentation together with the analysis assumptions and data used.

4.1.1.3 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 1E-6/year:

(i) the total area/compartment fire ignition frequency

(ii) 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 PSA model was used to determine the conditional core damage probability.

4.1.2 Assumptions and Other Modeling Considerations for Screening

4.1.2.1 Success Criteria

In some cases the SSA and PSA are not consistent with respect to the system success criteria. The SSA is generally the more demanding of the two. The specific cases are discussed below and conclusions drawn as to which of the Appendix R SSA systems need not be considered in this analysis with respect to the PSA success criteria.

- The HBRSEP Appendix R SSA includes equipment necessary to achieve hot shutdown (HSD), cooldown (CD) and cold shutdown (CSD). The IPEEE fire analysis was limited to achieving a safe, stable state (i.e., same success criteria as assumed in the PSA) as characterized by a constant RCS temperature, pressure and inventory. Although not specifically stated in the PSA, this criteria was assumed reasonably consistent with a hot shutdown condition associated with Appendix R.

- In the PSA model, rod insertion is sufficient to achieve reactivity control and boration, using the CVCS or SI system, is not a requirement as in the SSA. Furthermore, 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 equipment from either within or outside the control room. The potential for fire events to prevent adequate reactivity control was therefore considered to be insignificant.

- RCS inventory make up from the CVCS or SI, to specifically account for losses, is not considered in the PSA for non LOCA accidents unless feed and bleed is required. The implicit assumption was that letdown isolation is highly reliable and any RCS shrinkage or normal losses will be made up for by the pressurizer. The CVCS system is, however, addressed in the PSA as one means of cooling the RCP Seals in order to prevent RCP seal failures leading to LOCA. In the SSA, the CVCS and SI are challenged for inventory control. Pressurizer PORV isolation is achieved by de-energizing a single

PORV in each of the two lines which causes the valves to fail closed. Similarly, the SSA letdown isolation is achieved by de-energizing associated valves. In the case of the letdown lines, redundant isolation valves are provided; however, both are located in the same area/compartment. Consequently, in the qualitative screening analysis, fire damage to the CVCS/SI shutdown systems and letdown was considered, as well as the possibility of an open PORV. The modeling of fire induced LOCAs was further refined in quantitative screening and detailed modeling, as discussed in section 4.1.2.3.

- Both the SSA and the PSA challenge secondary side cooling which utilizes the motor driven or turbine driven auxiliary feedwater (AFW) pumps taking suction from the CST and feeding to the steam generators. Consequently, the potential for fire damage to the Auxiliary Feed Water system was accounted for in this analysis. Similarly, both analyses utilize the steam generator mechanical safety relief valves which are not susceptible to fire damage. In addition, the SSA addresses the use of the Steam Generator PORV's for cooldown. However, since these are not required for hot shutdown per the PSA success criteria, fire damage to these components was not addressed in this qualitative screening analysis.

- The SSA makes use of the RHR system to maintain the plant in a cold shutdown condition after a fire. Credit was taken for this mode of operation as well as for use of the RHR pumps for cold leg recirculation from the containment sump. This is consistent with the PSA modelling.

- All of the SSA support systems, including Component Cooling, Service Water and Electrical Distribution are required to achieve a safe, stable state within the PSA success criteria and were evaluated for fire damage.

- In the PSA, plant monitoring instrumentation was not specifically modeled. The implicit assumption is that sufficient redundancy is provided such that the availability of instrumentation is not a factor in determining the capability of the operators to achieve a plant shut down. This is also true in the case of a severe fire where an alternate instrument was located in another fire area. However, per the FIVE Methodology, given a single random failure with the fire, the redundant instrument may not be available. Consequently the potential for fire to damage SSA plant monitoring instrumentation required to achieve a safe stable state was evaluated in the IPEEE.

- Passive mechanical components, such as 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 operated after the fire is extinguished if a local operational capability exists (i.e., handwheel)

A summary of all HBRSEP Appendix R SSA systems is included in Table 4.1-1. The potential impact of fires in each fire zone is summarized in Tables 4.1-1a through 4.1-1h.

4.1.2.2 Credit for Non Appendix R Systems

- Unlike the Appendix R analysis, offsite power was assumed to be available following a fire unless cabling for offsite power is present in the fire compartment. Based on the equipment and cable routing location analysis for offsite power systems, it was concluded that a postulated fire in Fire Compartments A/1, A/2, C/5, A/7, A/16, A/18, A/19, A/20, A/22, G/25 or G/26, as well as the Switchyard Area, could result in a loss of offsite power to the 480V ac Emergency Buses E1 and/or E2. Similarly offsite power to the DS bus may be lost due to fires in compartments A/19, A/20, A/22, G/25 and G/26.

- The deepwell pumps, which are non-safety and non Appendix R equipment, can supply water to the AFW suction when the CST is depleted (after about 6 hours). Cables and routes associated with power and control for these pumps were traced. Based on the results of this analysis one or more of the deepwell pumps may be lost due to fires in the compartments C/5, A/16, A/18, A/19, A/22, G/25 and G/26.

- Since the Appendix R study did not consider the routing of cables required for the automatic actuation of emergency systems, no credit was taken for such actuations with the exception of the automatic actuation of the AFW. For the AFW systems, a review was performed to determine routings throughout the plant. The results show that the following fires in the following compartments may disable automatic AFW actuation: fire compartments F/24, D/9, A/21, A/20 and A/19.

- A review was performed which established that a fire in the AFW pump room fire zone (A/6) would not disable the main feedwater/condensate system. The main feedwater system was therefore credited in the analysis of this compartment

- All non Appendix R PSA components with the exception of those discussed above were assumed to be failed in the screening analysis.

4.1.2.3 Fire Induced Opening of Valves in Hi-Lo Interface Pathway

A review was performed to identify those fire compartments where a significant potential may exist for a fire to induce a LOCA due to a breach of the hi-lo interface boundary. This scenario would require:

(i) the fire to cause a sufficient number of electrical faults to cause the spurious opening of valves at the Reactor Coolant System hi-lo pressure boundary, and

(ii) failure of the operator to isolate the LOCA prior to loss of significant RCS inventory.

The Appendix R Separation Analysis identifies hi-lo pressure interface valves. Many of the valves are pneumatic or solenoid operated and fail closed on loss of instrument air or DC power. Furthermore, both pre- and post-fire procedures are in place to de-energize component power supplies in order to minimize the potential for spurious valve operation resulting from hot shorts. Circuit analysis identified the minimum number of coincident hot shorts required to align hi-lo interface pathways. This analyses presumes pre/post fire safe shut down procedures are successfully implemented to de-energize valve control circuits and demonstrates that the only potential mechanism for fire induced LOCA requires multiple external (cable to cable) hot shorts. The probability of such an event occurring (prior to circuit grounding or going open circuit) was considered to be so low as to be non credible for the purposes of both the Appendix R and this IPEEE analysis.

Within the framework of the IPEEE, the possibility of the operator failing to isolate the appropriate circuits, as well as coincident random equipment failures, must be addressed. Given the former event, the potential may exist for a fire to cause spurious valve operation via a single internal hot short (conductor to conductor within a single cable) which is significantly more credible than an external hot short. Once the hi-lo interface pathway has been aligned (due to such a short), subsequent isolation valve closure would be required, which introduces the potential for a hardware failure leading to loss of isolation capability.

A list of the potential hi-lo interface LOCA pathways is provided in Table 4.1-2 together with a summary of the associated pre and post fire mitigating actions and the conclusions of the critical circuit analyses. Each of these pathways was evaluated qualitatively to determine if operator error /delay or hardware failure may result in a significant potential for an unisolated LOCA:

The hi-lo interface pathways fall into three categories:

The first category includes pathways which are protected by a mechanical check valve, as well as a fail closed isolation valve; these include the normal charging and auxiliary spray lines. The probability of the check valve failing to reseat coincident with operator failure to de-energize the circuit and hot shorts leading to spurious valve operation was judged to be negligible. Fire induced LOCA's via these pathways were therefore discounted.

Second, there are pathways for which the control circuit of at least one of the associated valves is normally de-energized prior to plant start up. These include the RHR suction and the Reactor Head and Pressurizer Vent paths. The probability of operator error leading to these control circuits not being de-energized, the fault remaining undiscovered until the time of the fire, and the fire causing control circuit hot shorts which open at least two valves in series was considered to be extremely low. Fire induced LOCA's via these pathways were therefore discounted.

The third category contains pathways for which the control circuits of the associated valves are normally energized but are required to be de-energized according to dedicated shutdown procedures following a fire: these include CVCS Letdown, CVCS Excess Letdown and the

Pressurizer PORV paths. In these cases internal hot shorts may result in an unisolated LOCA given the following:

(i) operator fails to de-energize control circuit prior to core uncovery.

(ii) operator successfully de-energizes circuits but the isolation valves fails to close.

In the case of the CVCS letdown and excess letdown the line sizes (2" dia with flow limiting orifices and 3/4") are such that, even with no charging flow, substantially longer than 2 hours are available to de-energize the valve control circuits and isolate the LOCA before the loss of RCS inventory would jeopardize the core. (This was based on the PSA small break LOCA analysis which indicates 2 hours are available before core damage given a 1.5" dia break). The probability of operator error is therefore very low. Furthermore, since redundant (fail closed) isolation valves are located in each line, the likelihood of not isolating the LOCA due to valve failures is also small. Consequently the possibility of an unisolated LOCA in the CVCS letdown or excess let down paths was discounted.

In the case of the Pressurizer PORV paths, at least 2 hours would be available to close the PORV or block valve based on the PSA small break LOCA analysis. The likelihood of successful operator action is again high. However, the cable associated with the PORV block valves may also be damaged by the fire, leaving the PORV as the only means of isolating the LOCA. Therefore, failure to isolate the LOCA due to random valve failure may be significant.

In conclusion, the only potentially significant mechanism for a fire to induce an unisolated LOCA was judged to be as a result of an internal hot short associated with a Pressurizer PORV circuit. Such circuits are located in fire compartments A/19, A/20, A/21, A/22, A/23, D/9, E/10 and G/25. Consequently, the potential for spurious operation of the PORVs leading to LOCA was modelled for fires in these compartments.

4.1.2.4 Treatment of Operator Actions

The HBRSEP SSA takes credit for post fire manual repositioning/de-energization, in addition to other local actions. The need for such actions in a particular area/compartment indicates a potential degradation of an Appendix R system/component and as such the associated area/compartment was not screened out during the qualitative screening, unless it could be demonstrated with confidence that the impact was insignificant.

During the quantitative screening and detailed analyses, operator error probabilities were developed for each action as discussed in section 4.6.1.

4.1.2.5 Self Ignited Cable Fires

In evaluating self ignited cable fire frequencies the FIVE methodology draws a distinction between IEEE 383 qualified cables and non-qualified cables. FIVE assigns no potential to the former, whereas the frequency associated with the latter is 6.3×10^{-3} per plant year. Electric cables currently being installed at HBRSEP are specifically required to meet the IEEE- 383 standards. During plant construction, cables were tested to determine the flame resistant quality of various coverings and installation; however IEEE 383 testing was not applicable at the time. Such cables were coated with a flame retardant material. For the purposes of evaluating fire frequencies, coated non-qualified cable was assumed to be equivalent to qualified cable in that no propagating self ignited cable fires were considered credible. (However, the cables were treated as non qualified when addressing their susceptibility to damage from exposure fires in the detailed fire modeling evaluations (see section 4.4.1).

4.1.3 Analysis Results

4.1.3.1 Qualitative Screening Analysis

The qualitative screening analysis was completed using the revised FIVE screening methodology as discussed in Section 4.1.1. It includes all plant fire areas and zones that are addressed in the HBRSEP Appendix R Safe Shutdown Analysis (SSA) submittal (CP&L, FSAR) plus three zones which are not included in the report but are shown on the Fire Barrier Drawings (HBR2-9716 series).

4.1.3.1.1 Sub-Division Of Areas into Compartments and Fire Compartment Interaction Analysis

Consistent with the FIVE methodology [FIVE page 2-1 and section 2.2] the analysis adopted the SSA fire area definitions. The areas were further broken into individual compartments which correspond to the designated fire zones, also defined within the SSA. At HBRSEP all fire area and fire zone boundaries with the exception of exterior walls and ceilings, and outdoor areas, are bounded by three hour rated fire barriers with sealed assemblies which have an equivalent rating. All such boundaries are subject to an established maintenance and surveillance program. Consequently at all interfaces the fire zone boundaries satisfy the FIVE screening compartment interaction criteria for establishing independent compartments.

> FIVE Criteria 2: Boundaries that consist of a 2-hour or 3-hour rated fire barrier on the basis of fire barrier effectiveness.

For the purposes of the IPEEE each zone can therefore be treated as being completely independent.

4.1.3.1.2 Qualitative Screening Analysis for the Compartments

Table 4.1-3 presents the results of the qualitative and quantitative screening analysis.

Generally, two screening questions are asked in the table. A "Yes" answer to either of the questions, (i.e. there are Appendix R equipment in the compartment OR a plant shut down required given a fire in the area) indicates that further analysis of the compartment was required. A "No" in the table to both these questions indicates that the fire compartment is screened at this level and no further analysis was required. In some specific cases, noted in the table, fire zones were dismissed for other reasons on a case by case basis. As indicated in Table 4.1-3, ten HBRSEP fire compartments can be screened out by this process:

Fire Compartment A/13 - Chemical Storage/Boric Acid Batch Tank Room
Fire Compartment A/14 - Solid Waste Handling Room,
Fire Compartment H/27 - RHR Pit
Fire Compartment G/28 - New and Spent Fuel and Hot Shop,
Fire Compartment G/31 - Refuelling Water Storage Tank,
Fire Compartment G/32 - Primary Water Storage Tank,
Fire Compartment G/33 - Condensate storage tank,
Fire Compartment G/34 - C Battery Room,
Fire Compartment /35 - Radwaste Building, and
Fire Compartment /36 - "B" and "C" Waste Evaporator Enclosure, and
Fire Compartment F/24 - Containment.

4.1.3.2 Fire Ignition Frequencies for Quantitative Screening

For each compartment that was not screened out in the previous step, estimates of fire ignition frequency were prepared for use in the quantitative screening analysis. These estimates were based on data from the Fire Events Database for US Nuclear Power Plants from the Electric Power Research Institute (EPRI, 1992b) and adjusted for HBRSEP 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 4.1-3 contains a summary of the fire compartment ignition frequencies obtained from the individual ISDS for each HBRSEP compartment. As can be seen from Table 4.1-3, HBRSEP fire compartments do not screen out based solely on fire ignition frequency (i.e. none of the ignition frequencies were below the 10^{-6} per year criteria).

4.1.3.3 Quantitative PRA Screening Analysis

The FIVE methodology includes a second level of screening which provides for a conservative estimation of the contribution to core damage frequency. All equipment cable in a compartment was assumed to fail due to a fire. Using an event tree representative of the most significant failure, the contribution to the core damage frequency was calculated. The initiating event frequency was set equal to the frequency of fire in the compartment. If this contribution was less than 10^{-6}/yr the compartment was screened out. For this analysis, fault tree and event tree models from the PSA were used.

A summary of the screening analysis results is presented in Table 4.1-3. The following compartments can be screened out on the basis that their total contribution to fire induced CDF is less than 1 x 10^{-6} per year:

Fire Zone A6	Auxiliary Feedwater Pump Room,
Fire Zone B4	Charging Pump Room,
Fire Zone A8	Boron Injection Tank Room,
Fire Zone A12	Waste Hold Up Tank/ RHR Hx Room,
Fire Zone A17	HVAC Equipment Room, and
Fire Zone G30	Diesel Oil Storage Tank.

4.1.4 Detailed Fire Modelling

The second part of the fire analysis deals specifically with the potentially significant fire areas 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 zones was damaged. 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 (AFSS), 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. The analysis for the majority of the non-screened compartments is discussed in sections 4.3 to 4.6. The control room analysis uses a somewhat unique approach, which is discussed in sections 4.6.3. The evaluation of the effects of the fire on the containment systems are described in Section 4.7.

4.2 REVIEW OF PLANT INFORMATION AND WALKDOWN

4.2.1 Plant Information Sources

For this analysis, HBRSEP plant information was obtained from plant drawings, plant procedures, and other documents such as the IPE (CP&L, 1991a), the UFSAR (CP&L, FSAR) and the HBRSEP Safe Shutdown Cable/Component Separation Analysis report (CP&L, SSD). The specific sources of information are discussed below.

The HBRSEP Fire Protection Procedures contain 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). Specific examples include: FPP-

007, Fire Fighting Equipment; FPP-012 - Fire Protection - Minimum Equipment and Compensatory Actions; and FPP-014, Control of Fire Barrier Penetrations.

The HBRSEP UFSAR (section 9.5.1C) defines the post fire safe shutdown methodology including the definition of the safe shutdown functions, systems and components. This report also addresses associated circuits and dedicated shutdown (DS) capability. In addition, assessments for all areas containing safe shutdown equipment are provided.

The HBRSEP UFSAR and the Fire Area/Zone location drawing (HBR2-9719) define the fire area and zone boundaries within the plant. It also documents the in-situ combustible loading in each fire zone. In addition, the fire protection systems for the zones are provided.

The above reports provide information on the method of Appendix R compliance, combustible loading analysis, exemption requests and engineering analysis. 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 equipment/cables located in each fire zone, and the combustible loading characteristics, including individual cable tray loadings and flammable liquid inventories.

The HBRSEP cable database was used to identify all Appendix R cabling that runs through specific fire zones. Selected non-Appendix R cables, such as those which support offsite power supplies, were identified using control wiring diagrams and cable/conduit lists. Plant cable tray and conduit drawings were utilized to provide the cable routing within each zone.

Plant drawings were also used for locating Appendix R 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 HBRSEP fire zones.

Post fire safe shutdown procedures (DSPs) were utilized in defining applicable recovery actions and in evaluating the Human Error Probabilities (HEPs).

4.2.2 Outstanding Modifications

Generic Letter 88-20, Appendix 4, section 4.3 requires licensees to provide a discussion of the status of Appendix R modifications. All Appendix R modifications at HBRSEP have been completed.

4.2.3 Plant Walkdowns

Several plant walkdowns were performed for the HBRSEP 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 Fire Risk Scoping Study Evaluation. Another objective was to confirm that 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 (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 screening analysis and obtain specific information on the type and location of ignition sources in each compartment. The locations of all ignition sources were recorded with the aid of simplified sketches showing the compartment layouts.

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:

(i) The type of sealing and venting of electrical cabinets,
(ii) The type of confinement provided for potential oil spills,
(iii) The separation of redundant components/wireways provided within control room cabinets,
(iv) The type and proximity of fire detectors to specific fire sources, and
(v) The proximity of exposed combustibles to ignition sources.

All walkdowns were carried out by NUS and/or CP&L personnel. The participants were either fire protection engineers or PSA/IPE specialists who, between them, possessed the following qualifications:

(i) Familiarity with the Appendix R Safe shutdown paths, equipment and cable raceway layouts and Appendix R Shutdown Procedures,

(ii) Familiarity with the plant fire protection design and standards, including fire barrier characteristics, fire detection and suppression systems and fire prevention measures, and

(iii) Understanding of PSA models and assumptions made in fire PSA analysis.

4.3 FIRE GROWTH AND PROPAGATION MODELING

4.3.1 Fire Scenarios

The basic steps involved in performing the fire modeling are described below. Steps G1-G7 provide general fire hazard and suppression system reliability data, which was subsequently used for evaluating each of the fire compartments in turn .

Step G1: Potential types of target (safety related equipment or secondary combustibles sources) were identified together with their respective criteria for damage and ignition. These are defined in section 4.4.

Step G2: Various types of potential ignition sources in the unscreened compartments were identified and characterized in terms of, frequency, heat release rate and potential total heat release. These are defined in sections 4.3.2.1.1 through 4.3.2.1.5 and the results summarized in Table 4.3-1.

Step G3: Secondary combustible configurations (overhead cable trays) were characterized and the corresponding heat release rates (kW) and potential total heat release (kJ) resulting from secondary ignition were evaluated. This is discussed in section 4.3.2.6 and the results summarized in Table 4.3-2.

Step G4: Based on the results of G1, G2 and G3, the critical separation distance (horizontal and vertical) between different fire sources and targets were determined using conservative fire modeling techniques. The potential for both damage to cable or electrical equipment was considered, as well as the possibility of tertiary ignition of exposed cable (i.e. ignition of cable separated horizontally from the source, due to the combined heat release from the primary fire and a secondary ignition of cable). The fire modeling was performed with the COMPBRN IIIe fire code using conservative input parameters. The results are summarized in Table 4.3-1, Table 4.3-3 and Figure 4.3-1.

Step G5: For each of the various fire sources defined in steps G2 and G3, the ceiling jet layer thickness and minimum horizontal separation distances (for targets in the jet) to avoid damage or secondary/tertiary ignition were evaluated. The fire modeling in this case was performed using the FIVE methodology. The results are summarized in Table 4.3-5.

Step G6: The minimum room volume which may result in the hot gas layer temperature exceeding the damage temperature for electrical equipment and cable was determined. This analysis was performed for a range of total fire heat releases which encompass the ignition sources identified in step G1 and G2. The analysis is summarized in Figure 4.3-2 and the results shown in Table 4.3-4.

Step G7: The reliability of the fire suppression system types installed in the unscreened fire compartments was determined, as was the probability of manual suppression. The approach is discussed in section 4.5.

The remaining steps FP1 - FP9 are performed for each individual compartment. A flow chart through the analysis is shown in Figure 4.3-3.

Step FP1: This step was performed to determine if exposed cable insulation may ignite due to the primary fire source. The burning cable insulation would then become a secondary fire source. COMPBRN analyses have demonstrated that for secondary ignition to occur, the cable must be located in the plume of the primary source (i.e. directly overhead) for all but the largest cabinet fire source. For the large cabinet fire, the cable tray target must be within approximately 1 ft of the plume to cause ignition. Secondary ignition results in a substantially higher heat release, increasing the damage range of the fire as well as the possibility of tertiary ignitions.

For each ignition source in the compartment, the proximity of exposed cable insulation was reviewed and any possibility of secondary ignition was identified. The critical distances for ignition are specified in Table 4.3-3.

If no cases of secondary ignition were identified, Steps FP2 and FP3 are skipped.

Step FP2. If secondary ignition of overhead cable was determined to be possible in step FP1, the additional heat release rate (kW) from the secondary source was determined from Table 4.3-2, based on the number and size of overhead cable trays. The combined total heat release (kJ) of the primary and secondary source was determined from Figure 4.3-1.

Step FP3. The information provided in step FP2 was used initially to determine if the potential for tertiary ignition exists. Tertiary ignition implies ignition of combustibles which are not in (or very close) to the plume of the primary fire. If tertiary ignition is determined to be possible, the generic fire models may not be applicable due to the additional heat release and detailed fire model of the specific configuration is considered unless the effect can be shown to be negligible or bounded in some fashion.

Tertiary ignition can occur via any of the three mechanisms which were evaluated using the steps outlined below:

Step FP3a: The potential for the hot gas layer (HGL) temperature to exceed that required for ignition of cable insulation. The spontaneous ignition temperature for unqualified PE/PVC cable was determined to be 750K.

(i) Based on the total heat release from the primary and secondary source (Step FP2), the minimum free volume necessary to prevent the HGL exceeding 750K was determined from Table 4.3-4 (or Figure 4.3-2 if specific case not evaluated in Table 4.3-4).

(ii) The actual free volume in the compartment was compared to the minimum volume obtained in (i). If the actual volume is less than the minimum volume, tertiary ignition of cable insulation was assumed.

Step FP3b: An evaluation was made to determine if the ceiling jet layer temperature rise at any cable tray location within the compartment would cause tertiary ignition.

(i) Based on the configuration of the fire source (primary ignition source and overhead cable trays) and the ceiling height for the compartment being analyzed, the thickness of the ceiling jet (beneath the ceiling) was determined from Table 4.3-5. If there were no cable trays located within the ceiling jet, tertiary ignition via this mechanism is not a concern.

(ii) If cable trays were located within the ceiling jet, the critical horizontal separation distance from the tertiary cable tray target to the source was determined using Table 4.3-5. If the actual separation distance of the cable trays in the compartment is less than the critical value tertiary ignition was assumed to occur.

Step FP3c: An evaluation was made to determine if tertiary ignition would occur due to direct radiant heat from the fire

(i) Based on the additional heat release rate of the secondary source (determined in step FP2) Figure 4.3-1 was used to determine the critical separation distance to the tertiary cable tray target for ignition. If the actual separation distance of the cable trays in the room is less than the critical value, tertiary ignition was assumed to occur.

Steps FP4-FP9 were only completed using the generic fire models if no tertiary ignition was possible or if the effect can be bounded in some manner. Otherwise a specific fire model for the actual configuration was developed.

<u>Step FP4a, b, c:</u> If no tertiary ignition could occur then steps FP3a, FP3b and FP3c were repeated in order to determine the potential for component/cable damage (rather than ignition). Using this approach a worst case damage state corresponding to each primary ignition source was determined. For transient combustible fires which were capable of causing damage or ignition, the probability that they are located in a space which is within the critical separation distance of the target was determined based on the ratio of the critical location floor area to the total free floor area in the compartment.

<u>Step FP5:</u> If the area is fitted with automatic detection and suppression, an evaluation was made to determine if the response time was fast enough to terminate any the fire scenarios prior to any intermediate damage states.

<u>Step FP6.</u> For each fire damage state, it's associated frequency, based on the frequency of the ignition, probability of being in critical location (for transient fires only) and the probability of suppression system failure were determined.

Step FP7: The conditional core damage frequency for each fire damage state was determined.

Step FP8: The core damage frequency for each state as the product of the fire damage state frequency and the conditional core damage frequency was determined.

Step FP9: For those compartments which could not be screened using the results of the generic fire modeling techniques, more refined compartment specific modeling was performed if a significant reduction in the risk estimate was feasible.

4.3.2 Fire Modeling Including use of COMPBRN

The computer fire code COMPBRN IIIe (EPRI, 1991) and the EPRI FIVE methodology have been accepted by the NRC for performing fire damage calculations for the IPEEE. However, to model each individual ignition source and target would be an extremely time consuming task. Therefore a series of generic COMPBRN analyses were developed which were used to bound the specific configurations at HBRSEP. Generally, the concern was with floor based exposure fires resulting from ignition sources, such as pumps, cabinets or transients, damaging cable or electrical equipment.

Four general types of fire damage phenomena need to be addressed:

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

Representative combinations of exposure fire, target type and configurations were analyzed in order to establish minimum vertical and horizontal separation distances in order to avoid component damage or secondary/tertiary ignition of exposed combustible (exposed cable insulation).

In attempting to utilize the results of a generic set of COMPBRN IIIe analyses there is always the possibility that, for a specific enclosure, the room effects will not have been adequately treated with respect to a specific plant configuration; more specifically, the potential for hot gas layer formation and re-radiation from the walls and ceiling. Care was therefore taken to always model the worst case situation, as follows:

(i) A small enclosure size was modeled (10m long x 5m wide x 5m high). This maximizes the hot gas layer, and ceiling jet layer and plume temperature effects

(ii) The fire compartment is assumed to be unventilated which again maximizes the hot gas layer temperature effects.

(iii) The fire source and targets are located in the corner of the enclosure maximizing the heat radiated back into the fire and onto the target.

For closed rooms COMPBRN models a homogeneous hot gas layer extending from the floor to the ceiling. Consequently all components in a fire compartment are exposed. Because of the relatively small room size chosen to maximize the severity of the other fire damage phenomena in the generic model, the hot gas layer temperatures predicted would be hot enough to cause damage to electrical equipment in most cases and damage to cables in a few cases. However, many of the fire compartments at HBRSEP have a substantially larger volume than that used in the generic model. Consequently a simple model was used to determine if the specific room size is sufficient to prevent (or limit) damage accounting for the specific ignition sources and targets present (see Figure 4.3-2).

4.3.2.1 Fire Modeling Considerations, Inputs and Assumptions

4.3.2.1.1 Pump, Compressor and Fan Fires

The frequency of fires associated with pumps, compressors and fans has been evaluated for each compartment (EPRI, 1993). 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 can burn. In either case the heat release rates are not easily defined. For motor fires a conservative bounding heat release rate equivalent to a small electrical cabinet is recommended (i.e. 69 kW) (EPRI, 1994b). Such fires are assumed to burn at this rate for 30 minutes resulting in a total heat output of 1.2 E+5 kJ.

For oil fires the burning rate must be determined on a case by case basis, using COMPBRN. The spill area accounts for the total oil inventory of the pump reservoirs and any confinements, including trays, dikes, floor slopes and drains, etc. Based on the Fire Events Data Base, 18% of pump motor fires and 2% of compressor fires involved oil spills (EPRI, 1994b). These fractions were factored into the preliminary screening analysis as necessary where such fires were determined to be significant.

Minimum safe separation distances from unqualified cable and electrical equipment targets were evaluated using COMPBRN IIIe, for motor fires (with no significant amount of oil). The results are presented in Table 4.3-1. The minimum room volume required to prevent damage to specific target types due to hot gas layer formation resulting from motor fires is presented in Table 4.3-4.

4.3.2.1.2 Electrical Cabinet Fires

The frequency of fires associated with electrical cabinets in each compartment has been evaluated during the quantitative screening analysis (see section 4.1.3.2). Each cabinet in a given compartment is assumed to have an equal chance of ignition unless otherwise noted.

A review of the EPRI Fire Event Data Base (FEDB) (EPRI, 1992b) indicated that 19% of fires originating in cabinets self extinguished. Consequently, in considering the potential risk from fire propagation, the cabinet fire frequency was reduced by this amount. Those non self extinguishing fires were considered as follows:

The heat release rate from fires associated with major electrical cabinets (e.g. MCCs, Switchgear) is based on a review of the SANDIA cabinet fire tests (NRC, 1987c) performed by EPRI (EPRI, 1994b). The rates were used assuming cables inside cabinets at HBRSEP are generally not qualified.

Cabinet Configuration	Heat Release Rate
Open Top	897 kW (850 btu/s)
Closed Top, Ventilated	413 kW (400 btu/s)

Small cabinets, containing minimal combustibles, are represented by 69kW fires, which corresponds to the heat output from small cabinet fires observed during the Sandia fire tests.

In performing the COMPBRN analysis the heat output rate was ramped up to the above maximums. The profile was based on that observed during tests. As a result the total heat releases from the fires predicted by COMPBRN were approximately 700,000 kJ and 500,000 kJ for the open cabinet and closed cabinets respectively. These results agree closely with the actual tests performed by SANDIA (e.g. the total heat release from Test 25 was 500,000 kJ).

The following approach is used to characterize the specific cabinets at HBRSEP (EPRI, 1994b).

Cabinet Configuration	*Fire Modeling Assumption*
1 No ventilation	• Cannot propagate (self limiting due to oxygen starvation)

2 No top penetration	•	Fire Source at height of ventilation louvers, limited radiation, subtract 20% of HRR (EPRI, 1994b.)
3 Open top penetration	•	Source at top of cabinet. Open cabinet HRR
4 Sealed top penetration	•	"Fire rated" (same as no top penetration) "non fire rated" (same as open top penetration)
5 Top penetration is D<2",L>1'or D=2",L>2'or Conduit has rated seal	•	Same as "No Top Penetration" if otherwise ventilated, same as "No Ventilation" if not otherwise ventilated.

Because of the uncertainty as to the rating of cabinet penetration seals, all cabinets with top penetrations were assumed to be open unless connected to conduits.

COMPBRN IIIe analyses were used to predict minimum safe elevations and horizontal separations from unqualified cable trays and electrical equipment. The results are summarized in Table 4.3-1. The minimum room volume required to prevent damage to specific target types due to hot gas layer formation resulting from cabinet fires is presented in Table 4.3-4. The potential damage ranges are used to exclude specific cabinets as potentially significant fire sources.

4.3.2.1.3 Miscellaneous Small Fixed Source Fires

There are numerous fixed fire sources with a low combustible content which are legitimate fire ignition sources. These sources include battery chargers and inverters, small pumps/compressors with either sealed bearings or lubricated by small amounts of grease or oil, small dry transformers, small electrical panels and small ventilation system fans. Due to their low combustible loading and spatial separation from exposed combustible or safety related equipment, these components can be screened out using engineering judgement. As a reference point the critical damage range calculated for electrical motor winding fires may be used (see Table 4.3-1). In some cases even smaller separation distances may be tolerated without danger of cable or equipment damage. Sealed cabinets are assumed to self extinguish prior to any significant heat release.

Engineering judgement, used to screen these low combustible ignition sources, was based on the proximity of the ignition source to the target (i.e., either safe shutdown equipment or intervening combustibles), and the presence of structures/equipment which could impede fire spread or provide radiant shielding (e.g., walls, non-essential equipment which are non-combustible, curbs, etc.).

Since every case involving these small fixed sources is unique, each was individually analyzed, and if screened, full documentation providing the screening bases is required. The details of the enclosures and location with respect to potential targets was verified by field walkdowns.

A review of EPRI FEDB indicated that 40% of battery charger fires self extinguished. Consequently, in considering the potential risk from fire propagation, the charger fire frequency was reduced by this amount.

Self ignited cable fires were screened out at HBRSEP due to cables being IEEE 383 rated or coated in a flame retardant material. For similar reasons significant junction box/cable splice fires are also ruled out. In the FEDB two such events are reported; however, in neither case did sustained combustion or propagation occur (EPRI, 1993).

4.3.2.1.4 Transient Fire Sources

By their very definition, transient fires can be located in any unoccupied floor space. However, such fires are only significant if they are located within the minimum safe separation distance, or can result in an excessive hot gas layer temperature within the compartment. The probability of such a fire being in a location where damage might result was based on the ratio of the floor area within the safe separation distance and the total unoccupied floor area within the compartment.

The HBRSEP fire protection procedure (FP-005, "Hot Work Permit, Rev. 11") severely restricts the unattended storage of transient fire loads, defined as 0any combustible or flammable material not permanently installed or stored in a designated storage area,1 in safety related areas. The general provisions for transient fire loads in safety related areas are as follows:

- Transient combustibles shall be removed or protected from ignition sources in areas where a Hot Work Permit applies in accordance with FP-005.

- Use of wood is minimized. Where scaffolding and platforms are needed, non-combustible material should be used if possible. All wood in safety related areas during maintenance, modifications, or refueling operations such as lay down blocks or scaffolding shall be flame retardant treated wood.

- Combustible material shall not be left unattended during lunch breaks, shift changes, or similar periods. Immediately following completion of a work activity, or at the end of each work shift, whichever comes first, remove the following from the safety related area:

  a. all waste, debris and scraps
  b. oil spills
  c. flammable or combustible liquids
  d. flammable gasses (acetylene, etc.), oxygen cylinders

e. other combustibles resulting from the work activity

- Equipment or supplies (example: new fuel) shipped in untreated combustible packing containers may be unpacked in safety related areas if required for valid operating reasons. However, all combustible materials shall be removed from the area immediately following the unpacking. Transient combustible materials shall not be left unattended during lunch breaks, shift changes, or other similar periods. Loose combustible packing material such as wood or paper excelsior not removed from the area shall be placed in metal containers with tight fitting, self closing metal covers or approved fire stop tops.

- Only approved fire retardant tarpaulins or plastic sheeting may be used at HBRSEP. Plastic bags lining trash cans and anti-contamination clothing drums are acceptable since fire stop tops are provided on these containers. Small plastic bags used for tools, parts, and small equipment may also be used due to the insignificant fire load. Plastic sleeving may be used to collect and channel leaks and to sleeve hoses but should be kept to a minimum.

  1. Oil rags or solvent soaked rags should be placed in the approved oil rag safety can provided.

  2. Flammable liquids shall be stored and handled in an approved safety can which shall be labeled as to the contents. A maximum of one (1) gallon of a flammable liquid in a safety can may be used in a safety related area.

Exceptions to the above requirements may be granted with written approval from the Fire Protection Staff. The Staff shall review the reason for the material needed, the fire hazard of the material, the quantity, the fire area/zone where the material will be located and the duration the material will be in the safety related area. Additional fire protection features such as an area watch and additional fire protection equipment may be required.

Given these 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 may occur. Therefore, the transient type analyzed for HBRSEP 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 (EPRI, 1994b). The fire types were as follows:

Fire compartment type	Typical fire Fire Size	Worst Case
Frequently occupied by plant personnel	*Human Occupancy*: Polyethylene bag, paper cups, towels	325kW
Only occupied for maintenance/ inspection and occasional operational reasons	*Maintenance Refuse*: Polyethylene wash bottles, buckets, cardboard, Kimwipes®, small amounts of acetone.	145kW
RCA fire compartment where used protective clothing (PCs) may be temporarily discarded	*Protective Clothing:* Protective clothing stored in bins or polyethylene bags	145 kW

The total combustible associated with each of these representative fires is approximately 100,000 kJ.

What type of transient fire, human occupancy trash or maintenance refuse or protective clothing, must be determined for each fire compartment on a case by case basis. In general, most fires modeled in the auxiliary building consisted of maintenance refuse fires, and not human occupancy fires. The latter are consistent with normally occupied or heavily trafficked areas (e.g., offices, designated break areas, etc.). Since the maintenance refuse and protective clothing fires are relatively close in size, the former was chosen to represent both types.

COMPBRN IIIe analyses were used to predict minimum safe elevations and horizontal separations from unqualified cables and electrical equipment. The results are presented in Table 4.3-1. These elevations and horizontal separation distances were used to define the critical transient fire area in each fire compartment. When computing the critical transient fire area, the effects of intervening combustibles which may extend the damage range of the fire were included.

Using a simple floor area ratio, the transient fire IEF can be reduced to a value proportional to the critical floor area (CFA) ratio. For example, given a room floor area of 400 ft^2, and a critical transient fire area of 100 ft^2, the critical floor area ratio is simply 100/400 = 0.25. If the transient fire IEF for the fire compartment was 1.0E-03, it can be reduced by a factor of 0.25 to 2.5E-04. That is, although the transient fire IEF for the fire compartment is 1.0E-03, only 2.5E-04 will be involved in the potential increase in fire induced CDF.

The minimum room volume required to prevent damage to specific target types due to hot gas layer formation resulting from transient fires is presented in Table 4.3-4.

4.3.2.1.5 Welding Fires

Significant welding fires involving cables are not considered to be credible at HBRSEP due to the stringent procedural conditions applied to welding, and the protective fire retardant spray coating applied to all non-IEEE rated cable (new cables installed in the plant are not coated, but are IEEE rated). Any work involving open flames, welding, grinding or temperatures that would exceed the heat of ignition of materials in contact with that work is controlled by using a permit process as delineated in fire protection procedure FP-005 which includes the following requirements:

Welding and cutting equipment is inspected to ensure that it is in good repair.

Within 35 feet of the hot work location, the floor has been swept clean of loose combustibles, wall and floor openings and cable trays have been covered with an approved fire retardant material, and combustibles and flammables have been protected by flame-proof covers, guards or shields.

A fire watch equipped with a portable fire extinguisher is provided during and 30 minutes after work completion to ensure the work area is safe from fire danger.

Given the two lines of defense (i.e., procedures and practices related to "hot work" at HBRSEP, and the additional protection provided to the cables from the fire retardant coating), it was not deemed credible for a welding or cutting operation to produce a viable cable fire. This is supported by the welding fires in the fire events database (FEDB) which were either self-extinguishing or manually suppressed by the welder or fire watch in a short period of time.

Although it could be argued that due to the stringent hot work procedures at HBRSEP, welding/ ordinary combustible fires should be dismissed as well, this single procedural line of defense is not deemed sufficient for screening these fires. Instead, due to their similar natures and combustible types, welding/ordinary combustible fires were treated as a transient combustible fire.

4.3.2.1.6 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. The angle of horizontal spread is assumed to be 35°. The cable tray configuration and degree of fire spread is depicted in Figure 4.3-4.

The above model was used to predict the area of burning cable tray that might be involved in a secondary fire, given a range of overhead cables and sizes. The total heat release rate for each configuration was the calculated assuming a specific heat release rate of 51.89Btu/s/ft^2 (FIVE, Table 1E). The results are shown in Table 4.3-2. The critical separation distance for unqualified cable damage and ignition were then determined using the COMPBRN IIIe code for a range of heat release rates (kW) corresponding to those predicted for secondary fires sources in Table 4.3-2. The critical separation distances versus heat release rate is shown in Figure 4.3-1. Figure 4.3-1 also shows the cumulative heat (kJ) release from secondary fires versus heat release rate (kW). This is computed assuming a 80% cable tray fill and a mass burn out when 30% of the fuel is remaining.

The above approach is adapted from that used in the EPRI Fire PRA Implementation Guide (EPRI, 1994b, Appendix K) for modeling hot gas layers.

4.4 EVALUATION OF COMPONENT FRAGILITIES AND FAILURE MODES

Potential targets fall into the three categories considered below:

4.4.1 Cables serving shutdown equipment:

All cable jacket material installed at HBRSEP is either IEEE 383 rated or has a flame retardant coating applied. For the purposes of the simplified fire modeling a conservative damage temperature of 523K (EPRI, 1994b) is assumed for cables exposed to the fire plume, ceiling jet or hot gas layer. This value is based on SNL oven test data for uncoated PE/PVC cable during which no electrical failures were observed below 523K. COMPBRN IIIe suggests a range of damage temperatures for non coated PVC/PE cable (lower bound 400K, point estimate 500K, upper bound 700K). In modeling cables adjacent to the fire source and thereby exposed to direct radiant heat, damage was assumed if the incident heat fluxes exceeded 5700 W/m^2 cable. This is based on the screening damage criteria of 0.5 Btu/ft^2 recommended by FIVE (EPRI, 1992a, p. 10.4-7).

Based on the tests performed by SANDIA and described in the EPRI fire PRA guide (EPRI, 1994b, Appendix T), it was assumed that the coated non-qualified cables at HBRSEP will not ignite for at least 3 minutes for large exposure fires (open and closed vented cabinets) and at least 10 minutes for small exposure fires (small electrical source).

4.4.2 Electrical Equipment

Electrical equipment, such as cabinets and motors, is generally located just above floor level and is not subjected to the high temperatures associated with fire plumes or hot gas layers. However, damage may occur due to direct radiant heat or, in special circumstances, by a descending hot gas layer. Electrical equipment was assumed to fail if the impinging heat flux exceeds 10kW/m^2 (EPRI, 1992a, p6-14) or if it's temperature is elevated above its damage threshold. The damage temperature assumed are as follows (EPRI 1994b, pG-3):

Sensitive electrical components (eg. solid state equipment)	150°F (339K)
Electric Motors (20% above operating limit)	150°F (339K)
Relays, switches	320°F (433K)

4.4.3 Intervening combustibles

The major source of intervening combustible material is cable insulation. In this case the most critical parameters are the pilot ignition and spontaneous ignition temperatures. The piloted ignition temperature for cable is taken to be 773K (EPRI, 1994b, Appendix G). The spontaneous ignition temperature is conservatively assumed to be the same as the pilot ignition temperature.

4.5 FIRE DETECTION AND SUPPRESSION

The detailed fire analysis involves the evaluation and merging of two competing processes, namely fire progression and fire suppression. Before fire suppression can be possible, successful detection is required. The degree of fire damage is therefore dependent on the timing of detection/suppression as compared to the rate of fire growth and progression.

The detection of a fire event was evaluated based on three parameters: i) the type of fire detection systems available in the compartment under consideration; ii) the location of the detector with respect to a specific source, and iii) the expected environmental changes resulting from the fire. In general, the time to detect an electrical cabinet induced fire by smoke detectors was estimated based on the available experimental data presented in NUREG/CR-4527 (NRC, 1987c). The time to detect a fire by heat detectors was estimated using COMPBRN predictions of the environmental temperature increase, the detector set points, and available experimental data.

Given successful detection, the probability of successful suppression is dependent on:

i) The time available for such actions; this in turn is dependent on the time it takes to reach a particular fire damage state which is estimated based on the results of COMPBRN models and on experimental data where applicable,

ii) The type of automatic fire suppression systems available, and

iii) The availability of manual suppression.

Reliability data for automatic suppression was obtained from NSAC/179L (EPRI, 1994a). Credit can be taken for automatic fire suppression systems (AFSS) if it can be shown that the critical equipment in a fire compartment will not be damaged prior to successful suppression. In some cases, engineering judgment was used to supply justification for credit of an AFSS.

Automatic fire suppression system failure rates (FS) were developed in NSAC-179L (EPRI,1994a), and are the following:

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

Specific consideration was given to the redundancy provided for the Halon systems located in the HBRSEP cable spreading and 480v switchgear rooms (see section 4.6.2.4).

Credit for manual suppression was given if the time interval between fire detection and damage was shown to be greater than the fire fighter response times recorded during drills. If the maximum drill response times was less than the detection-damage interval, the probability of non suppression was taken to be 0.1. If the detection-damage interval fell in the range of drill response times the probability of non suppression was taken to be 0.5.

For each fire source, i, in a compartment, j, the reduced partial CDF due to successful suppression, CDF_{sij}, was calculated as follows:

$$IEF_{ij} \times CCDF_{j} \times FS_{j} = CDF_{sij}$$

where;
IEF_{ij} = IEF for the ith source in compartment j
$CCDF_{j}$ = CCDF for fire compartment j
FS_{j} = AFSS x Manual Suppression failure probability for the system type in compartment j.

For sources where the use of the AFSS and manual suppression was not justified, FS_j was assigned a value of one. The total CDF following the suppression analysis for fire compartment j, CDF_{sj}, is then just the sum of all the partial CDFs from each source in the compartment, that is:

$$CDF_{sj} = \sum_{i} CDF_{sij}$$

4.6 ANALYSIS OF PLANT SYSTEMS, SEQUENCES AND OPERATOR RESPONSE

4.6.1 General Discussion

Given a fire damage scenario, an evaluation of accident sequences is required. This evaluation includes consideration of fire induced initiating event types, components failed by the fire, degraded mitigating system hardware impact on operator response actions modeled in the PSA, as well as additional recovery actions.

Using the results of the fire modeling analyses, which determined the extent and frequency of fire damage for each fire damage state, the CDF is calculated using the HBRSEP PSA models modified to account for the fire induced damage. Consistent with the PSA, the event trees and fault trees were solved using the CAFTA computer code. This code solves Boolean equations using the linked fault tree approach. Component failure probabilities were represented in the fault trees as basic events (i.e., fail-to-start or fail-to-run). If a component could be affected by a fire, a flag was added to the fault tree. The flag was assigned a value of one (i.e. failure) for components in the fire damage state being analyzed. The applicable event trees utilized were the transient and transient-induced LOCA event trees. Human error probabilities (HEPs) were identified as appropriate to account for changes in operator actions as a result of fires. The accident sequences were then re-solved with the revised fire-induced failures and modified HEPs. The resulting cutsets thus contain the appropriate combination of random failures, human errors and failures due to fires.

As mentioned above, HEPs were evaluated to determine the effects of postulated fires on operator actions. There were two categories of operator action evaluated. First, there were the operator actions necessary to mitigate a transient or transient induced LOCA that were modelled in the IPE. Second there were new actions added to the model to account for specific, post fire recovery. These actions included activities such as manual operation of motor operated valves when a fire in another area disables the valve remote operation capability and operator actions necessary to establish control using the alternate shutdown panels.

The first category of operator actions were quantified, as in the IPE, using conservative screening values. Many of these actions were assigned a screening value of 1.0 to allow subsequent dependency analysis on other failed operator actions in the sequence. During the dependency analysis higher levels of dependency were generally assigned to account for the fact that the fire increased the operator burden, stress, etc.

The second category of operator actions added for specific fire recovery, were assigned a screening HEP of 1.0 for quantification. During review of the resulting cutsets, a more realistic HEP was assigned to these actions as follows: The actions of concern are generally valve re-alignments associated with AFW, CCW, SW and SI systems. For most of the scenarios in which they appear, there should be a reasonable amount of time to perform the function. Since the fire initiated scenarios are not large LOCAs or ATWSs, there is at least 45 minutes for recovery. The action was assigned a base line (zero dependence) HEP of .01. This value was then adjusted to account for dependency upon any other operator actions (of the first category) in the cutset. Combinations of specific fire recovery actions (second category) were considered highly to completely dependent upon one another.

It was assumed that adequate indication of key parameters was available to operators for all scenarios. This is because the information is displayed not only in the control room but on several dedicated shutdown panels located in the plant.

4.6.2 Detailed Fire Evaluation of Fire Compartments - Except Control Room

As mentioned in previous sections a total of ten compartments were screened out qualitatively. An additional six were screened out based on conservative analyses which assumed total loss of all equipment in a compartment given a fire occurs. This leaves nineteen compartments for detailed fire analysis. The analysis approach for the control room and hagan room is discussed in section 4.6.3. The detailed analysis of the remaining seventeen compartments is discussed in this section.

In summary this detailed analysis aims to provide answers to the following questions.

1) Given the fire ignition sources identified in the ISDS for an area, what fixed or transient fire ignition sources are potentially capable of causing damage beyond the fire source?

2) What would the severity of a fire be from the fire sources identified as a result of answering question 1?

3) What would the impact of the fire be on the equipment required for the safe shut down of the plant?

4) Can a fire be suppressed before damage occurs?

5) What is the frequency associated with the potential damage?

For the sake of efficiency the detailed fire analysis was performed in two phases:

In the preliminary phase, the fire modelling approach described in section 4.3 was used to identify and screen out fire sources which were incapable of causing damage to safe shutdown equipment or may be extinguished prior to causing such damage.

The preliminary screening quantifications were then performed using an appropriately reduced fire frequency, i.e. discounting the contribution from fire sources which could not cause any damage and reducing the frequency of extinguishable fires according to the reliability of the AFSS. No credit was given for manual suppression during this phase of the analysis. In this analysis the conditional core damage frequencies (CCDFs) derived in the quantitative screening analysis were used (see Table 4.1-3). These CCDFs are often conservative with respect to the actual damage sustained in any specific fire scenario. However, if the compartment screened out (CDF < 1.0×10^{-6} per year), no further analysis was necessary. The results of the preliminary detailed analyses are summarized in Table 4.6-1. An additional nine compartments were screened out at this point. They are:

Fire Compartment A/3	Safety Injection Pump Room,
Fire Compartment C/5	Component Cooling Water Pump Room,
Fire Compartment D/9	North Cable Vault,
Fire Compartment E/10	South Cable Vault,
Fire Compartment A/11	Pipe Alley,
Fire Compartment A/15	Auxiliary Building Second Level,
Fire Compartment A/18	Unit 1 Cable Spreading Room,
Fire Compartment A/21	Rod Control Room, and
Fire Compartment A/23	Hagan Room.

For those compartments which did not screen out, refined detailed analyses were performed for individual fire sources and scenarios to specifically address the actual fire damage which may be sustained (i.e. often limited to overhead cable trays and conduits in the vicinity of the source), as well as taking credit for both automatic and manual suppression.

The details of the analyses for five representative fire compartments are provided in this report; namely for the Diesel Generator "B" Room (A/1), the Auxiliary Building Corridor (A/7), the Battery Room (A/16), the Emergency Switchgear Room (A/20) and the Transformer Yard (A/26). Additional areas requiring detailed analysis are documented in the tier 2 documentation. A compilation of fire scenarios for fire compartments which were examined during this phase of the analysis is presented in Tables 4.6-2a through h. The results of the final CDF calculations are presented in section 4.6.4.

4.6.2.1 Fire Compartment A/1 - EDG "B" Room

General Description: Fire compartment A/1 is the emergency diesel generator "B" (EDGB) room. Automatic fire detection in this fire compartment consists of two infrared flame detectors, a heat detector and two heat actuated devices. Automatic fire suppression equipment in this fire compartment consists of a high pressure CO_2 system which is automatically actuated by the heat activated devices, and also prevents the operation of fuel oil transfer pump B.

The room is large, about 44 feet long, 20 feet wide and 18 feet high, with the EDG in the center of the floor and the control panel, ventilation fan, battery charger, starting air compressor and day tank all lined along the east wall of the room. The EDG control panel is located in the southeast corner.

Targets: The potential targets and associated plant impact can be divided into two groups: those which effect the EDG-B control panel, and those that do not.

The EDG B control panel, if damaged, could result in a total loss of power to emergency bus E2, with an additional loss of both fuel oil transfer pumps A and B, due to selected ground faults and hot shorting of cables in the panel. (While the control panel may become inoperable at temperatures as low as 150 deg F, the actual degradation of insulation material required to cause shorting would not occur unless temperatures were substantially greater). Without the fuel oil transfer pumps, the operation of emergency diesel generator "A" (EDGA) would be limited to the fuel in the day tank which is only sufficient to supply the EDG for approximately 1.5 hours.

Other targets in the room include the EDG B and its remaining auxiliaries. The potential impact of fire damage to these components is limited to loss of EDG B; hence, there would be no perturbation in the plant running status (i.e., offsite power still providing power to all plant buses, and no normally operating equipment would be affected), and the plant would continue to operate. Even if the plant were to be manually tripped, the contribution to CDF is negligible.

Hot Gas Layer: The total room free volume of the fire compartment A/1 is approximately 11,340 ft^3. As discussed above, critical damage to the control cabinet requires a short; the minimum damage temperature for this failure mode is assumed to be is 523 K (i.e. minimum temperature at which non qualified cable degrades). In general, Table 4.3-4 shows that only very large fires can produce a hot gas layer of 523 K or greater, and electric motor, cabinet, or transient fires pose no such threat. A large liquid combustible spill or intervening combustible must ignite to cause a hot gas layer of sufficient temperature to cause significant damage. The potential for such fires is examined below.

Ignition Sources and Associated Fire Scenarios: The ignition sources (and their corresponding contribution to total annual IEF) in this fire compartment consist of EDG B (2.60E-02), the control panel (2.40E-03), transient sources (3.71E-04), welding (8.86E-04), the ventilation system (1.48E-04), the battery charger (3.64E-04) and the starting air compressor (3.36E-04) The total IEF for these sources is 3.05E-02 per year.

The potential for each ignition source to result in damage to the EDG control panel and/or propagate prior to actuation of the AFSS is considered below:

<u>EDG</u> - During normal operation, the EDGs are generally in standby and do not represent an ignition source (i.e., no hot surfaces). Although the diesels are run for periodic or post maintenance tests, an operator would normally be present and any fire would be detected immediately. EDG fires may be associated with the lube oil, fuel oil or exhaust systems. That portion of the EDG containing the lube and fuel oil lines is on the north end of the compartment, with the closest lines being approximately 12 feet from the EDG control panel. Furthermore, as can be seen from the layout drawing, the view of the panel from the potential source of fire is poor. Therefore, only very large fires would likely be capable of causing hot shorts or ground faults in the control panel. Furthermore, the shielding provided for the cables by the metal walls of the panel would provide sufficient time for the heat actuated devices to actuate the CO_2 system. Therefore, AFSS actuation prior to control panel damage is judged to occur.

Control Panel - Due to the nature of the control panel fire, damage is assumed prior to actuation of the AFSS.

Transient Sources and Welding - Since this is not a heavy traffic area, the transient combustible ignited by either the transient or welding source is likely to be the type described as 0maintenance refuse1 in section 4.3.2. Since the control panel damage described above requires cable damage (e.g., shorting), the same horizontal separation distance, six inches, reported in Table 4.3-1 apply. In fact, given the additional radiant shielding provided by the cabinet face and sides (the Table 4.3-1 analyses assumed bare cables in trays), it is doubtful whether damage would even occur; nevertheless, damage is conservatively assumed. Then, given an approximate net floor area of 500 ft^2 (total area minus approximate area of floor mounted equipment), and an approximate critical floor area of 3 ft^2 surrounding the control panel, the CFA ratio corresponding to control panel damage is 6.00E-03.

Ventilation System - The ventilation system consists of a small motor operated fan; therefore, the separation distances for small motors/electric panels apply. The fan is located in the northeast corner of the zone horizontally separated from the control panel by approximately 35 feet. Although the fan is located in the same corner of the compartment as the day tank, the day tank is located on the floor, and the fan is located well above the day tank. Hence, a fire in the fan is unlikely to ignite the day tank contents. Therefore, given the separation distance to the control panel, the low combustible loading and the lack of combustible continuity to the control panel, the fan is judged to damage no equipment other than itself, and is thus screened from further review (i.e., CCDF = 0).

Battery Charger - The battery charger contains a limited amount of combustible; therefore, the separation distances for small motors/electric panels apply. The charger is horizontally separated from the panel by about 6 feet with no intervening combustible. Since the charger separation distance exceeds that for small motors/electric panels no direct damage to the panel is anticipated. Therefore, given the separation distance to the control panel, the low combustible loading and the lack of combustible continuity to the control panel, the charger is judged to damage no equipment other than itself, and is thus screened from further review (i.e., CCDF = 0).

Air Compressor - The small starting air compressor consists of an electric motor coupled to a compressor. The combustible loading of the assembly is equivalent to the motor windings as the bearings contain an insignificant amount of lubricant; therefore, the separation distances for small motors/electric panels apply. The compressor is horizontally separated from the panel by about 25 feet. Since the air compressor separation distance exceeds that for small motors/electric panels no direct damage to the panel is anticipated. The closest intervening combustible is the day tank which is greater than 3 feet from the compressor, and partially shielded by the expansion tank. Hence, a fire in the compressor is unlikely to ignite the day tank contents. Therefore, given the separation distance to the control panel, the low combustible loading and the lack of combustible continuity to the control panel, the compressor is judged to damage no equipment other than itself, and is thus screened from further review (i.e., CCDF = 0).

Preliminary CDF Recalculation: Then, by applying the CFA ratio as described in section 4.3.2 for transients and welding fires, using an FS value of 0.04 for the automatic CO_2 system for EDGB and setting the CCDF to 0 for the ventilation system, battery charger and compressor fires, the CDF for this fire compartment can be re-calculated as follows.

Source	IEF (year^{-1})	CCDF	CFA Ratio or FS	CDF (year^{-1})
Control Panel	2.40E-03	6.36E-03	1.00	1.52E-05
EDGB	2.60E-02	6.36E-03	.04	6.61E-06
Transients	3.71E-04	6.36E-03	6.00E-03	1.42E-08
Welding	8.86E-04	6.36E-03	6.00E-03	3.38E-08
Vent. System	1.48E-04	0.00E-00	1.00	0.00E-00
Battery Charger	3.64E-04	0.00E-00	1.00	0.00E-00
Air Compressor	3.36E-04	0.00E-00	1.00	0.00E-00
Total CDF				2.19E-05

After eliminating as many sources as possible, and crediting automatic fire suppression, the preliminary cdf total is still greater than 1.0E-06 per year, therefore the Fire compartment A/1 was not screened out.

Refined CDF Calculation

Approximately 70% of the contribution to CDF from this compartment arises due to a fire within the EDG control panel. Since the critical damage may occur within the panel itself, no credit can be given for any means of suppression.

The remaining contribution from CDF arises due to a large EDG fires coupled with a random failure of the automatic suppression system to actuate. The ignition frequency model for diesel generators is based on 65 fires which were suppressed as follows:

10 self extinguished
3 de-energized
23 portable extinguisher
1 portable extinguisher and de-energized
3 portable extinguisher and inside hose streams
3 inside hose streams
1 inside hose stream after autogas system failed
1 inside hose stream after autogas system failed

2 automatic gas system
19 unknown

Those fires which self-extinguished or were extinguished using portable extinguishers were obviously small fires and therefore not of a magnitude capable of causing damage like the critical damage postulated in the HBRSEP EDG room. On this basis only 21% of the fires for which the method of suppression is known were potentially significant with respect to the postulated damage at HBRSEP. Assuming that the 19 fires for which the type of suppression is unknown have the same relative breakdown, the frequency of potentially significant EDG fires may be reduced to:

2.6E-02 x 0.21 = 5.46E-03 per year.

The control cabinet and the EDG fire scenarios are summarized in table 4.6-2a.

4.6.2.2 Fire Compartment A/7 - Auxiliary Building Hallway

General Description: Fire Compartment A/7 is the 226'-0" EL. auxiliary building hallway which houses three fire detection compartments, 11, 12 and 13. Fire detection compartment 11 is the hallway near the diesel generators; 12 is the hallway near the air compressors; and 13 is the hallway near the component cooling room. In addition to the hallway described above, Fire Compartment 7 encompasses the demineralizer room, gas stripper and waste evaporator equipment room, boric acid evaporator equipment room and non-regenerative heat exchanger room. These attached rooms are all isolated by fire doors, contain non-essential equipment and have low combustible loadings. The approximate free volume of the fire compartment, not counting the volume associated with the attached rooms is 1700 m^3 (60,000ft^3); this assumes an 18 foot average ceiling height, and assumes that objects in the fire compartment account for approximately 250 m^3 (8,800ft^3). The approximate free floor area for the entire fire compartment is 5000 ft^2; this assumes that approximately 1500 ft^2 of the floor is occupied by equipment.

Automatic fire detection in this fire compartment consists of:

- four heat detectors and four ionization smoke detectors in fire detection zone 11,
- three heat detectors, three ionization smoke detectors and seven photoelectric smoke detectors in fire detection zone 12, and
- four heat detectors and four ionization smoke detectors in fire detection zone 13.

Automatic fire suppression equipment in this fire compartment consists of a preaction sprinkler system. The preaction valve, which opens to charge the spray header, is automatically opened by the detection system; melting of a fusible link opens the individual sprinkler heads. Due to the cramped condition of the ceiling in this fire compartment, the fire compartment was conservatively analyzed assuming there is no sprinkler system.

Targets: The only fire susceptible components in this fire compartment are MCC-5, MCC-10, Lighting Panel 26 and cables in elevated trays and conduit. Damage to these components and/or cables could potentially cause a loss of offsite power and/or full or partial loss of AFW, SW, charging, CCW, SI, Instrument Bus 1, DC Train "A" and EDG B room ventilation resulting in a CCDF of 3.39E-01 (see Table 4.1-3). Note that some degree of separation of these components and cables exist; therefore, individual ignition sources are not anticipated to damage all targets. Furthermore, the potential exists for recovery of some components.

Hot Gas Layer: The approximate free volume of the fire compartment, not counting the volume associated with the attached rooms is 1700m^3 (60,000ft^3). As discussed above the only targets in the compartment consist of MCC-5, MCC-10 and cable. The damage temperature for cables is 523K, while the damage temperature for the MCCs (which are assumed to contain sensitive electrical equipment) is 338K. Based on Table 4.3-4 individual transient fire sources, electric motor or electrical cabinet fires pose no threat to develop an hot gas layer greater than 338K. Either intervening combustibles or large oil spill fires pose the only such threat and are examined below.

Ignition Sources and Associated Fire Scenarios: The ignition sources (and their corresponding contribution to total annual IEF) in this fire compartment consists of twenty electrical cabinets (9.74E-03), two pumps (2.00E-03), transient sources (3.71E-04), welding (8.86E-04), nine transformers (1.11E-03), nine fire protection panels (7.20E-04) and five air compressors (1.18E-03). The total IEF for these sources is 1.60E-02 per year; note that this differs slightly from that IEF given in Table 4.1-1 due to the exclusion of the contributions from welding/cable and junction box sources. Preliminary screening of the electrical cabinets, pumps, transformers, fire protection panels and air compressors is described in the following tables.

Electrical Cabinet Screening

Electrical Cabinet	Configuration	Screening Criteria	Targets Within Damage Range	Screen? (Y/N)
LP-25, LP-26, LP-29 & LP-33 Primary X-tie Heat Trace Panel Secondary X-tie Heat Trace Panel Aux. Waste Disposal Panel Primary Boric Acid Heat Tracing Panel Secondary Boric Acid Heat Tracing Panel Fire Damper Power Supply Panel PASS Panel MCC-1	Sealed panels	No damage outside panels.	N/A	Y
Waste Evap. Equipment Panel Gas Stripper Panel A Gas Stripper Panel B	Closed, Vented Vertical Cabinet - 94" tall	Cable CVSD = 6'-6" Cable CHSD = 2'-6" SEE CHSD = 2'-10"	Trays R-76, R-78, R-79 and R-84	N

Electrical Cabinet Screening				
Electrical Cabinet	**Configuration**	**Screening Criteria**	**Targets Within Damage Range**	**Screen? (Y/N)**
MCC-5	Closed, Vented Vertical Cabinet - 90" tall	Cable CVSD = 6'-6" Cable CHSD = 2'-6" SEE CHSD = 2'-10"	Trays R-10, R-49, R-50, R-73 and R-75	N
Boric Acid Evap. Equipment Panel A Boric Acid Evap. Equipment Panel B	Closed, Vented Vertical Cabinet - 94" tall	Cable CVSD = 6'-6" Cable CHSD = 2'-6" SEE CHSD = 2'-10"	Trays CR100-SA, PR100-SA, R-76, R-78, R-79 and R-84	N
MCC-10	Closed, Vented Vertical Cabinet - 90" tall	Cable CVSD = 6'-6" Cable CHSD = 2'-6" SEE CHSD = 2'-10"	Trays R-10, R-49, R-50 and R-75	N
Waste Disposal Boron Recycle Panel	Closed, Vented Vertical Cabinet - Low Combustible Loading	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	None - Closest target is tray R-79, vertical distance from panel vents to R-79 is 4'-4"	Y

Using the screening criteria given in Section 4.3, thirteen of twenty electrical cabinets can be eliminated from further consideration. To incorporate this into the analysis, the electrical cabinet IEF was reduced. This reduction is accomplished by multiplying the electrical cabinet IEF by a reduction factor (RF) of 0.35 (7/20).

Pump Screening				
Pump	**Configuration**	**Screening Criteria**	**Targets Within Damage Range**	**Screen? (Y/N)**
Service Water Booster Pump A and B	Electric motor driven pump - minimal lubricant	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	None - Closest target is tray R-75 which is elevated over 10' above the floor, the pump motor stands no higher than about 3'	Y

Using the screening criteria given in Section 4.3, both pumps can be eliminated from further consideration. To incorporate this into the analysis, the pump IEF was reduced. This reduction is accomplished by multiplying the pump IEF by a reduction factor (RF) of 0.00 (0/2).

Transformer Screening				
Transformer	**Configuration**	**Screening Criteria**	**Targets Within Damage Range**	**Screen? (Y/N)**
2 wall mounted 25 kVA transformers east of MCC-10 (Detection Zone 12) 3.0 kVA transformer west of FDAP-A1 (Detection Zone 12) LP-29 normal transformer LP-29 alternate transformer	Small sealed transformers	No damage outside transformers	N/A	Y
3 wall mounted transformers above the secondary boric acid heat tracing panel (Detection Zone 13)	Small, vented, dry type transformer	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	None - No overhead targets, closest target is Tray R-75 which is horizontally separated by about 18"	Y
25 kVA PASS transformer	Small, vented, dry type transformer	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	None - Closest target is tray R-85 which is vertically separated by at least 4-1/2'	Y

Using the screening criteria given in section 4.3 all nine transformers can be eliminated from further consideration. To incorporate this into the analysis, the transformer IEF was reduced. This reduction is accomplished by multiplying the transformer IEF by a reduction factor (RF) of 0.00 (0/9).

Fire Protection Panel Screening				
Fire Protection Panel	**Configuration**	**Screening Criteria**	**Targets Within Damage Range**	**Screen? (Y/N)**
FDAP-A1 FDAP-B1	Sealed panels	No damage outside panels	N/A	Y
Transeiver #1,2,3,4,5,6 & 7	Sealed panels	No damage outside panel	N/A	Y

Using the screening criteria given in Section 4.3, all nine fire protection panels can be eliminated from further consideration. To incorporate this into the analysis, the fire protection panel IEF was reduced. This reduction is accomplished by multiplying the fire protection panel IEF by a reduction factor (RF) of 0.00 (0/9).

Air Compressor Screening				
Air Compressor	**Configuration**	**Screening Criteria**	**Targets Within Damage Range**	**Screen? (Y/N)**
Waste Gas Compressors "A" & "B"	Electric Motor and Lube Oil	N/A	None - No targets in compressor room, isolated from fire compartment 15 by fire door 31	Y
Station Air Compressor Instrument Air Compressors "A" and B	Electric Motor	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	None - Closest target is tray R-79 which is elevated over 12' above the floor, the compressor motor stands no higher than about 3'	Y
Station Air Compressor Instrument Air Compressor "A" and "B"	Lube Oil - 4.5 gallons each	Requires detailed analysis	Unknown	N

From Section 4.3.4.1, oil fires only constitute 2% of all air compressor fires. Furthermore, using the screening criteria given in Section 4.3.4, all five air compressor motor fires, and two of five air compressor oil fires can be eliminated from further consideration. To incorporate this into the analysis, the air compressor IEF was reduced. This reduction is accomplished by first dividing the air compressor IEF into those fires associated with oil spill fires (2%), and those associated with motor fires (98%), and then by multiplying the air compressor motor IEF by a reduction factor (RF) of 0.00 (0/5), and by multiplying the air compressor oil fire IEF by a reduction factor (RF) of 0.60 (3/5).

Transient Sources and Welding - Transient combustibles in the compartment are likely to consist of maintenance refuse as defined in Section 4.3.2. From Table 4.3-1, all cables above 6'-11" are safe from damage from these type fires, given no intervening combustibles exist. For cable below 6'-11", the transient source must be 6" or closer to cause damage. The only cable run less than 6'-11" high is an 18' section of tray R-85 which runs along the west wall of the CCW room. This portion of tray R-85 is 1 foot wide; therefore, the critical floor area with respect to this tray is 28.5 ft^2.

The minimum horizontal separation distance for sensitive electrical equipment is 1'-7". There are no components in the compartment containing sensitive electrical equipment required for safe shutdown.

Therefore, the critical floor area (CFA) ratio for transient fires in this compartment can be calculated as the critical floor areas for cables (28.5 ft^2) divided by the free floor area 5000 ft^2, or 5.70E-03.

Preliminary CDF Recalculation: Then, by removing the contribution to total IEF from individual screened components by applying the RFs calculated above, and applying the CFA ratio for transients and welding fires as described in section 4.3, the CDF for this fire compartment can be calculated as follows:

Source	IEF (year^{-1})	CCDF	CFA Ratio or RF	CDF (year^{-1})
Electrical Cabinets	9.74E-03	3.39E-01	0.40	1.32E-03
Pumps	2.00E-03	3.39E-01	0.00	0.00E-00
Transients	3.71E-04	3.39E-01	5.70E-03	7.17E-07
Welding	8.86E-04	3.39E-01	5.70E-03	1.71E-06
Transformers	1.11E-03	3.39E-01	0.00	0.00E-00
Fire Prot. Panels	7.20E-04	3.39E-01	0.00	0.00E-00
Air Comp. (Motor)	1.16E-03	3.39E-01	0.00	0.00E-00
Air Comp. (Oil)	2.36E-05	3.39E-01	0.60	4.80E-06
Total CDF				1.32E-03

Since the total CDF is greater than 1.0E-06 per year, Fire Compartment A/7 is not yet screened.

Refined Fire Scenario Definition and CDF Calculation

Twelve individual ignition sources remain unscreened. Due to similarities in the source types and locations, these twelve scenarios can be reduced to seven. These scenarios were requantified to determine more accurate CCDFs for each source.

Scenario 7-1: The Waste Evap. Equipment Panel and Gas Stripper Panels A and B are all closed vented vertical cabinets approximately 94" tall located on the same side of the hallway outside of the EDG rooms. The same cable trays pass by all three cabinets at the same elevation and horizontal separation; none run directly over these cabinets.

Using Table 4.3-1, no secondary ignition is predicted; hence, the hot gas layer and ceiling jet characteristics are that for the cabinet fire alone.

From Table 4.3-4, the minimum room volume to result in a hot gas layer temperature of 338 K (sensitive electrical equipment damage criteria) is 570 m^3 (20000 ft^3). Since this is smaller than the free volume of the room, no hot gas layer damage is predicted. Furthermore, the maximum plume temperature and ceiling jet thickness are calculated as about 400 °F and 1'-6", respectively. The plume temperature is not sufficient to damage cable, but sensitive electrical equipment would be susceptible. However, since there is no sensitive electrical equipment anywhere near the ceiling, no damage is predicted. The remaining potential source of damage is direct radiant and convective heat from the source fire.

From Table 4.3-1, the CHSD for cable is 2'-6", and 2'-10" for sensitive electrical equipment. Trays R76, R78, R79 and R84 are within the CHSD; the closest sensitive electrical equipment is MCC-5 which is about 12 feet away. The route of all SSD cable associated with these trays, and of cables required for offsite power, deepwell pumps and AFW auto actuation were traced. Only those cables which actually pass over or near the sources were assumed to be damaged. If the route of a cable could not be determined, it was assumed to pass over the source. The results of the analysis are presented in Table 4.6-2c.

Scenario 7-2: The Boric Acid Evap. Equipment Panels A and B are both closed vented vertical cabinets approximately 94" tall located on the same side of the hallway outside of the EDG rooms. The same cable trays pass by both cabinets at the same elevation and horizontal separation. Trays PR100-SA and CR100-SA pass over these panels. Using Table 4.3-1, secondary ignition is predicted for CR100-SA which is within the 3'-6" CVSD for ignition; PR100-SA will eventually ignite once CR100-SA is burning. An additional 483 kW and 210,000 kJ is predicted from these two 18" trays (see Table 4.3-2 and Figure 4.3-1, respectively).

From Table 4.3-4, the minimum room volume to result in a HGL temperature of 338 K (sensitive electrical equipment damage criteria) is 810 m^3. Since this is smaller than the free volume of the room, no HGL damage is predicted. Furthermore, the maximum ceiling jet thickness is less than 1 foot. Since there is no cable or sensitive electrical equipment within about two feet of the ceiling, no damage is predicted. Therefore the only potential source of additional damage is direct radiant and convective heat from the cabinet/cable fire.

From Table Figure 4.3-1, the CHSD for cable is 3'-6", and 3'-8" for sensitive electrical equipment. Trays R76, R78, R79 and R84 are within the CHSD; the closest sensitive electrical equipment is MCC-5 which is greater than 6 feet away. The route of all SSD cable associated with these trays, and of cables required for offsite power, deepwell pumps and AFW auto actuation were traced. Only those cables which actually pass over or near the sources were assumed to be damaged. If the route of a cable could not be determined, it was assumed to pass over the source. The results of the analysis are presented in Table 4.6-2c.

Scenario 7-3: This scenario represents possible damage from transient and welding fires. The room size was previously shown to preclude damage from the formation of excessive hot gas layer temperatures. In fact the only target within the damage range of the maintenance refuse fire was found to be cables in a portion of tray R85. As previously discussed, the combination of CFA ratio and IEF make damage to MCC-5 and MCC-10 insignificant. The route of all SSD cable associated with tray R85, and of cables required for offsite power, deepwell pumps and AFW auto actuation were traced. Only those cables which actually pass through the vulnerable portion of R85, or whose route could not be determined, were assumed to be damaged. The results of the analysis are presented in Table 4.6-2c.

Scenarios 7-4 and 7-5: These scenarios involve oil spill fires of the station or instrument air compressors. Each compressor contains approximately 4.5 gallons of lubricating oil. From Figure 4.3-2, a fire of this magnitude (7.65E+05 kJ) can result in a hot gas layer temperature of 352 K for a volume of 1700 m^3 given that 100% of the oil burns and assuming a heat loss factor of 0.85. Therefore, only sensitive electrical equipment damage is predicted from the hot gas layer (MCC-5 and MCC-10). Additional heat release may be possible from the burning of overhead cable; however, this is judged to not occur for the following reasons.

First, there is a fixed sprinkler system installed overhead in the area containing the compressors, Detection Compartment 12. Although, not credited with extinguishing the fire (due to the placement of the sprinkler heads), they would provide protection against the secondary ignition of the overhead cables by wetting and cooling the cables.

Also, there are floor drains in front of each compressor which would: 1) confine the spill size, thus limiting the HRR, 2) drain some of the oil prior to burning, thus limiting the total amount of heat released. Also, it is highly unlikely that 100% of the oil would spill; some amount would undoubtedly remain in the sump, bearings and piping, thereby again limiting the total heat release.

However, given the high expected HRR, the plume temperature at the ceiling should be sufficient to damage cable. Furthermore, a ceiling jet should form with temperatures most likely capable of damaging cables in the vicinity of the fire. Therefore, since local damage is likely to occur to objects in the plume and ceiling jet (as well as from direct radiant and convective heating), but the hot gas layer is insufficient to damage distant cable, it is assumed that all cable in the hallway near the air compressors is damaged. This will include all cables in trays and conduit running past the air compressors on either side of the hallway.

Finally, note that the hot gas layer temperature is not likely to reach 352 K. This is due to the fact that, as described above, the total heat release is probably less than 7.65E+05 kJ, and more importantly, the room is not truly closed, but is ventilated. If the hot gas layer temperature were less than 338 K, MCC-5 and MCC-10 would not be damaged. However, since the conduits carrying the power supply cables to MCC-5 (which in turn powers MCC-10) from both Bus E1 (DS040A, DS048A and DS048B) and the DS Bus (DS041B) run through Detection Compartment 12, they are therefore assumed to be damaged.

The route of all SSD cables associated with trays near each compressor and of cables required for offsite power were traced. Only those cables which actually pass through the hallway near each compressor, or whose route could not be determined, were assumed to be damaged. The results of the analysis are presented in Table 4.6-2c.

Scenario 7-6: MCC-5 is classified as a sealed cabinet. The result of a fire in MCC-5 is a loss of control power to, or spurious operation of, numerous plant components supplied by MCC-5. The components are listed in Table 4.6-2c.

Scenario 7-7: MCC-10 is classified as a sealed cabinet. The result of a fire in MCC-10 is a loss of control power to, or spurious operation of, various plant components supplied by MCC-10. The components are listed in Table 4.6-2c.

The fire compartment A/7 fire scenarios are summarized in table 4.6-2c.

4.6.2.3 Battery Room - Fire Compartment A/16

General Description: Fire Compartment A/16 is the battery room. Automatic fire detection in this fire compartment consists of two ionization smoke detectors and two explosion proof heat detectors. There is no automatic fire suppression equipment in the compartment; however, a manual hose station and fire extinguisher are provided.

Targets: The targets in this compartment mainly consist of components and supporting DC power trains A and B. These components include 125 VDC MCCs A and B, battery chargers A, A-1, B and B-1, batteries A and B, exhaust fans HVE-8A and 8B, air handling unit cooling A and B, and unit heaters A and B. In addition, cables required for Inverters A and B, Aux. Fuse Panels DC and GC, EDG-A, EDG-B, and the steam driven AFW pump run through the room.

Hot Gas Layer: The total room free volume of Fire Compartment A/16 is approximately 5,650 ft^3, or about 160 m^3. The targets include both sensitive electrical equipment (e.g., DC MCCs, and battery chargers) and cable. The damage temperature for cables is 523 K, while the damage temperature for the sensitive electrical equipment is 338 K. Table 4.3-4 shows that maintenance refuse, electric motor and small electrical cabinet fires are not capable of developing a hot gas layer temperature greater than 338 K. Furthermore, in order to damage cable, the heat released must be greater than that released from a fire equivalent to the closed vented cabinet fire.

Ignition Sources and Associated Fire Scenarios: The ignition sources (and their corresponding contribution to total IEF) in this fire compartment consist of two electrical cabinets (9.74E-04), two battery banks (3.20E-03), transients sources (3.34E-04), welding (8.86E-04), six ventilation systems (8.91E-04) and four battery chargers (1.45E-03). The total IEF for these sources is 7.74E-03 per year. Note that this differs from the IEF in Table 4.1-3 due to the exclusion of the contribution from welding/cable and junction box sources.

Preliminary CDF Recalculation: Only the ventilation fan fires were shown to be incapable of damaging safe shutdown equipment. Since these did not represent a significant fraction of the compartment fire frequency no preliminary CDF calculations were performed.

Refined Fire Scenario Definition and CDF Calculation

Fire Scenario 16-1 - Both electrical cabinets, 125 VDC MCCs A and B have several open conduits and were therefore treated as open cabinet configurations (897kW). Due to the small size of the battery room (160 m^3), damage to the redundant DC MCC due to excessive hot gas layer formation may occur within 5 minutes after ignition. This was determined using the approach described in Figure 4.3-2. COMPBRN modelling of the hot gas layer formation in the room was also performed in an effort to take credit for the forced room ventilation system. However, no significant increase in the time to damage was observed. Since the fire brigade response time to this area (based on drills) is between 6 and 13 minutes, manual fire suppression may not be possible prior to damage and was not credited.

Fire Scenarios 16-2 and 16-3 - The batteries are contained in two racks. A rack on either side of the room supplies emergency DC power to it's respective DC MCC. Failure of an individual battery train would have a minimal effect on CDF as the batteries are only used as a backup source for the inverters and dc power. Thus fire damage which is confined to the battery itself is not significant.

The batteries are of the wet cell variety with plastic cases; similar to car batteries. Fires involving these type batteries are typically caused by faults at terminal connections; intense fires involving burning of the cases are not common. Of the four events reported in the data base (EPRI, 1992b), one self extinguished, two were suppressed with portable extinguisher and the method of extinguishing the fourth is unknown. Therefore, battery fires will be characterized as a small electric panel fire (69kW). Recall that this size fire will not produce a damaging hot gas layer. Given the geometry in this room, the ceiling jet from either battery fire would be defined as "unconfined". Therefore, using Table 4.3-5a, no damage from the ceiling jet would be expected. However, above each rack is a set of cable trays which, if damaged, could result in a loss of power from that train's DC MCC. The trays over both battery racks are within the CVSD; therefore, damage is assumed.

Since the frequency of battery fires appears to be trending down, and at least one of the fires self extinguished (and is therefore presumed to be insignificant), the frequency of fires capable of propagating at HBRSEP has been reduced by a factor of 2 to 8.0E-04/yr per battery bank.

Transient combustibles in the compartment are likely to consist of maintenance refuse, as defined in Section 4.3.2. From Table 4.3-1, all cables above 6'-11" are safe from damage from these type fires, given no intervening combustibles exist. For cable below 6'-11", the transient source must be directly under the tray or separated by no more than 6" to cause damage. As shown in Table 4.3-1, secondary ignition of cable from maintenance refuse fires is not predicted. Lastly, from Table 4.3-4, maintenance refuse fires must be confined in a maximum volume of 125 m^3 to be of concern with respect to HGL temperatures exceeding the damage criteria for sensitive electrical equipment. No damage due to hot gases is predicted; however damage to the overhead cable trays running on either side of the room may occur.

In both the case of battery fire and transient fire damage manual suppression prior to damage is possible. Damage to overhead cables will not occur for 10 minutes. Since the response time to this area during drills is in the range of 6 - 13 minutes, a probability of non suppression of 0.5 was applied (see section 4.5).

The frequency for each battery rack scenario 4.00E-04, can be calculated as the product of the fire frequency for one battery rack, 1.60E-03, the reduction factor for fires capable of propagating (0.5) and 0.5. The frequency for transient sources scenarios 8.35E-05, can be calculated as the product of the fire frequency for all transient sources, 3.34E-04, the critical area ratio (0.5) and the probability of none suppression. The IEF for welding fires, 6.65E-05, can be calculated as the product of the IEF for all welding fires, 8.86E-04, the critical area ratio and probability on non suppression. The IEF for both scenarios, 5.50E-04, is then just the sum of the frequencies for the battery rack, transient and welding fires. The results of the analysis are presented in Table 16-1.

Fire scenarios 16-4, through 16-7 - Four battery chargers are located in the room. As in the case of the batteries, failure of an individual charger would have a minimal effect on CDF. Thus fire damage which is confined to a charger itself is not significant.

Battery chargers are classified as small electrical fire sources (69kW) due to their low combustible loading and therefore do not present a threat due to hot gas layer formation in this room for the same reasons discussed under battery fires. They do however present a threat to overhead cable trays and conduits

In both the case of battery fire and transient fire damage manual suppression prior to damage may be possible but has not been credited in the analysis due to the low risk contribution of these fires.

Consistent with the approach discussed in section 4.3.2, the frequency of battery charger fires which are capable of propagating (i.e. do not self extinguish) was determined to be 60% of the total frequency.

The frequency for each battery charger scenario, 1.09E-04 can be calculated as the product of the battery charger fire, the reduction factor for non propagating fires (0.6) and the probability non suppression (0.5).

Based on a similar rationale to that described for scenarios 16-2 and 16-3, a probability of non suppression prior to damage of 0.5 was applied.

The frequency for each battery charger scenario, 1.09E-04 can be calculated as the product of the battery charger fire, the reduction factor for non propagating fires (0.6) and the probability of non suppression (0.5).

The fire compartment A/16 fire scenarios are summarized in Table 4.6-2d.

4.6.2.4 Emergency Switchgear Room -Fire Compartment A/20

General Description: Fire Compartment A20 is the emergency switchgear and electrical equipment room. Automatic fire detection in this fire compartment consists of four photoelectric smoke detectors, four ionization smoke detectors and six heat detectors. Automatic fire suppression equipment in this compartment consists of a halon system which is automatically activated by at least one detector on each train.

The halon system consists of two independent redundant trains which are shared with the cable spreading room. Section 4.5 lists the individual halon suppression system failure rate, FS, as 0.05 for a single train. The total failure rate (F_{tot}) for two independent trains can be calculated as follows:

$$F_{tot} = [FS \times (1\text{-}b)]^2 + [FS \times b]$$

where b = conditional common cause failure factor (beta factor). Using a generic beta factor of 0.1, F_{tot} = 7.03E-03. Note that although both halon system trains utilize the same detectors, detection failure would not be significant due to the redundancy and uniqueness of the detectors.

Targets: The targets in this compartment consist of Emergency Buses E1 and E2, MCCs 2, 6 and 9, Instrument Buses 1, 2, 3, 4, 6, 7, 8 and 9, Inverters A and B, and cable in trays and conduit.

Hot Gas Layer: The approximate free volume of the fire compartment, not counting the volume associated with the attached rooms is 570m^3 (20,100ft^3). As discussed above the only elevated targets are cables. The damage temperature for cables is 523K, while the damage temperature for the busses and MCCs (which are assumed to contain sensitive electrical equipment) is 338K. Based on table 4.3-4 individual transient fire sources and small electrical sources pose no threat to develop an HGL greater than 338 K. However, open cabinet fires may produce a damaging HGL.

Ignition Sources and Associated Fire Scenarios: The ignition sources (and their corresponding contribution to total IEF) in this fire compartment consist of twenty-one electrical cabinets (7.50E-03), transients sources (3.34E-04), welding (8.86E-04), fifteen transformers (1.85E-03) and twelve fire protection panels. The total IEF for these sources is 1.15E-02 per year. Note that this differs from the IEF in Table 4.1-1 due to the exclusion of the contribution from welding/cable and junction box sources. Preliminary screening of the ignition sources is performed below:

<u>Electrical Cabinets</u> - The following electrical cabinets are contained in this fire compartment:

Electrical Cabinet Screening				
Electrical Cabinet	**Configuration**	**Screening Criteria**	**Targets Within Damage Range**	**Screen? (Y/N)**
Emergency Bus E-1, E-2, MCC-2, MCC-6, MCC-9	Open Top Vertical Cabinet	HGL < 338K	All sensitive electrical equipment	N

Electrical Cabinet Screening				
Electrical Cabinet	**Configuration**	**Screening Criteria**	**Targets Within Damage Range**	**Screen? (Y/N)**
Pressurizer Heater Control Cabinet	Small Electric Panel	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	R68	N
Inverter "A"	Small Electric Panel	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	Trays R65, R73, R74 and Inverter "A"	N
Inverter "B"	Small Electric Panel	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	Inverter "B"	N
Inverter "C"	Small Electric Panel	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	None	Y
Auxiliary Relay Racks 50 - 64	Open Top Vertical Cabinet	HGL < 338K	All sensitive electrical equipment	N
Auxiliary Relay Rack Inverter	Small Electric Panel	Cable CVSD = 3'-6" Cable CHSD = 0'-0" SEE CHSD = 1'-0"	Trays R7, R39, DS224, DS225 and Auxiliary Relay Rack 64	N
Instrument Buses 1-4, 6-9	Sealed Panels	No damage outside panel.	Individual Instrument Bus	N

Using the screening criteria given in Section 4.3, only one electrical cabinet (Inverter C) can be eliminated from further consideration. To incorporate this into the analysis, the electrical cabinet IEF should be reduced. Normally this would be accomplished by multiplying the electrical cabinet IEF by a reduction factor (RF). However, in this case a different method will be used.

The standard RF method was judged to be not suitable in this case due to the varying cabinet types and sizes, i.e., a small instrument bus should not be weighted as heavily as a large bus or MCC with respect to probability of fire occurrence. An alternative approach is to assume that larger cabinets contain more wiring and more complex circuitry than smaller cabinets or panels; hence, the likelihood of fires increases.

A simple method of dividing the IEF between the cabinets is to assign a fraction of the total IEF to a cabinet in proportion to its fraction of the total floor area of all the cabinets. Using this method the fractional area of Inverter C is calculated to be 7.94E-03. Then, the reduction factor (RF) for the cabinets is set to 1 - 7.94E-03, or 9.92E-01.

<u>Transient Sources and Welding</u> - This fire zone is not a heavy traffic zone, and transient sources in the zone are likely to be the type described as maintenance refuse fires in Section 3.3.4. As stated above, transient fires can not produce a significant hot gas layer in this fire zone unless

intervening combustibles are involved. The only significant source of intervening combustibles in this fire zone is cable insulation. However, as shown in Table 4.3-1, maintenance refuse fires are incapable of igniting cable; therefore, only direct heating effects need be examined.

With respect to overhead cable, only a section of tray R6 is within the CVSD of 6'-11" (see Table 4.3-1). Using the CHSD for sensitive electrical equipment of 1'-7" (see Table 4.3-1), the CFA for this failure mode could be calculated. However, due to the insignificance of this source with respect to electrical cabinets, a detailed CFA calculation was not performed; instead an estimate of 10% of the floor area was assumed critical (i.e., CFAR = 0.10). This estimate is based on a visual inspection of layout drawings.

Successful fire detection and suppression via the AFSS is judged to occur prior to equipment damage in this fire zone due to the amount of smoke likely to be emitted from this type of combustible and the redundancy of smoke detection devices in the room. Furthermore, as described in Section 4.5, a conditional manual non-suppression failure probability of 0.15 was included for welding sources. Therefore, the reduction factor (RF) for transient fires is calculated as the product of the CFAR and F_{tot}, 7.03E-04; and the RF for welding fires is 1.05E-04.

Transformers - All fifteen transformers are small, wall mounted, dry type transformers. These transformers contain very little combustible material, and judged to be equivalent to small electric panel sources. They have no cable or other intervening combustibles either overhead or laterally within at least a foot. The transformers can be screened since no damage to targets in the zone is anticipated (see Table 4.3-1 for damage ranges). Therefore, given the small amount of combustible associated with these transformers, the lack of intervening combustibles and the spatial separation between these transformers and any potential target, these transformers are judged to be insignificant fire hazards, and are screened from further review (i.e., CCDF = 0).

Fire Protection Panels - The twelve fire protection panels include the Fire Door Monitoring Relay Panel, FDAP-A2, FDAP-B2, and Transceivers 8, 9, 10, 11, 12, 13, 14, 15 and 16. All of these sealed panels conform with the description for non-vented cabinets which are assumed to self-extinguish prior to damaging any other equipment (see cabinet configuration #1, Section 3.3.2). Therefore, their contribution to CDF is judged to be negligible (i.e., CCDF = 0).

CDF Recalculation: Then, by applying the RFs calculated above for the electrical cabinets, transient and welding sources, and by inserting a CCDF of 0 for the transformers and fire protection panels, the CDF for this fire zone can be re-calculated as follows:

Source	IEF (year^{-1})	CCDF	RF	CDF (year^{-1})
Electrical Cabinets	7.50E-03	4.50E-01	9.92E-01	3.35E-03
Transients	3.34E-04	4.50E-01	7.03E-04	1.06E-07
Welding	8.86E-04	4.50E-01	1.05E-04	4.19E-08
Transformers	1.85E-03	0.00E-00	1.00	0.00E-00
Fire Prot. Panels	9.60E-04	0.00E-00	1.00	0.00E-00
Total CDF				3.35E-03

Since the total CDF is greater than 1.0E-06 per year, Fire Zone A20 can not yet be screened from further quantitative analysis.

Fire Scenarios: Note that the CDF is dominated by the electrical cabinets. In fact, the total CDF from transient and welding sources is only 1.48E-07. So, due to their low contribution to total CDF, transient and welding fires were not modeled further (i.e., they are allowed to damage all cable and cabinets in the room). Only scenarios developed for the thirty-two unscreened electrical cabinets were developed.

Each scenario included a loss of AFW auto actuation since analysis determined that cable relating to the auto actuation of AFW is contained somewhere in this fire zone. Finally, if the AFSS is successfully actuated, the fire will be extinguished prior to any damage outside the cabinet.

Scenarios 20-1 to 20-8: These scenarios describe the damage states resulting from fires in Instrument Buses 1, 2, 3, 4, 6, 7, 8 and 9, respectively, whether or not the AFSS fails. Since each source is classified as a sealed panel, damage to the source alone (and the equipment it controls) is always assumed. All SSD, offsite power and deepwell pump cables associated with the instrument buses were identified, and assumed to be damaged. The IEF for each of the instrument buses, based on a fractional area of 3.17E-03, is 2.38E-05.

Scenario 20-9: This scenario describes the damage state resulting from a fire in the pressurizer heater control panel if the AFSS fails. If the AFSS is successful, only the control panel will be damaged, and since this is not significant with respect to CDF, only the AFSS failure case need be examined. As described above, tray R68 is located within the CVSD of this panel. All SSD, offsite power and deepwell pump cables associated with the control panel or tray R68 were traced. Only those cables which actually pass over or near the panel were assumed to be damaged. The IEF for the pressurizer heater control panel, 4.19E-07, is calculated as the product of the total IEF for all cabinets in Fire Zone A20, 7.50E-03, the fractional area associated with the panel, 7.94E-03, and F_{tot}, 7.03E-03.

<u>Scenario 20-10</u>: This scenario describes the damage state resulting from a fire in Inverter A if the AFSS fails. If the AFSS is successful, only the inverter will be damaged, and since this is not significant with respect to CDF, only the AFSS failure case need be examined. As described above, trays R65, R73 and R74 are located within the CVSD of this inverter. All SSD, offsite power and deepwell pump cables associated with Inverter A or trays R65, R73 and R74 were traced. Only those cables which actually pass over or near the inverter were assumed to be damaged. The IEF for Inverter A, 4.19E-07, is calculated as the product of the total IEF for all cabinets in Fire Zone A20, 7.50E-03, the fractional area associated with the panel, 7.94E-03, and F_{tot}, 7.03E-03.

<u>Scenario 20-11</u>: This scenario describes the damage state resulting from a fire in Inverter B if the AFSS fails. If the AFSS is successful, only the inverter will be damaged, and since this is not significant with respect to CDF, only the AFSS failure case need be examined. As described above, no trays are located within the CVSD of this inverter. All SSD, offsite power and deepwell pump cables associated with Inverter B were traced. Only those cables which actually pass over or near the inverter were assumed to be damaged. The IEF for Inverter B, 4.19E-07, is calculated as the product of the total IEF for all cabinets in Fire Zone A20, 7.50E-03, the fractional area associated with the panel, 7.94E-03, and F_{tot}, 7.03E-03.

<u>Scenario 20-12</u>: This scenario describes the damage state resulting from a fire in the Auxiliary Relay Rack Inverter if the AFSS fails. As described above, trays R7 and R39, and conduit DS224 and DS225 are located within the cable CHSD and CVSD of this inverter. In addition, Auxiliary Relay Rack 64 is located within the sensitive electrical equipment CHSD of this inverter. All SSD, offsite power and deepwell pump cables associated with the Auxiliary Relay Rack Inverter, Auxiliary Relay Rack 64, or trays R7 and R39 were traced. Only those cables which actually pass over or near the inverter were assumed to be damaged. In addition, the cables associated with conduit DS224 and DS225 were identified, and assumed to be damaged. The IEF for this Auxiliary Relay Rack Inverter scenario, 1.76E-06, is calculated as the product of the total IEF for all cabinets in Fire Zone A20, 7.50E-03, the fractional area associated with the panel, 3.33E-02, and F_{tot}, 7.03E-03.

<u>Scenario 20-13</u>: This scenario describes the damage state resulting from a fire in any auxiliary relay rack if the AFSS fails. If the AFSS is successful, only a single relay rack will be disabled and since this is not significant with respect to CDF, only the AFSS failure case need be examined. As described above, failure to extinguish these open top cabinet fires would result in damage to all sensitive electrical equipment in the fire zone. Furthermore, there are numerous trays located overhead which, if ignited could possibly increase the total heat released to a point where the remainder of the cable in the zone would become damaged. Given that both bus E1 and E2 contain sensitive electrical equipment, all safe shutdown equipment powered by these buses would be failed even prior to ignition of any overhead cable. The IEF for all auxiliary relay racks in this scenario, 1.32E-05, is calculated as the product of the total IEF for

all cabinets in Fire Zone A20, 7.50E-03, the total fractional area associated with all of the racks, 2.51E-01, and F_{tot}, 7.03E-03.

<u>Scenarios 20-14, 20-16, 20-18, 20-20, 20-22</u>: These scenarios describe the damage states resulting from fires in 480 V buses El and E2, and MCCs 2, 6 and 9, respectively, given AFSS success. If the AFSS is successful, only the cabinet where the fire started will be damaged; however, since these cabinets control SSD, their damage may be significant with respect to CDF. All SSD, offsite power and deepwell pump cables associated with these cabinets were identified, and assumed to be damaged. The IEFs are 1.71E-03 for bus El, 1.97E-03 for bus E2, 4.73E-04 for MCC-2, 6.42E-04 for MCC-6 and 1.18E-04 for MCC-9. These IEFs are calculated as the product of the total IEFs for all cabinets in Fire Zone A20, 7.50E-03, the fractional areas associated with each bus, 2.29E-01 for bus El, 2.64E-01 for bus E2, 6.35E-02 for MCC-2, 8.63E-02 for MCC-6 and 1.59E-02 for MCC-9, and (1 - F_{tot}), 9.93E-01.

<u>Scenarios 20-15, 20-17, 20-19, 20-21 and 20-23</u>: These scenarios describe the damage states resulting from a fire in any 480 V bus or MCC in Fire Zone 20 if the AFSS fails. As described above, failure to extinguish these open top cabinet fires would result in damage to all sensitive electrical equipment in the fire zone. Furthermore, there are numerous trays located overhead which, if ignited could possibly increase the total heat released to a point where the remainder of the cable in the zone would become damaged. Given that both bus El and E2 contain sensitive electrical equipment, all safe shutdown equipment powered by these buses would be failed even prior to ignition of any overhead cable. The IEFs are 1.21E-05 for bus El, 1.39E-05 for bus E2, 3.35E-06 for MCC-2, 4.55E-06 for MCC-6 and 8.37E-07 for MCC-9. These IEFs are calculated as the product of the total IEFs for all cabinets in Fire Zone 20, 7.50E-03, the fractional areas associated with each bus, 2.29E-01 for bus El, 2.64E-01 for bus E2, 6.35E-02 for MCC-2, 8.63E-02 for MCC-6 and 1.59E-02 for MCC-9, and F_{tot}, 7.03E-03.

Based on fire drills manual fire fighter response times to this fire compartment are between 6 and 9 minutes. Since damage to overhead cables will not occur for 10 minutes, a probability of non-suppression prior to damage of 0.1 was applied in addition to the credit taken for the automatic suppression system.

The fire scenarios described above are summarized in Table 4.6-2f.

4.6.2.5 Transformer Yard - Fire Compartment G/26

Fire Compartment G/26 is the yard transformer area which includes the three Main Transformers, the Auxiliary Transformer, the Start Up Transformer and a spare. Automatic fire detection and suppression in this fire compartment consists of heat actuating devices which activate the open head deluge systems on all yard transformers (except the unconnected spare unit which is not considered to be a fire source). Each transformer contains several thousand gallons of oil, which in the event of leakage would be retained within two pits. All three Main Transformers share the same oil collection pit and deluge system, and the Auxiliary and Start-Up Transformers share the other collection pit and deluge system.

Transformers such as the Start Up Transformer and the Auxiliary Transformer use an inhibited mineral insulating oil as a cooling medium. The oil is circulated by pumps around the hot core of the transformer and then transports the heat to the cooling radiators on the outside of the transformers. Most accidents where the transformer oil becomes involved in a fire occur due to a rupture of the transformer casing resulting from an internal fault. The internal fault occurs when explosive gases such as carbon monoxide build up inside the casing due to a degradation of the transformer internals. The fault may eventually lead to electrical arcing which provides an ignition source for the flammable gases that have formed. When this internal fault erupts, the pressure causes the oil to be vented out through a relief valve, or in more severe cases the casing of the transformer splits. This can result in much of the insulating oil being dumped to the ground in the area surrounding the transformer. The oil is usually then ignited and burns across the surface area of the oil.

As discussed above, at HBRSEP the propagation of burning oil is mitigated by concrete lined catch basins which are filled with stone. The voids around the stone trap the oil. Since the transformers are fitted with automatic suppression, it is expected that in addition to the oil spilled, deluge water will be dispersed around the transformers accelerating the filling of the basin. Adjacent to the transformer basin are large storm drains. Ground elevation is sloped from a high point to the west of the Auxiliary Transformer to a low point east of the Start Up Transformer. Consequently any overflow of the basin, should it occur, is likely to travel to the storm drains where it would be carried away from the protected area to the settling ponds. In fact, during persistent rains on 6-6-95, water flow was noted leaving the area of the transformers and travelling east to the storm drain.

Targets The only potential targets of concern in the immediate transformer yard are the Start-Up Transformer and cables/bus ducts which supply offsite power to the 4.16 kV busses. However, a conduit associated with the DS Diesel is routed on the outside of the turbine building approximately 20' from the closest transformers and at an elevation of 14' above grade. Given the potential size of a fire in this area, the turbine building contents themselves must also be considered as potential targets. In particular the 4.16kV switchgear room, which contains DS equipment and controls may be susceptible to damage or be uninhabitable due to smoke ingress. Use of other DS local control stations within the turbine building may also be hampered due to smoke, although operators are trained to use breathing apparatus. Possible intermediate combustible sources within the turbine building include cable insulation and the hydrogen seal oil system unit. The latter is located approximately 30' from the Unit Auxiliary transformer with no intervening physical barrier.

Hot Gas Layer: This is an open area and thus not an issue. However smoke ingress into the turbine building must be addressed, as discussed above.

Fire Frequencies and Scenarios Based on Generic Data in FEDB

The EPRI FEDB subdivides yard transformer fires into the following categories:

	Frequency (/year)
Fires Propagating to the Turbine Building	4.0E-03
Fires with Loss of Offsite Power	1.6E-03
Other Yard Transformer fires	1.5E-02

Three of the five recorded events, which propagated to the turbine building, were caused by an explosive failure mode of the transformer. Oil floating on top of excess water from the deluge system transported the fire to the turbine building. In another case a switchgear in the turbine building was re-energized and caught fire due to the same electrical fault. In the fifth case only damage to the outside of the turbine building was noted. Fire suppression required automatic deluge or hose stream. Fire durations are known for three of the fires and were 40 minutes, 1 hr 20 minutes and 3 hours.

Two events are recorded in the FEDB as giving rise to a loss of offsite power. One was extinguished manually and offsite power was recovered within 38 minutes. The other fire was extinguished in 20 minutes and offsite power recovered in 0.17 minutes.

Eighteen other yard transformer fires are recorded in the data base with fire durations ranging from 3 minutes to two hours.

Fire Frequencies and Scenarios Based on Generic Data in FEDB

Although the FEDB categorization of the data implies that fires which propagate to the turbine building and those which result in loss of offsite power are mutually exclusive, this is not necessarily the case given the juxtaposition of the transformers and turbine building at HBRSEP. (Furthermore there is also some indication that one of the events that propagated to the turbine building also resulted in a loss of offsite power.) A specific analysis of the data and plant arrangement has therefore been made based on the following considerations and assumptions:

1) Although the Start Up Transformer and Auxiliary Transformer are in relatively close proximity to the turbine building, the transformer fires would not propagate via the mechanisms described in the FEDB due the slope of the grade and the drainage provided in the area.

2) A COMPBRN analysis has demonstrated that, providing the base of fires are confined to an area immediately surrounding the transformers (i.e. within the oil collection pits), fires in the Main Transformer will not impact the Start Up transformer or Turbine Building, including the DS conduit mentioned above (i.e. the incident flux will be less than 10 kW/m2). However, an unsuppressed fire associated in the Auxiliary or Start Up transformer may expose the DS conduit to a radiant heat flux of 23 kW/m2, which is sufficient to cause cable damage.

3) Reviewing the arrangement drawing, the phase 1 and 2 Main Transformers are located approximately 50 feet from the turbine building and at least that distance from the Start Up Transformer. Consequently, even if an associated fire was explosive in nature, it is judged unlikely that the resulting fire would propagate sufficiently to damage equipment in the turbine building, the DS conduit or the Start Up transformer. Both the loose rock fill and the transformer oil collection pits installed at HBRSEP would limit the spreading of burning oil. However, the phase 3 Main Transformer is closer to the Start Up transformer and overhead transmission lines, and an explosive fire is assumed to be capable of causing a loss of offsite power (LOSP).

4) For explosive fires in the phase 1 and 2 Main Transformer, and non explosive fires of any of the Main Transformers which are accompanied by a random failure of the deluge system, the lack of immediate suppression is assumed to lead to severe smoke ingress into the turbine building which would hamper the operators ability to use the DS system. However, there would be no loss of offsite power.

5) The Auxiliary and Start Up Transformers share the same oil collection pit and deluge system and are separated from the turbine building by 21'. An explosive fire in either of these transformers or non-explosive fire coupled with a random failure of the automatic deluge system is therefore assumed to result in a loss of offsite power and damage to the DS system.

6) A non-explosive fire in the Start Up Transformer with success of automatic suppression is assumed to lead to a loss of offsite power. However, this would be recoverable in the long term (8 hours) by back feeding through the main transformers once the turbine generator disconnect links were removed.

7) All other transformer fires with successful suppression result in a transient with no loss of mitigating systems.

In total there have been twenty five yard transformer fires in 1264.27 years of plant operation. Conservatively assuming that each plant has only five major transformers (as in the case of HBRSEP), and discounting the three fires which were obviously small and capable of being extinguished using portable extinguisher, the fire frequency per transformer year is estimated to be:

$$22/(1264.27 \times 5) = 3.48E\text{-}03$$

Of those fires, three are reported as being explosive in nature and therefore may be transported beyond the bounds of the oil collection pits at HBRSEP. The conditional probability of such an event, given a significant fire, is therefore estimated to be:

$$3/22 = 1.36E\text{-}01$$

The probability of automatic deluge system (AFSS) failure is 0.05 (see section 4.5)

Based on the above discussion the following scenarios and associated frequencies were determined:

Fire Scenario 1 Yard Transformer fires which result in a loss of offsite power and disable the operability of the DS system.

F(Offsite Power and DS system failure) =

F(Explosive Auxiliary Transformer fire) +
F(Explosive Start-Up Transformer Fire) +
F(Non-Explosive Auxiliary Transformer Fire and AFSS failure) +
F(Non-Explosive Start Up Transformer Fire and AFSS failure) =

3.48E-03 x 1.36E-01 + 3.48E-03 x 1.36E-01 +
3.48E-03 x (1 - 1.36E-01) x 5.0E-02 +
3.48E-03 x (1 - 1.36E-01) x 5.0E-02 =

1.25E-03 per year

Fire Scenario 2 Yard Transformer fire which results in LOSP and hampers operation of DS system due to smoke

F(Explosive Phase 3 Main Transformer fire) =

3.48E-03 x 1.36E-03 = 4.73E-04

Fire Scenario 3 Yard transformer fires which result in a loss of offsite power

F(Offsite Power failure) =

F(Non-Explosive, Suppressed, Start Up Transformer fire) =

3.48E-03 x (1- 1.36E-01) x (1 - 5.0E-02) =

2.86E-03 per year

Fire Scenario 4 Yard Transformer fires which result in a transient and hamper the operation of the DS system due to smoke.

F(DS system Failure) =

F(Explosive, phase 1 and 2 Main Transformer fires) +

F(Non Explosive Non Suppressed, phase 1, 2 or 3 Main Transformer fires) Unit 3 fires)

2 x 3.48E-03 x 1.36E-01
3 x 3.48E-03 x (1 - 1.36E-01) x 5.00E-02 +

1.40E-03 per year

<u>Fire Scenario 5</u> Yard transformer fires which only result in a transient

F(Transient only) =

F(Non Explosive, Suppressed, Main Transformer fires) +
F(Non Explosive, Suppressed, Auxiliary Transformer fires) =

3 x 3.48E-03 x (1 - 1.36E-01) x (1 - 5.00E-02)) +
3.48E-03 x (1 - 1.36E-01) x (1 - 5.00E-02) +

= 1.14E-02 per year

The fire compartment G/26 scenarios are summarized in Table 4.6-2h.

4.6.3 Risk Analysis of Main Control Room and Hagan Room

This section identifies the potential fire damage scenarios in the HBRSEP Main Control Room (Fire Compartment A/22). The resulting contribution of these fire damage states on core damage frequency is also evaluated. The general approach, which is similar to that adopted in NSAC 181, (EPRI, 1993) includes the following steps:

- Identify fire sources and evaluate associated frequencies.
- Evaluate extent of damage and likelihood of control room evacuation.
- Determine conditional accident sequence frequencies for each damage stage postulated by requantifying the internal events PSA model.

The analysis of the Hagan room (Fire Compartment A/21) was performed in a similar manner to the control room due to similar characteristics and close proximity to the control room. The Hagan Room fire scenarios were screened out. The results are presented in Table 4.6-1.

4.6.3.1 Description of Control Room and Associated Fire Protection

4.6.3.1.1 Description of Fire Compartment

The layout of the control room is shown in Figure 4.6-1.

The control room is situated on the third level of the Auxiliary Building at 254' elevation and is separated from other adjacent fire compartments by three-hour rated fire barriers. The dimensions of the Control Room fire compartment are approximately 44' x 40' x 11' (high) and consists of the following areas:

- Main Unit 2 Control Board Area,
- Locker Area (adjoining MCR),
- Conference Room (adjoining MCR),
- Security Office/Foyer,
- Shift Foreman's Office, and
- Rest Room/Kitchen.

(The control room arrangement has been modified since the IPEEE analysis of the control room was completed. However, the changes do not significantly impact the results of the study.)

Only the Reactor Turbine Generator Room (RTGB) cabinets A, B, C, D and E contain cable and equipment important to plant safety (Figure 4.6-1). These cabinets are adjoining with no intervening physical barriers. However, all other cabinets in the control room are separated from the RTGB cabinets by at least a two sheet steel cabinet walls with an intervening air space. During a plant walkdown the RTGB cabinets were noted as having ventilation grills and unsealed conduit penetrations in their tops.

A suspended ceiling is installed in the Control Room. The Control Room ceiling is a partial-coverage design, with no ceiling panels installed directly above the control panels to facilitate conduit entry into these panels. As such, the Control Room main area and the suspended ceiling space above communicate through a large opening in the ceiling, which runs approximately the full length and width of the RTGB. Two cable trays are installed above the RTGB bench boards, below the suspended ceiling, to accommodate routing of a minimal quantity of non-safety related cables. These cables are IEEE 383 and enter the RTGB bench boards via conduits which are up to 4" in diameter.

Electrical circuits are limited to those associated with lighting, instrumentation and control. Lighting circuits are 120 volt; instrumentation and control circuits are either 120 volts AC or 125 volt DC, or at the millivolt level. All 120- 125 -volt circuits are protected against overload and short circuits by either fuses or circuit breakers. Most lighting circuit wiring is within steel conduits or metal raceways built into lighting fixtures. All instrumentation and control wiring is inside the panels for the control boards in which the wires are terminated.

The control room HVAC system recirculates air at a rate of 4400 cfm. However, the purge rate is only 200 cfm. In the event of a fire, smoke removal is achieved using portable fans.

4.6.3.1.2 Fire Detection and Suppression Capability

Fire detection in the Control Room consist of four heat detectors and 12 ionization smoke detectors. Six of 12 ionization smoke detectors are located inside of the RTGB cabinets. The remaining detectors are mounted both above and below the suspended ceiling. These detectors will actuate the Fire Alarm Console in the Control Room. There is no automatic fire suppression system installed in the Control Room. Two fire hose stations and portable halon fire extinguisher are provided for fighting fires in their early stages.

In the event of a fire in this area, normal control of all accident mitigating systems may be lost and operators may be forced to evacuate the control room and utilize the Dedicated Shut down system (DS) capability. For the control room fire, the operators will follow Dedicated Shutdown Procedure DSP-002, "Hot Shutdown Using the Dedicated/Alternate Shutdown System, Rev. 13," to achieve and maintain safe shutdown.

4.6.3.1.3 Dedicated Shutdown Capability

Remote shutdown panels are located throughout the plant, normally in the same plant area as the equipment controlled by them. For a fire in the Control Room, the use of the Dedicated Shutdown Diesel is credited to supply power to the charging pump, component cooling water pump, service water pump and shutdown related instrumentation. The cabling associated with the dedicated shutdown system is routed independent of the Control Room and within dedicated conduits.

The Safe-shutdown Separation Analysis does not credit any equipment or controls in the Control Room; consequently, no specific separation criteria are imposed.

The possibility of fires within the Control Room is minimized by the use of flame retardant materials and segregation of possible sources of control board wiring fires.

4.6.3.2 Fire Hazard Review

The combined fire frequency in the Control Room and the Hagan Room is evaluated in the Quantitative Screening Analysis as 9.5×10^{-3}/yr.

The Control Room/Hagan Room fire frequency is based on twelve fires which actually occurred in control rooms, eleven of which were cabinet fires and one 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.

NSAC 181 (EPRI, 1993) indicates that the only significant control room fires are those which occur in cabinets and that transient fires do not pose a significant risk in the control room 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. The Fire Events Data Base indicates that plant wide components should not be applicable to the control room (EPRI, 1992b, p.3-17).

The fire hazard is distributed between the control room and the hagan room based on the relative number and size of cabinets in each compartment.

4.6.3.3 General Approach for Fire Evaluation of Control Room

4.6.3.3.1 General Philosophy For Control Room Evaluation

The general philosophy for fire evaluation of control room fires follows the approach suggested in NSAC 181 (EPRI, 1993) and EPRI RP 3385-01 (EPRI, 1994b). It is similar to that adopted in other areas but differs in two respects.

1. 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 alternate/dedicated shutdown system.

2. Detailed fire propagation analysis was not performed since there are no acceptable models for modeling propagation within cabinets. Instead it will be generally assumed that cabinet fires in the Control Room will not spread from the confines of the cabinet in which they originate to another cabinet, providing the cabinet has solid metal or fire resistant boundaries. This is supported by the results of the Sandia cabinet fire tests in which all tested fires self-extinguished, and by the reports of control room fires in the data base. Fire propagation to overhead cable trays is not significant since these trays contain a minimal amount of non safety related cable. Loss of all balance of plant systems, with the exception of offsite power and deep well pumps (which do not have any associated cables in these overhead trays) will be assumed for all control room scenarios.

The evaluation of control room fires requires the analyst to determine those cabinets or combination of connected cabinets in which enclosed fires might cause significant degradation of accident mitigating systems. In particular the location of fires which might result in a LOCA due to spurious PORV operation must be identified. Fire scenarios in such cabinets are evaluated individually. Fires in the remaining cabinets are evaluated as a group for their potential to cause the operators to evacuate the Control Room and shut down the plant using the Alternate/Dedicated Shutdown System.

The methods for frequency analysis, propagation analysis and suppression analysis are discussed below.

4.6.3.3.2 Fire Induced equipment failures for enclosed cabinet fires

The technique used to determine the effects of fire-induced equipment failures for enclosed cabinet fires includes the following primary steps:

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

2) Determine the likelihood of a fire occurring in a critical cabinet. The frequency of fire in each individual cabinet was evaluated as a function of; (i) the floor area occupied by the cabinet (ii) the total floor area occupied by all cabinets in the Control Room and the Hagan Room, and (iii) the fire compartment fire frequency. This approach attempts to account for the relative number of potential ignition sources in cabinets of different size.

   As an example, the frequency of fire for each of the cabinets containing SSD equipment are calculated as follows:

   Primary Plant Control Panel, RTGB Section "A"

   Fire Frequency = 9.5E-03 x (19.25/338)
   = 5.4E-04 per year

   Secondary Plant Control Panel, RTGB Section "D"

   Fire Frequency = 9.5E-03 x (20.125/338)
   = 5.6E-04 per year

   RTGB Section "C"

   Fire Frequency = 9.5E-03 x (12.25/338)
   = 3.4E-04 per year

3) Determine how severe a fire would have to be to fail the critical functions supported by a cabinet or combination of connected cabinets.

   The only ignition sources present within the electrical cabinets occur due to electrical overload resulting from a faulted component. If the damage can be confined to the site of the overload (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 PSA model. However, the design of the HBRSEP cabinets do not adhere to any design criteria for spatial separation of redundant components or wireways. A review of the front panel layout drawings coupled with an inspection of the panel interiors revealed that controls for redundant components may be located within a few inches of one another. Thus once ignition of a fire occurs it is not possible to discern which components served by the cabinet will be affected.

However, as discussed above, the HBRSEP RTGB cabinets are protected by in-cabinet smoke detectors placed on the cabinet ceilings. Sandia cabinet fire tests (EPRI, 1987c) (pertinent data summarized in Table 4.6-3) indicate a 5 minute time lapse between an in-cabinet fire detector detecting smoke, to the point where actual flames were observed. The tests referred to utilized vertical and benchboard cabinets loaded with unqualified cables (i.e. similar to those installed in the HBRSEP control room) which were ignited using an electrical ignition source (165W).

Despite the lack of physical separation of redundant components and wire ways within the HBRSEP control room cabinets, the potential for significant damage is highly unlikely prior to flame ignition. Thus an initial five minute time window for manual suppression is accounted for in modeling the risk from control room fires. No significant damage is postulated within this time period. During this phase, ignition may be prevented by de-energizing the faulted component and /or applying halon extinguisher located in the control room.

However, it will be assumed that all equipment in the cabinet where the fire originates will fail if the electrical fault progresses beyond the pre-ignition production stage to actual ignition.

In cases where cabinets are inter-connected and a solid intervening barrier does not exist (as in the case of the RTGB cabinets), inter cabinet fire propagation is assumed to occur unless the fire is extinguished prior to significant heat production. Again the evidence from the Sandia cabinet fire tests can be used to establish the time required for fire progression. These tests indicate that between 8 to 10 minutes elapsed between initiation of the in-cabinet smoke detector and significant heat generation (10-20 kW). The tests also indicate that once fire growth begins it may progress rapidly, as may the rise in cabinet air temperature. Thus, credit will be given for preventing fire propagation to inter-connected cabinets within the first 9 minutes after the in-cabinet detectors initiate. Once the fire has propagated to other cabinets all functions associated with those cabinets are assumed to fail.

(Historically all 12 control room fires have been extinguished without significant damage).

4) 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 in the pre-ignition phase or the pre-growth phase discussed above. The probability of non suppression of control room fires as a function of time is obtained using a model to interpret the control room fire durations in the EPRI data base. Such a model is developed in the EPRI Fire PRA Implementation guide (EPRI, 1994b, Appendix J). See Figure 4.6-2 which was extracted from the EPRI Fire PRA Guide. The probabilities of non suppression derived from the model (case 1 is recommended) are as follows:

p(non suppression within 5 minutes) = 1.2E-01

p(non suppression within 9 minutes) = 2.2E-02

4.6.3.3.3 Adverse Effects of Smoke

The technique used to incorporate the potential effects of smoke generated by a control room fire includes the following primary steps:

1) Determine the level of smoke which will impair the effectiveness of the operators

   The Sandia cabinet fire tests (EPRI, 1987c) 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 changes per hour. All of the actual control room fires in the 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 will be assumed that any fire is capable of producing sufficient smoke given it is allowed to continue burning for a sufficient period of time.

2) 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 (11016 ft^3) with ventilation rates about 14 room changes per hour. However, only one of these was electrically initiated (PCT5) and indicated visual obscuration within 13 minutes (time zero is the point at which smoke was first observed from the cabinet). Five tests were performed in larger enclosures (48000 ft^3), two of which were electrically initiated. In both cases the MCR control board was obscured within 20 minutes based on visual observations (EPRI, 1994b). The ventilation rate in one case was 1 room change per hour, and in the other, 8 room changes per hour.

   The size of the HBRSEP control room is approximately 14700 ft^3. The number of air changes per hour is less than 1. Therefore, the effective free volume of the HBRSEP control room is approximately 30% larger than the small test enclosure. However the ventilation rates are lower. For the large enclosure tests, the ventilation system did not appear to substantially affect the rate of smoke build up.

Based on the above discussion it is concluded that the rate of smoke build up in the HBRSEP control room will not be any faster than that observed for the small test enclosure. That is, the lower ventilation rate of the latter will be more than compensated for by the larger volume. Furthermore, in the event of a serious fire operators would immediately open doors to the outside patio area and utilize portable fans to exhaust the smoke. Thirteen minutes is considered to be a reasonable estimate for the time between in-cabinet smoke detectors actuating and the control board being obscured, for fires in the HBRSEP control room.

For non RTGB cabinets, which are not fitted with in-cabinet smoke detectors, an additional time lag of 3 minutes is assumed for local smoke detectors to alarm. Thus for fires in those cabinets, the time available for suppression would be: 13-3 = 10 minutes.

3) Determine probability of detection and suppression prior to smoke level reaching level of visual impairment.

Using the time reliability approach (see Figure 4.6-2) to determine probability of non-suppression within the time frames available prior to the control boards being obscured, the probabilities are determined to be as follows:

p(non suppression within 13 minutes) = 6.0E-03

p(non suppression within 10 minutes) = 1.6E-02

4.6.3.4 Fire Scenario Identification and Frequency Determination

Potential fire scenarios were identified by assuming that a potential fire would initiate in each of the control room cabinets (Table 4.6-4). The control indication functions lost because of this fire were then determined using control room board front view drawings. The component lists were then used to define and group control room fire scenarios and to determine the conditional core damage probability for a fire initiating event.

Based on the review, it is concluded that there are only three control room cabinets which contain instrumentation and controls required to support safety related safe shut down equipment. These are: RTGB A, RTGB C and RTGB D. Fire scenarios which impact these cabinets and/or cause control room evacuation are described below:

As discussed in above three phases of fire development are considered:

Pre-Ignition Phase: Period between in-cabinet smoke alarm initiation and flame production

Pre-Growth Phase: Period between initial flame production and flame spread.

Pre-Evacuation Phase: Period between initial flame spread and control board being obscured by smoke.

All fire scenarios originating in the RTGB cabinets which develop beyond the pre-ignition phase are conservatively assumed to result in a loss of the main feedwater/ condensate due to lack of information regarding the location of its numerous controls. Fires which are extinguished in the pre-ignition phase are not considered to be significant for reasons discussed in above.

Fire Originates in RTGB A Cabinet:

Immediate damage during Pre-Growth phase

Control of the following systems is assumed to be lost from the control room.

Charging
CCW
High Pressure Injection
Low Pressure Injection
EDGs and Fuel Oil Pumps
Pressurizer PORVs and Block Valves
RHR and SI valves
Main Feedwater/Condensate
RCS, RHR and RCP seal leak off temperature monitoring
Pressurizer pressure monitors
CCW flow to the RCPs
SI flow monitoring

Providing the fire is extinguished prior to flame spread, operators are expected to remain in the control room and utilize plant monitoring instrumentation provided on the RTGB, as well as controls for all AFW and valves which would be unaffected by the fire. For all other functions the operators would rely on equipment provided as part of Alternate Shutdown Method A (DS System), controlled from Auxiliary Panels outside of the control room. (Note: offsite power and emergency AC/DC power would not be affected directly by this fire).

Damage due to fire propagation to adjoining cabinets

RTGB A cabinet fires which are not extinguished during the pre-growth are assumed to spread to the RTGB C and D. Based on the extensive damage caused in RTGB D (see below), only Alternate Shutdown Method A is assumed to be available. No further propagation is relevant since the worst case control room fire damage is assumed.

Fires Originating in RTGB C cabinet

Immediate damage during pre-growth phase

Steam Generator Level monitoring in the control room is assumed to be failed.

Providing the fire is extinguished during the pre-growth stage operators are expected to remain in the control room and utilize SG level monitoring instrumentation provided on auxiliary panels.

Damage due to fire propagation to adjoining cabinets

RTGB C cabinet fires which are not extinguished during the pre-growth are assumed to spread to the RTGB A and D. Based on the extensive damage caused in RTGB D, only Alternate Shutdown Method A is assumed to be available. No further propagation is relevant since the worst case control room fire damage is assumed.

Fires Originating in RTGB D cabinet

Immediate damage during pre growth phase

Control of the following systems would be lost as a result of a fire in this cabinet:

Multiple BOP, 4kV, 480V breakers
AFW Motor and Turbine Driven Pumps
EDG HVAC
Service Water Pumps

Fires in this cabinet are capable of disabling control of all balance of plant and safety related AC power supplies as well as important plant monitoring instrumentation. Only equipment associated with Alternate Shutdown Method A is assumed to be available (as analyzed as during quantitative screening analysis with the exception that PORVs do not spuriously open).

The fire is assumed to result in a loss of offsite power initiating event.

Fire Propagation to Adjoining Cabinets

RTGB D cabinet fires which are not extinguished during the pre growth phase are assumed to spread to the RTGB A and C which may result in a spurious PORV opening. No further propagation is relevant since the worst case control room fire damage is assumed.

Fire Originating in RTGB B or E

Immediate damage during pre-growth phase

The control of safety related equipment or instrumentation required for shutdown will not be degraded as a result of localized fires in any of these cabinets. However, it is assumed main feedwater condensate is failed for reasons described above.

Fire Propagation to Adjoining Cabinets

RTGB B or E fires which are not extinguished at the incipient stage are assumed to propagate to RTGB Panels A and D. No further propagation is relevant since the worst case control room fire damage is assumed.

Fire Originating in All Other Control Room Cabinets

The following non-RTGB control room cabinets are separated from the RTGB cabinets, spatially or by a double sheet metal wall:

Incore Instrumentation Rack,
Radiation Monitoring Panel,
Nuclear Instrumentation Rack,
Post Accident Monitoring Panel,
Containment Fire Alarm Panel, and
Generator Supervisory Panel.

Furthermore, fires in such cabinets are not capable of degrading systems required for plant shutdown. Since inter-cabinet fire propagation is not an issue, due to separation, fires in these cabinets may only contribute to core damage in the event that they are not extinguished prior to a requirement for control room evacuation due to smoke. Operators are subsequently assumed to return to the control room after 60 minutes and regain control of all safety related equipment. Non safety equipment is assumed to be disabled due to the potential damage to overhead cable trays.

4.6.3.5 Control Room Fire Damage Event Tree

Based on the above discussion an event tree depicting the possible fire scenarios was developed in Figure 4.6-3. The event tree headings and their associated frequencies and probabilities are as follows:

Fire occurs in a control room or hagan room Cabinet	-Frequency = 9.5E-03/yr (see 4.6.3.2)

Fire Location Probability	-Based on cabinet floor area ratio
Fire Extinguished during pre-ignition phase	-Probability = 1.2E-01 (see 4.6.3.3)
Fire Extinguished during pre-growth phase	-Probability = 2.2E-02 (see 4.6.3.3)
Fire Extinguished prior to control board being obscured	-Probability = 6.0E-3 or 1.6E-02 (see section 4.6.3.3)

The damage states and associated probabilities are shown in Table 4.6-4

4.6.3.6 Control Room Fire Scenario PRA Evaluation

The quantification of the PRA model for control room fire sequences was performed in an identical manner to that described for other fire compartments, as described in section 4.2. The results of the quantification are discussed in section 4.6.4.

4.6.4 Results of Core Damage Frequency Analyses

Each of the fire scenarios discussed in sections 4.6.2 and 4.6.3 and summarized in Table 4.6-2a through h and Table 4.6-4 was evaluated to determine their contribution to core damage frequency. The method used to perform this analysis is summarized in section 4.6.1. The resulting core damage frequencies are presented in Table 4.6-5. The overall annual core damage frequency due to fires was 2.22E-04.

4.7 CONTAINMENT EVALUATION

The evaluation of containment performance following core damage resulting from fires requires the consideration of mechanisms which may lead to containment by-pass (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 modes which are significantly different from those found in the internal events IPE.

Containment bypass was evaluated in the IPE, and the two significant mechanisms identified were the interfacing system LOCA (large containment bypass) and the unisolated steam generator tube rupture (small containment bypass). Mechanical failure or spurious valve operation is the cause of both of these events. 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 4.1.2.3. No significant by-pass mechanisms were identified. In addition, accident sequences resulting from fire events will not cause more excessive RCS pressures than those addressed in the internal events accident sequence analysis, and thus no new induced steam generator tube rupture events. Therefore the containment bypass conclusions presented in the IPE are not altered by the fire analysis.

Containment isolation failure was also considered in the IPE for HBRSEP, the principal mechanism being associated with isolation valve failure to close. Containment isolation valves are air- operated and fail closed on loss of air supply or power to their solenoid valves. Thus fire damage is likely to lead to actuation of containment isolation. However, in the event of fire induced hot shorts which cause the valves to open or prevent closure, manual recovery actions may be taken. Since fire induced accidents do not lead to early vessel failures, several hours would be available to take such action. Furthermore, the most risk significant fire scenarios are located in compartments 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 penetration areas were screened out. Consequently degradation of containment penetration seals due to a fire is not a concern. Thus the likelihood of failure of containment isolation following fire scenarios is not significant and no new mechanisms were identified.

The availability of containment heat removal systems following core damage was explicitly modeled in the IPE event trees, and their status reflected in the core damage state binning process. Since the fire IPEEE analysis was mainly limited to evaluating the impact on systems defined within the Appendix R safe shutdown analysis, containment heat removal systems were generally not addressed. However, the most significant fire scenarios, namely those associated with the battery room, transformer yard, control room, and 480 v switchgear room result from a loss of power distribution in the short term which, although assumed to result in core damage, would be recoverable in the longer term, well prior to containment failure. For example, the dominant battery room fire scenario (16-1) leads to failure of all related DC power which in turn is assumed to cause a loss of all ac power (except DS) due to lack of breaker control power. However, in the longer term, within 1-2 hours, there is a high likelihood that the electrical breakers would be manually operated to restore power distribution and containment systems.

Based on MAAP analyses performed for the IPE, the containment pressure is not expected to increase to design limits as long as the RHR heat exchangers are available to remove heat.

This analyses assumes that energy is being added to containment through either a small LOCA or feed and bleed cooling. Thus, the fan coolers represent an additional heat removal mechanism but are not required for successful containment cooling as long as the RHR heat exchangers are present. Failure of this heat removal function will result in containment heatup and pressurization.

The pressurization is predicted to occur over many hours and would not result in an early, rapid containment overpressurization. It is concluded that containment fan coolers are not needed to ensure early containment integrity.

In conclusion, no new mechanisms associated with loss of containment heat removal were identified, and the frequency of non-recoverable containment heat removal systems was not significantly increased due to fires.

4.8 TREATMENT OF FIRE RISK SCOPING STUDY ISSUES

Under the NRC-sponsored Fire Protection Research Program, Sandia National Laboratories developed the Fire Risk Scoping Study, NUREG/CR-5088 (NRC,1989b), hereafter referred to as the "FRSS". The objectives of this study were to:

(1) Reassess certain fire risk scenarios, in light of the availability of enhanced fire event databases and improved fire modeling techniques,

(2) 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, and

(3) 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:

(1) Potential seismic/fire interactions,

(2) Fire barrier qualification issues,

(3) Manual fire fighting effectiveness,

(4) Total environment equipment survival,

(5) Potential control systems interactions, and

(6) Improved analytical codes.

The above issues, which were not addressed by earlier fire probabilistic risk assessments (PRA's), 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 section 7.0 of the FIVE Methodology (EPRI, 1992a).

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) has been accepted by the NRC for use in the IPEEE. Accordingly, this analysis is limited to a HBRSEP specific assessment of only the first five issues.

In addition, no effort with regards to the impact on smoke damage is required, since the NRC recognized the scarcity of information on the fragility of components and the limited conclusions which can be drawn in the PRA study.

4.8.1 Seismic Fire Interactions

4.8.1.1 Seismic Induced Fires

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

(1) Hydrogen piping and volume control tank,

(2) Diesel fuel oil piping, day tanks, and storage tanks,

(3) Turbine lubricating oil storage tank(s) and associated piping,

(4) Turbine generator (hydrogen envelope),

(5) Hydrogen seal oil unit and associated piping and tanks, and

(6) Auxiliary boilers and associated piping.

The specific location of these and similar hazards was identified through the fire walkdown phase of the IPEEE process and the seismic ruggedness of each identified component is addressed under the Seismic Walkdown phase of the IPEEE.

4.8.1.2 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.

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

- Diesel Generator "B" Room (Fire Area/Zone A/1; automatic high pressure CO_2),

- Diesel Generator "A" Room (Fire Area/Zone A/2; automatic high pressure CO_2),

- Component Cooling Pump Room (Fire Area/Zone C/5; partial wet pipe sprinkler),

- Auxiliary Building Ground Floor Hallway (Fire Area/Zone A/7; supervised air preaction sprinkler; The CO_2 cylinders for the DG systems are located in this area),

- North Cable Vault (Fire Area/Zone D/9; automatic high pressure CO_2),

- South Cable Vault (Fire Area/Zone E/10; automatic high pressure CO_2),

- Pipe Alley (Fire Area/Zone A/11; the CO_2 cylinders for the Cable Vaults are located in this area),

- Unit No. 2 Cable Spreading Room (Fire Area/Zone A/19; automatic 1301 Halon) **Note:** The Halon cylinders for this system are located in an area of the Turbine Building, and

- Emergency Switchgear Room & Electrical Equipment Area (Fire Area/Zone A/20; automatic 1301 Halon) **Note:** The Halon cylinders for this system are located in an area of the Turbine Building.

As stated in CP&L's response to BTP APCSB 9.5-1, Item A.5, "Fire Suppression Systems" (Appendix 9.5.1B), a detailed analysis for inadvertent actuation through any open sprinkler head was not considered necessary since the flow would be bounded by the pipe rupture analysis. The following description summarizes the effects of various postulated fire water system pipe ruptures on safety-related equipment from a detailed analysis provided in a CP&L letter dated June 12, 1980 and accepted by the NRC in the Safety Evaluation Report supplement dated December 8, 1980.

- The pipe rupture analysis was performed by considering pipe breaks in the Auxiliary Building, which included a four-inch pipe break in the pipe tunnel (Pipe Alley, fire zone 11, elevation 226) and the hallway near MCC No. 5 (fire zone 7, elevation 226).

- The analyses show that the floor drain system will prevent flooding of electrical safety-related equipment on the second floor. The line break in the hallway near MCC No. 5 was found to have the potential to damage safety-related equipment by direct impingement, which was resolved by the installation of a spray shield to protect MCC No. 5. All other breaks could cause equipment damage on the 226 elevation by flooding, but time was determined to be available for corrective action by the operator to terminate water flow by closing the appropriate isolation valve.

- In addition, subsequent to this analysis, an Abnormal Operating Procedure, AOP-032 has been issued to mitigate the consequences of a break in the fire protection system. This procedure was initiated as a result of the HBRSEP IPE.

The issues raised in IN-83-41 have therefore been addressed at HBRSEP and dispositioned satisfactorily.

4.8.1.3 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 being dislodged 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.

Inadvertent operation of the CO_2 or Halon systems leading to failure of the diesel generators for example, is not probable based on the walkdown of the systems, which are judged to be seismically rugged.

4.8.2 Fire Barrier Qualifications

Fire Barrier elements (barriers, doors, penetration seals and dampers) were reviewed in accordance with the six checklist questions identified in FIVE. All checklist questions were answered affirmatively, indicating that the fire barrier elements are being maintained and that the HBRSEP Fire Protection Program provides adequate control measures consistent with the objectives of the FIVE methodology.

4.8.2.1 Fire Barriers

HBRSEP procedure, OMM-002, section 5.9, provides a general discussion with regards to barriers and refers to other procedures regarding controls and compensatory actions. The following procedures demonstrate that the installation and maintenance of fire barriers/seals (including doors and dampers) at HBRSEP are controlled and consistent with industry practices. These procedures are applicable to all Appendix R fire areas and BTP 9.5-1, Appendix A fire zones.

- FP-012 - section 5.9 establishes the compensatory actions when penetration fire barriers are declared inoperable.

- FP-014 - this procedure provides instructions for documenting operability, maintenance, and modifications of fire barrier penetrations. It also invokes the appropriate procedure (OST) following repairs or maintenance of any penetration seal, fire door and damper.

- OST-623 - this procedure inspects penetration seals only.

- OST-624 - this procedure inspects penetration dampers only.

- OST-625 - this procedure inspects penetration doors only.

- CM-620 - this procedure provides instructions for the installation of fire doors in masonry walls.

- CM-621 - this procedure provides instruction for establishing and repairing fire barriers with 3 hour ratings.

4.8.2.2 Fire Doors

As discussed above, fire doors are included in both FP-012 and FP-014 with regards to controls and compensatory actions. Fire doors are surveilled semi-annually in accordance with OST-625.

4.8.2.3 Penetration Seal Assemblies

<u>Penetration Seal Inspection and Surveillance Program</u>

The surveillance of fire barrier penetration seals is addressed above.

<u>Evaluation and Implementation of Applicable NRC IE Notices</u>

The FIVE methodology identifies three NRC Information Notices which have specific applicability to fire barrier penetration seals:

(1) 88-04, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals."

(2) 88-04 Supplement 1, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals."

(3) 88-56, Potential Problems With Silicone Foam Fire Barrier Penetration Seals."

On-site Nuclear Safety OEF Evaluation packages have been assembled for each of the above NRC I&E Notices(IEN) as discussed separately below:

- **IN-88-04** - A memo from H.P. Hines to E.M. Harris, Jr.(NED-R-3098, page 4 lists various documents which support CP&L's position that HBRSEP complies with the NRC guidance in the IN.

- **IN-88-04, Supplement 1** - This IN alerts all holders of operating licenses to problems caused by potential misapplication of silicone foam material, specifically when used around diesel generator exhaust pipes. On May 27, 1985, a similar fire occurred at HBRSEP on a temporary seal at a wall penetration of the Diesel Generator exhaust pipe. An on-site investigation by Dow Corning was conducted which determined that the fire was cause by a combination of the ignition of the "paper" backing to the outer metal jacketing around the calcium silicate insulation and the direct contact of the silicone foam to the exhaust pipe due to the removal of the insulation. As noted in the On-site Nuclear Safety OEF Evaluation package, subsequent to the fire described above, a specific seal design was provided for hot-pipes (i.e., CM-621).

- **IN-88-056** - As discussed in the On-site Nuclear Safety OEF Evaluation package, an LER (88-018) was issued with regards to problems found with cable tray penetrations, in which a deficiency was found in the installation procedure (CM-621) with regards to potential voids with covered cable tray penetrations. Corrective actions in the form of a procedure revision had been made.

In summary, the above three IN's have been recognized by CP&L and the associated concerns were addressed by appropriate revisions to plant procedures, where appropriate.

4.8.2.4 Fire Dampers

Fire Damper Inspection and Maintenance Program

As discussed above fire dampers are included in both FP-012 and FP-014 with regards to controls and compensatory actions. The dampers are surveilled during refueling (18 month interval) in accordance with OST-624.

Evaluation and Implementation of Applicable NRC Information Notices

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

(1) 83-69, "Improperly Installed Fire Dampers at Nuclear Power Plants."

(2) 89-52, "Potential Fire Damper Operational Problems."

An On-site Nuclear Safety OEF Evaluation package has been assembled for NRC Information Notices(IN) 89-52, however this program was not in use at the time IN 83-69 was issued and, as discussed below, resolution of concerns addressed by the latter IN has been verified by reference to other plant documents:

- **IN-83-69** - During 1985 and 1986 plant walkdowns were performed to validate the correct installation of dampers in both fire area and fire zone boundaries to demonstrate compliance to 10CFR50, Appendix R and BTP 9.5-1, Appendix A requirements. Plant Modifications M-850A/B[40] were specifically developed to install new dampers, as required.

- **IN-89-52** - This NRC notice alerts holders of operating licenses to potential problems affecting the closing reliability of curtain-type fire dampers under air flow conditions.

The notice cites a 10CRF21 notification to the NRC involving Ruskin Manufacturing and Northern States Power Company, but acknowledges that concern need not be limited to Ruskin, all stations use curtain-type fire dampers of similar design. An HBRSEP plant modification, M-650, provided specific upgrades to install negator springs with a stiffener as a method to address closure under air flow. The acceptance testing for M-650, specifically required the dampers to be tested **against "design air flow conditions"**. In addition, as with this modification, other upgrades (i.e., M-850A/B) also required testing of the dampers against design air flows and references procedure OST-624. Accordingly, the scope of the modification met the intent of IN-89-52.

Subsequent to these upgrades, it was decided not to perform routine damper closing in accordance with OST-624, due to personnel safety concerns associated with the very strong closing force exerted by the negator springs. If however any maintenance was performed on any damper, then a functional test would be conducted.

4.8.3 Manual Fire Fighting Effectiveness

This issue focused on the adequacy of training and preparedness of the plant fire brigade and 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 fire PRA.

Manual fire fighting effectiveness was reviewed in accordance with the 16 checklist questions identified in FIVE. All checklist questions were answered affirmatively, indicating that the plant's fire training and preparedness is consistent with standard expectations. In summary the programmatic criteria for the HBRSEP fire brigade training, practice/drills, and equipment complement are consistent with the guidelines of the FIVE methodology.

4.8.3.1 Reporting Fires

<u>Orientation of Plant Personnel to the Use of Portable Fire Extinguishers</u>

As described in OMM-002, section 5.10, all plant personnel granted unescorted access receive training as part of the general orientation and general employee training/retraining. As part of the classroom material, step-by-step instructions are provided with regards to the proper use of portable extinguishers. In addition, all personnel serving as fire watches are required to complete a 2.5 to 3 hour course where additional extinguisher training is provided. The employee training material stipulates that the control room be notified first, then attempt to extinguish the fire, if possible, using the appropriate extinguisher.

<u>Availability of Portable Extinguisher Throughout the Plant</u>

In response to BTP APCSB 9.5-1, section E.6, (section 9.5.1B), the distribution of portable fire extinguisher throughout HBRSEP is consistent with NFPA 10. OST-610 requires a visual inspection of extinguisher once per month for general plant areas. Fire extinguishers are distributed throughout the plant, as listed in the HBRSEP Fire Hazards Analysis, Appendix 9.5.1A. These consist of portable Halon 1211, ABC dry chemical, AFFF foam extinguisher, and large 150-lb. wheeled Halon 1211 extinguisher. Additional equipment is available in the Fire Equipment Building, including a 95-gpm foam cart and 120-gpm portable foam equipment.

Plant Procedure for Reporting Fires

As stated in OMM-002, section 5.2 and reiterated in the General Employee Training material any person discovering a fire is responsible for promptly notifying the Unit 2 control room. Upon receipt of notice of a fire or annunciation of a fire alarm, the Shift Supervisor shall implement FP-001[15]. FP-001 provides the appropriate action for alerting plant personnel (i.e., fire brigade, security and radiation control personnel).

Communication System to Allow Contact With the Control Room

As stated in the response to BTP APCSB 9.5-1, section D.5 (section 9.5.1B), sound-powered and amplified telephones are provided at selected locations throughout the plant, which are also identified on the appropriate pre-plans, OMM-003. In addition, the Operations/Fire Protection radio system utilizes 2-channel portables. One channel has a repeater powered from the Dedicated Shutdown (DS) supply and the second channel provides a talk-around feature which does not require a repeater.

4.8.3.2 Fire Brigade

Size of Fire Brigade

Per OMM-002, section 3.0, the on-duty Shift Supervisor is responsible for ensuring at least five Fire Brigade members are available in accordance with OMM-001.

Brigade Members Knowledgeable in Plant Systems and Operations

In accordance with OMM-001, at least five operations personnel shall be available each shift in support of the fire brigade. In addition, as defined in section 4.0 of OMM-002, the Fire Brigade Team Leader is normally the Off Control Operator or the Work Control Senior Control Operator who are qualified as a Team Leader. Furthermore, any licensed Operator can serve as Team Leader, if qualified. The on-duty Shift Supervisor or the Emergency Coordinator is also available to provide support to the Fire Brigade Team Leader, if necessary.

Annual Physical Examinations for Brigade Members

In accordance with TPP-219, each brigade member must satisfactorily complete an annual physical examination, including a respiratory examination. This training meets or exceeds the requirements of NUREG 0041 and the CP&L Corporate Respiratory Protection Program.

Minimum Equipment Provided/Available to Fire Brigade

In accordance with FP-008 and OST-639, the following fire brigade equipment is stored on site:

(1) Twenty sets of turnout gear, including coats, pants, helmets, boots, and gloves, (A minimum of five must be maintained at all times.)

(2) Nine portable radios,

(3) Five lanterns,

(4) Two smoke ejectors with 4 flexible ducts(tunnels),

(5) Portable fire extinguishers - throughout the station,

(6) One portable generator,

(7) Western foam cart with two extra 5-gallon containers of AFFF foam,

(8) Two angus foam units,

(9) One breathing air compressor with 4 charged cylinders, and

(10) A minimum of 10 SCBA's and 20 spare cylinders.

The equipment available to the fire brigade, therefore, is consistent with the FRSS criteria, and the equipment complement is verified periodically (monthly) under OST-639.

4.8.4 3 Fire Brigade Training

Initial Classroom Instruction Program

The Fire Brigade Team Members Training Course, as outlined in Attachment 6.3 of OMM-002, is required for all brigade leaders and members. This initial course shall be completed in its entirety prior to assignment to the Fire Brigade. Quarterly training sessions are held such that reviews, updates, modifications to fire protection systems, lessons learned from fire drills, changes in fire preplans, or advanced training on major topics of the Fire Fighting Course can be repeated every two years. The following are topics of this initial training which is consistent with the FIVE evaluation methodology.

(1) Duties and responsibilities - state the responsibilities of each member.

(2) Chemistry and extinguishment of fire - describes the theory of fire and how fire may be extinguished.

(3) Classification and uses of portable fire extinguisher - recognize each class of extinguisher and describe the condition under which each is effective and the use the extinguisher to extinguish a fire.

(4) Uses of water and foam in fire extinguishment - describe the advantages of water and foam and how each would be used to extinguish Class A, B and C fires. Also the use of foam equipment is practiced.

(5) Fire hydrants, hoses, and hose cabinet and houses - recognize each and state their primary use.

(6) Fire protection systems and building layout - describe each type of suppression system and demonstrate ability to use fire preplans with respect to access routes and local shutoff valves.

(7) Flammable liquids and gases - describe the fire hazards and fire fighting tactics.

(8) Smoke and toxic gases - describe hazards and fire fighting tactics.

(9) Communications - demonstrate the use of the page system, telephone, and portable radio.

(10) Lighting - demonstrate the use of the fire preplans for identification of normal and emergency lighting.

(11) Use of fire preplans for fire fighting and ventilation controls - demonstrate use of the fire preplans to identify hazards and of removing smoke.

(12) Hands on use of: SCBA, fire water supply systems (hydrants, hoses,etc.) and ladders - demonstrate ability to properly use SCBA under strenuous condition encountered in fire-fighting and to use fire hoses and ladders.

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

Practice

The fire brigade hands-on training program is included as part of the Fire Brigade team Member Training Course as describe in section 4.8.4.3 above. In addition, as described in TPP-219, one drill per year is conducted at a training facility where live fire-fighting is performed.

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

Drills

The fire brigade drills, as described in TPP-219, provide the following elements, consistent with the FIVE evaluation methodology:

(1) Drills are normally performed in the plant so that the brigade can practice as a team. Each Fire Brigade Team member should participate in every drill for that team and shall participate in at least two drills per year.

(2) Each shift Fire Brigade Team shall be drilled at least once per calendar quarter.

(3) Each brigade shift participates in at least one unannounced drill per year.

(4) At least one drill per year is performed on a backshift for each shift fire brigade.

(5) Drills are preplanned to establish training objectives as and post-critique by Fire Protection Staff or designee to determine how well the training objectives have been met.

(6) At 3 year intervals, drills are critiqued by a qualified outside individual. During this audit, the drill(s) are likely to be unannounced, as all HBRSEP Fire Brigade drills are typically unannounced.

(7) OMM-003, Fire Protection Pre-Plans/Unit No.2, has been developed, which includes safety and non-safety related areas.

(8) As noted in section 4.3.1, the preplans are an integral part of the Fire Brigade Training Course.

(9) The equipment available to the fire brigade is consistent with the FRSS criteria, which is maintained in accordance with OST-639.

Records

FP-011 delineates the methods to be used in documenting fire protection training and drills. The Fire Protection Staff annually reviews the records to ensure that all program training goals are being achieved for each fire brigade member.

4.8.4 Total Environment Equipment Survival

The issue is concerned with the potential effects on plant equipment by combustion products, spurious or inadvertent fire suppression system activation, and on operator action effectiveness given a fire at the plant. A summary discussion of each is given below.

4.8.4.1 Potential adverse effects on plant equipment by combustion products

The FIVE/FRSS methodology does not provide criteria for assessment of the potential non-thermal effects of 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. Regardless of this the handling of smoke is considered and is incorporated into the training (see section 4.3.1, items (8) and (11)).

4.8.4.2 Spurious or inadvertent fire suppression activity

The potential effects of spurious/inadvertent suppression system actuation are enveloped by Section 4.8.1 of this analysis.

4.8.4.3 Operator Action Effectiveness

Post-Fire Safe-Shutdown Procedures

Dedicated Shutdown Procedure, DSP-001, determines whether conditions exists that warrant the use of the DSP's and to provide guidance as to which specific DSP should be implemented. Each DSP provides instructions for mitigation of a fire located in a specific plant area. In addition, if a fire were to exist in the RTGB or an unknown control room location, DSP-001 directs personnel not responding to the fire to proceed to the Fire Equipment Building and implement Abnormal Operating Procedure, AOP-004.

Operator Training in Post-Fire Safe-Shutdown Procedures

Periodic operator training is conducted in accordance with PLP-009 for the HBRSEP operations unit.

Operator Reentry Into Affected Fire Area: Respiratory Protection

The DSP's do not specifically address operator effectiveness in smoke-filled areas, but the following apply:

(1) SCBA equipment is provided in the control room and at strategic locations throughout the plant.

(2) Fixed, battery-backed emergency lighting units are installed along post-fire shutdown access/egress routes and at equipment operating stations.

4.8.5 Control System Interactions

As stated in section 4.8.4.3 above, DSP-001 directs personnel to implement AOP-004, which further implements DSP-002. Since each DSP was developed from information obtained from the HBRSEP Safe Shutdown Analysis, specific steps are provided to ensure circuits are physically independent of, or can be isolated from, the control room (or other Appendix R area) due to fire.

The HBRSEP alternative shutdown features provide independent remote control and monitoring features. Therefore, the design of the HBRSEP alternative shutdown capabilities is generally immune to the effects of "control systems interactions" as defined within the scope of the FIVE methodology.

4.8.6 Conclusions

The results of the topical assessments performed under the FIVE Fire Risk Scoping Study indicate that the following FRSS issues have been adequately addressed by CP&L for HBRSEP, and the applicable aspects of the HBRSEP Fire Protection Program therefore are <u>in conformance</u> with the intent of the FRSS guidelines, as tabulated in Attachment 10.5 of the FIVE Methodology:

- Seismic/Fire Interactions
- Fire Barrier Qualifications
- Manual Firefighting Effectiveness
- Total Environment Equipment Survival
- Control Systems Interactions

4.9 USI A45

As discussed in the IPE submittal (CP&L, 1992), loss of decay heat removal is inherently considered in a PRA evaluation of core damage frequency. In the fire analysis, significant fire areas were identified on the basis of their contribution to core damage frequency. The significance of an area is not tied to the decay heat removal issue per se; however, resolution of issues arising from an identification of these areas as significant will resolve any issues related to USI A-45.

4.10 REFERENCES

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TABLE 4.1-1a - SAFE SHUTDOWN SYSTEM/FUNCTION VS. FIRE AREA MATRIX																					
Fire Area Number: "A"		Fire Area Description: Aux. Bldg. Fire Zones 1-3, 6-8 & 11-23																			
Fire Compartment/Zones		Safe Shutdown Systems/Functions																			
Zone Number	Description	CVC Alt A	CVC/SI Alt B	PZR PORV	AFW/MS Alt A	AFW Alt B	SG PORV	RHR *	PPM	RCT Alt A	RCT Alt B	NIS	CST LT	CCW Alt A	CCW Alt B	ACP Alt A	ACP Alt B	DCP Alt A	DCP Alt B	SW Alt A	SW Alt B
1	Diesel Generator "B" Rm																X				
2	Diesel Generator "A" Rm																X				
3	Safety Injection Pump Rm		1					R						P	X		1				X
6	Aux. Feed Pump Rm					X			1												X
7	Aux. Bldg. Hallway (Ground Floor)	L,P	X		P	X	L	R						L	X	P	X	P	X	L	X
8	Boron Injection Tank Rm		1																		
11	Pipe Alley	P	X		P	X		R						P	X		X			L, P	X
12	Waste Holdup Tk/RHR HX	P	1			X		R						L	X					L, P	X
13	Chem. Stor./Boric Acid Batch Tk																			L	1
14	Solid Waste Handling Room																				
15	Aux. Bldg. Second Level	P*	X		P			R									X	P			
16	Battery Rm				P								A				X	P	X		
17	HVAC Equip. Rm for Control Rm																				
18	Unit 1 Cable Spread Rm																P				
19	Unit 2 Cable Spread Rm	L,P	X	P	P	X	L	R	A		X	A	A	L,P	X		X	P	X	L,P	X
20	Elec. Swgr./Elec. Equip. Rm	L,P	X	P	P	X	L	R	A		X	A		L,P	X	P	X	P	X	L,P	X
21	Rod Control Rm	L,P	X	P	P	X		R	A		X	A								L,P	1
22	Control Rm	L,P	X	P	P	X	L	R	A		X	A	A	L,P	X		X			L,P	X
23	Hagan Rm	P*	P*	P*				R	A		X						X				

* - Cold Shutdown Only
1 - At least one path of function not in area/compartment
C - Compliance using 1 hr barrier with auto. suppression or 20' separation no combustible with auto. suppression
E - Exemption
L - Local Manual Action

X - Damage
R - Repair, or replace damage equipment/cable
B - Barrier of 3 hour fire rating
P - Post Fire Manual Repositioning/De-Energization
A - Alternate Instrument in another Fire Area

TABLE 4.1-1b - SAFE SHUTDOWN SYSTEM/FUNCTION VS. FIRE AREA MATRIX																					
Fire Area Number: "B"		Fire Area Description: Fire Zone 4; Charging Pump, VCT & Non-Regen Rm																			
Fire Compartment/Zones		Safe Shutdown Systems/Functions																			
Zone Number	Description	CVC Alt A	CVC/SI Alt B	PZR PORV	AFW/MS Alt A	AFW Alt B	SG PORV	RHR *	PPM	RCT Alt A	RCT Alt B	NIS	CST LT	CCW Alt A	CCW Alt B	ACP Alt A	ACP Alt B	DCP Alt A	DCP Alt B	SW Alt A	SW Alt B
4		X	1,P						A	X		A		X						X	1,P

TABLE 4.1-1c - SAFE SHUTDOWN SYSTEM/FUNCTION VS. FIRE AREA MATRIX																					
Fire Area Number: "C"		Fire Area Description: Fire Zone 5; Component Cooling Pump Rm																			
Fire Compartment/Zones		Safe Shutdown Systems/Functions																			
Zone Number	Description	CVC Alt A	CVC/SI Alt B	PZR PORV	AFW/MS Alt A	AFW Alt B	SG PORV	RHR *	PPM	RCT Alt A	RCT Alt B	NIS	CST LT	CCW Alt A	CCW Alt B	ACP Alt A	ACP Alt B	DCP Alt A	DCP Alt B	SW Alt A	SW Alt B
5		X	1		X			1						X	E					P	P

TABLE 4.1-1d - SAFE SHUTDOWN SYSTEM/FUNCTION VS. FIRE AREA MATRIX

Fire Area Number: "D"		Fire Area Description: Fire Zone 9; North Cable Vault																			
Fire Compartment/Zones		Safe Shutdown Systems/Functions																			
Zone Number	Description	CVC Alt A	CVC/SI Alt B	PZR PORV	AFW/MS Alt A	AFW Alt B	SG PORV	RHR *	PPM	RCT Alt A	RCT Alt B	NIS	CST LT	CCW Alt A	CCW Alt B	ACP Alt A	ACP Alt B	DCP Alt A	DCP Alt B	SW Alt A	SW Alt B
9			P	*	X	P		R	A	X	A	A									

* - Cold Shutdown Only
1 - At least one path of function not in area/compartment
C - Compliance using 1 hr barrier with auto. suppression or 20' separation no combustible with auto. suppression
E - Exemption
L - Local Manual Action

X - Damage
R - Repair, or replace damage equipment/cable
B - Barrier of 3 hour fire rating
P - Post Fire Manual Repositioning/De-Energization
A - Alternate Instrument in another Fire Area

TABLE 4.1-1e - SAFE SHUTDOWN SYSTEM/FUNCTION VS. FIRE AREA MATRIX

Fire Area Number: "E"		Fire Area Description: Fire Zone 10; South Cable Vault																			
Fire Compartment/Zones		Safe Shutdown Systems/Functions																			
Zone Number	Description	CVC Alt A	CVC/SI Alt B	PZR PORV	AFW/MS Alt A	AFW Alt B	SG PORV	RHR *	PPM	RCT Alt A	RCT Alt B	NIS	CST LT	CCW Alt A	CCW Alt B	ACP Alt A	ACP Alt B	DCP Alt A	DCP Alt B	SW Alt A	SW Alt B
10			P	*	X	P,L		R	A	X	A	A								X	

TABLE 4.1-1f - SAFE SHUTDOWN SYSTEM/FUNCTION VS. FIRE AREA MATRIX

Fire Area Number: "F"		Fire Area Description: Fire Zone 24; Containment																					
Fire Compartment/Zones		Safe Shutdown Systems/Functions																					
Zone Number	Description	CVC Alt A	CVC/SI Alt B	PZR PORV	AFW/MS Alt A	AFW Alt B	SG PORV	RHR *	PPM	RCT Alt A	RCT Alt B	NIS	CST LT	CCW Alt A	CCW Alt B	ACP Alt A	ACP Alt B	DCP Alt A	DCP Alt B	SW Alt A	SW Alt B
24			P	*				R	C	C	C	C									

* - Cold Shutdown Only
1 - At least one path of function not in area/compartment
C - Compliance using 1 hr barrier with auto. suppression or 20' separation no combustible with auto. suppression
E - Exemption
L - Local Manual Action

X - Damage
R - Repair, or replace damage equipment/cable
B - Barrier of 3 hour fire rating
P - Post Fire Manual Repositioning/De-Energization
A - Alternate Instrument in another Fire Area

TABLE 4.1-1g - SAFE SHUTDOWN SYSTEM/FUNCTION VS. FIRE AREA MATRIX																					
Fire Area Number: "G"		Fire Area Description: Fire Zones 25, 26, 28-33																			
Fire Compartment/Zones		Safe Shutdown Systems/Functions																			
Zone Number	Description	CVC Alt A	CVC/SI Alt B	PZR PORV	AFW/MS Alt A	AFW Alt B	SG PORV	RHR *	PPM	RCT Alt A	RCT Alt B	NIS	CST LT	CCW Alt A	CCW Alt B	ACP Alt A	ACP Alt B	DCP Alt A	DCP Alt B	SW Alt A	SW Alt B
25	Turbine Building (General)	X	L,P	*	X	1,P	P,L	R	A	X	A	A	A	X	1		X		X	X	E,P
25	Turbine Building (DS Diesel)															X		X			
26	Yard Transformers																				
28	New & Spent Fuel Storage Hot Shop																				
29	Service Wtr Pump Area																			X	E,P
30	Diesel Oil Storage Tank																X				
31	Refueling Wtr Storage Tank																				
32	Primary Wtr Storage Tank																				
33	Condensate Storage Tank																				

* - Cold Shutdown Only
1 - At least one path of function not in area/compartment
C - Compliance using 1 hr barrier with auto. suppression or 20' separation no combustible with auto. suppression
E - Exemption
L - Local Manual Action

X - Damage
R - Repair, or replace damage equipment/cable
B - Barrier of 3 hour fire rating
P - Post Fire Manual Repositioning/De-Energization
A - Alternate Instrument in another Fire Area

TABLE 4.1-1h - SAFE SHUTDOWN SYSTEM/FUNCTION VS. FIRE AREA MATRIX

Fire Area Number: "H"		Fire Area Description: Fire Zone 27; RHR Pit																			
Fire Compartment/Zones		Safe Shutdown Systems/Functions																			
Zone Number	Description	CVC Alt A	CVC/SI Alt B	PZR PORV	AFW/MS Alt A	AFW Alt B	SG PORV	RHR *	PPM	RCT Alt A	RCT Alt B	NIS	CST LT	CCW Alt A	CCW Alt B	ACP Alt A	ACP Alt B	DCP Alt A	DCP Alt B	SW Alt A	SW Alt B
27			*P					R													

* - Cold Shutdown Only
1 - At least one path of function not in area/compartment
C - Compliance using 1 hr barrier with auto. suppression or 20' separation no combustible with auto. suppression
E - Exemption
L - Local Manual Action

X - Damage
R - Repair, or replace damage equipment/cable
B - Barrier of 3 hour fire rating
P - Post Fire Manual Repositioning/De-Energization
A - Alternate Instrument in another Fire Area

CVC Charging System
SI Safety Injection System
PZR Pressurizer
PORV Power-Operated Relief Valves
AFW Auxiliary Feed Water System
MS Main Steam
SG Steam Generator
RHR Residual Heat Removal System
PM Plant Monitoring
PPM Primary Plant Monitoring (PZR, SG)

RCT Reactor Coolant Temperature
NIS Nuclear Instrumentation System
CST Condensate Storage Tank
LT Level Transmitter
CCW Component Cooling Water
ACP AC Power System
DCP DC Power System
SW Service Water
Alt A Safe Shutdown Alternate "A" Method
Alt B Safe Shutdown Alternate "B" Method

Table 4.1-2 Summary of Hi-Lo Boundary Interfaces				
System Interface	**Boundary Components**	**Fire Zones**	**Pre / Post Fire Procedure**	**Conclusions of EPM Study Review**
Residual Heat Removal Hot Leg Suction	MOV RHR 750 MOV RHR 751	A7,E10,A15,A19,A20,A21,A22 D9,A19,A21,A22,G25	Both valves normally closed. The RHR 751 valve power breaker at the MCC is placed in the open position prior to power operation.	Spurious operation of the RHR 751 valve would require a combination of three 480V ac hot shorts to the power cable and the absence of grounding. This event is not considered to be credible.
CVCS Letdown (1"dia) (2" dia)	CVC 200A CVC 200B CVC 200C LCV 460A LCV 460B	B4,E10,A12,A19,A20,A21,A22 B4,A7,D9,A11,A12,A19,A20,A21, A22,G25 B4,D9,A12,A20,A21,A22,A19 E10,A19,A20,A21,A22 E10,A19,A20,A21,A22	Denenergize 125 V dc panels DC and GC post fire in any fire area A zones	Spurious opening of let down isolation valves LCV 460A and 460B (both air operated fail closed valves) in conjunction with spurious opening of one of the orifice isolation valves CVC 200A, 200B or 200C (also air operated fail closed valves) would align a potential hi-lo interface path. Based on the EPM studies this would require four external hot short and the absence of a ground fault.
CVCS Excess Letdown (3/4"dia)	RC 387 HCV 137	E10,A19,A20,A21,A22 D9,A19,A20,A21,A22	De-energize dc Auxiliary panel GC post fire in area	The excess letdown line is isolated by a normally closed air operated valve CVC 387 which fails closed on loss of air or power. In series with the isolation valve is the flow control valve HCV-137 which is modulated by a 4-20 ma signal. A minimum of three external hot shorts is required to align the path.
Pressurizer PORV (4")	RC 535 RC 536 PCV 456 PCV 455C	D9,A19,A20,A21,A22 D9,A19,A20,A21,A22 D9,E10,A19,A21,A22 E10,A19,A20,A21,A22	De-energize dc Auxiliary panels DC and GC post fire in Area A, D and G.	Both PORVs fail closed on loss of instrument air or dc power to their solenoid valves. A minimum of two external hot shorts is required to spuriously open a PORV.

Table 4.1-2 Summary of Hi-Lo Boundary Interfaces				
System Interface	**Boundary Components**	**Fire Zones**	**Pre / Post Fire Procedure**	**Conclusions of EPM Study Review**
Reactor Vessel and Pressurizer Head Vents (1"dia)	RC 567 RC 568 RC 569 RC 570 RC 571 RC 572	E10,A19,A20,A21,A22, D9,A19,A20,A21,A22 E10,A19,A20,A21,A22 D9,,A19,A20,A21,A22 E10,A19,A20,A21,A22 D9,A19,A20,A21,A22	Disconnect power and indication cables to all four vent valves prior to power operation. De-energize dc Auxiliary panels DC and GC post fire in Area A,D and E	The two parrallel pressurizer vent paths are isolated by normally closed solenoid valves RC 569 and RC 570 which fail closed on loss of dc power. The two parrallel reactor vent paths are isolated by normally closed solenoid valves RC 567 and RC 568 which fail closed on loss of dc power. These vent paths combine into a single header which in turn may be isolated by two solenoid valves RC 571 and RC 572. A combination of four or more external hot shorts is required to align a LOCA.
Pressurizer Auxiliary Spray	CVC 311	A19, A22	De-energize dc power supply	The electrical faults required to open this path have not been analyzed in the EPM report. However a check valve in the line would prevent a LOCA in the event of spurious valve operation.
Normal Charging	CVC 310A CVC 310B	E10,A19,A20,A21,A22,G25 A19, A22	De-energize dc power supply	The electrical faults required to open these paths have not been analyzed in the EPM report. However a check valve in the lines would prevent a LOCA in the event of spurious valve operation.

Table 4.1-3: RESULTS OF QUALITATIVE AND QUANTITATIVE SCREENING ANALYSIS							
Fire Area/Zone	Fire Area Description	Qualitative Screening Question		Ignition Freq.	Conditional CDF	Core Damage Frequency	Compartment Screens Out
		1	2				
A/1	Diesel Generator "B" Room	Y	Y	3.05E-02	6.36E-03	1.94E-04	
A/2	Diesel Generator "A" Room	Y	Y	3.05E-02	5.08E-04	1.55E-05	
A/3	Safety Injection Pump Room	Y	Y	6.70E-03	4.82E-03	3.23E-05	
B/4	Charging Pump, VCT & Non-Regen Room	Y	Y	5.23E-03	2.33E-05	1.22E-07	Yes
C/5	Component Cooling Pump Room	Y	Y	8.60E-03	8.58E-02	7.38E-04	
A/6	Aux. Feed Pump Room	Y	Y	3.70E-03	5.76E-06	2.13E-08	Yes
A/7	Aux. Bldg. Hallway (Ground Floor)	Y	Y	1.62E-02	3.39E-01	5.49E-03	
A/8	Boron Injection Tank Room	Y	Y	1.55E-03	3.18E-04	4.93E-07	Yes
D/9	North Cable Vault	Y	Y	1.40E-03	9.43E-02	1.32E-04	
E/10	South Cable Vault	Y	Y	7.64E-03	1.71E-03	1.31E-05	
A/11	Pipe Alley	Y	Y	1.40E-03	9.43E-02	1.32E-04	
A/12	Waste Holdup Tk/RHR HX	Y	Y	1.26E-03	7.05E-05	8.88E-8	Yes
A/13	Chem. Stor./Boric Acid Batch Tk	N	N	N/A	N/A	N/A	Yes
A/14	Solid Waste Handling Room	N	N	N/A	N/A	N/A	Yes
A/15	Aux. Bldg. Second Level	Y	Y	5.42E-03	2.71E-04	1.47E-06	
A/16	Battery Room	Y	Y	6.29E-03	2.97E-01	2.34E-03	
A/17	HVAC Equipment Rm for Control Rm	N	Y	2.15E-03	1.41E-05	3.04E-08	Yes
A/18	Unit 1 Cable Spread Room	Y	Y	1.53E-03	8.1E-03	1.24E-05	
A/19	Unit 2 Cable Spread Room	Y	Y	4.69E-03	4.80E-01	2.25E-03	

Qualitative Screening Questions: (1) Area contains appendix R SSD equipment (2) Fire Induced Intiating Event

Table 4.1-3 (cont): RESULTS OF QUALITATIVE AND QUANTITATIVE SCREENING ANALYSIS							
Fire Area/Zone	Fire Area Description	Qualitative Screening Question		Ignition Freq.	Conditional CDF	Core Damage Frequency	Area Screens Out
		1	2				
A/20	Elec. Swgr./Elec. Equip. Room	Y	Y	1.17E-02	4.50E-01	5.27E-03	
A/21	Rod Control Room	Y	Y	3.38E-03	1.40E-02	4.74E-05	
A/22	Control Room	Y	Y	4.75E-03	4.78E-01	2.27E-03	
A/23	Hagan Room	Y	Y	4.75E-03	3.47E-04	1.65E-06	
F/24	Containment	Y	Y	N/A	N/A	N/A	Yes[c]
G/25	Turbine Building (Water Treatment)	N	N	4.20E-03	1.00	1.06E-01	
G/25	Turbine Building (Condensate Polishing)	N	N	7.22E-03			
G/25	Turbine Building (Health Physics)	Y	Y	2.00E-03			
G/25	Turbine Building (Switchgear Room)	Y	Y	1.15E-02			
G/25	Turbine Building (General)	Y	Y	5.11E-02			
G/25	Turbine Building (DS Diesel)	N	Y	3.03E-02			
G/26	Yard Transformers	N	Y	6.97E-03	9.94E-03		
H/27	RHR Pit	Y	N	N/A	N/A	N/A	Yes[a]
G/28	New & Spent Fuel Stor., Hot Shop	N	N	N/A	N/A	N/A	Yes
G/29	Service Water Pump Area	Y	Y	1.15E-02	2.75E-02	3.16E-04	
G/30	Diesel Oil Storage Tank	Y	Y	1.15E-03	3.53E-05	4.06E-08	Yes
G/31	Refueling Water Storage Tank	Y	N	N/A	N/A	N/A	Yes[b]
G/32	Primary Water Storage Tank	N	N	N/A	N/A	N/A	Yes
G/33	Condensate Storage Tank	N	N	N/A	N/A	N/A	Yes[b]
G/34	"C" Battery Room	N	N	N/A	N/A	N/A	Yes
*/35	Radwaste Building	N	N	N/A	N/A	N/A	Yes
*/36	"B" & "C" Waste Evaporator Enclosure	N	N	N/A	N/A	N/A	Yes

Qualitative Screening Questions: (1) Area contains appendix R SSD equipment (2) Fire Induced Intitaing Event *No Fire Area Defined

a) Appendix R components in this zone are not required for hot shutdown.

b) Appendix R components not susceptible to fire damage.

c) Containment screened out per evaluation described in FIVE, page 6-6 i.e. plant experience does not indicate reccurring containment fires and redundant Appendix R trains not susceptible to damage by a single plume or hot gas layer.

Table 4.3-1 - Critical Separation Distances for Damage due to Primary Fire Sources[1]

Run ID	Source Description	Fire Duration (Minutes)	Maximum HRR (kW) Total HR (kJ)	HGL Temperature (K)[2]	Min. Cable Elevation above Source	Min. Cable Horizontal Distance from Source	Min. SWGR Horizontal Distance from Source
GENRUN1	Human Occupancy Trash Fire - HGL Forms	12	324	459	9'-3" [3] (8'-1")	1'-6"	2'-11"
GENRUN2	Maintenance Refuse Fire - HGL Forms	18	149	399	6'-11" [3] (5"-9")	0'-6"	1'-7"
GENRUN3	Closed Door, Vented Vertical Cabinet Fire - HGL Forms	37	338	461	6'-6"	2'-6"	2'-10"
GENRUN4	Open Vertical Cabinet Fire - HGL Forms	23	897	568	I[4]	5'-1" [4]	7'-4"
GENRUN5	Closed Door, Vented Vertical Cabinet Fire - No HGL Forms	37	332	N/A	6'-2"	1'-9"	2'-5"
GENRUN6	Open Vertical Cabinet Fire - No HGL Forms	24	877	N/A	9'-2"	3'-9"	5'-2"
GENRUN7	Electric Motor/ Small Electrical Panel Fire - HGL Forms	30	69	360	3'-6"	0'-0" [4]	1'-0"
GENRUN8	Electric Motor/ Small Electrical Panel Fire - No HGL Forms	31	69	N/A	3'-5"	0'-0"	0'-11"

1 Distances for cable are based on a temperature of 523 K and a heat flux of 5700 W/m^2. Distances for remote SWGR are based on a heat flux of 10 kW/m^2.
2 Predicted hot gas layer temperatures correspond to a room size of 10m x 5m x 5m, or 255 m^3 (8830 ft^3)
3 These values correspond to minimum elevation above the floor, the values in parentheses correspond to the distance between the top of the transient source to the bottom of the tray.
4 Flame height and plume temperature are sufficient to damage overhead cable at any elevation in this room.
5 For room size modelled using in COMPBRN hot gas layer temperature exceeds 523 K; Minimum separation distances are based on exceeding heat flux of 5700 W/m^2.
6 Damage is not predicted until the edge of the cable tray overlaps the edge of the fire.

Note : Potential for damage due to excessive hot gas layer temperatures in specific zones is determined using figure 3-1 and 3-4.

Table 4.3-2: Heat Release Rate From Cables Trays (Secondary Fire Source) Burning Above Electrical Cabinet (Primary Fire Source)

Number of Trays Above Cabinet	Total Heat Release Rate from Burning Cable Trays (kW)		
	Tray Width = 1 ft	Tray Width = 1.5 ft	Tray Width = 2 ft
1	55 kW	124 kW	220 kW
2	267 kW	483 kW	754 kW
3	636 kW	1078 kW	1603 kW
4	1162 kW	1909 kW	2766 kW
5	1845 kW	2995 kW	4243 kW

Table 4.3-3 Critical Separation Distances for Ignition of Overhead Cable Tray due to Primary Fire Sources

Run ID	Source Description	Fire Duration (Minutes)	Maximum HRR (kW) Total HR (kJ)	HGL Temperature (K)	Min. Cable Elevation above Source	Min. Cable Horizontal Distance from Source
GENRUN11	Closed Door, Vented Vertical Cabinet	37	338	461	3' 6"	0'0"
GENRUN12	Open Vertical Cabinet	24	897	568	7' 5"	0' 11"

Table 4.3-4: Minimum Hot Gas Layer Space to Avoid Damage/Ignition vs Exposure Fire Type and Size

Fire Type	Maximum Heat Release Rate (kW)	Total Heat Release (kJ)	Maximum Short duration Heat Release (kJ)	Min. HGL Space to Prevent Damage (m^3)							
				Short Duration[1]				Long Duration[2]			
				TI=773	TD=523	TD=433	TD=338	TI=773	TD=523	TD=433	TD=338
Human Occupancy Trash Fire	325	1E+5	9.8E+4	41*	69*	102*	279*	17	28	42	114
Maintenance Refuse Fire	145	1E+5	4.4E+4	18*	31*	46*	125*	17	28	42	114
Closed Door, Vented Vertical Cabinet	337	5E+5	1.0E+5	42	70	104	285	84*	141*	208*	570*
Open Door, Vented Vertical Cabinet	895	7E+5	2.7E+5	113	190	281	769	117*	197*	291*	798
Electric Motor/Small Cabinet	69	1.2E+4	2.1E+4	9	15	22	60	20*	34*	50*	137*
Closed Door, Vented Cabinet with one 2'wide tray above	558	7.6E+5	1.6E+5	67	112	167	456	128*	214*	316*	865*
Closed Door Vented Cabinet with two 2' wide trays above	1092	1.0E+6	3.3E+5	139	231	344	941	168*	281*	417*	1141*
Closed Door Vented Cabinet with three 2' wide trays above	1941	1.65+6	5.8E+5	243	407	605	1653	277*	463*	688*	1882

(1) Corresponds to a heat loss factor of 0.85 which is recommended for short duration intense fires
(2) Corresponds to a heat loss factor of 0.94 which is recommmended for fire durations longer than 5 minutes
(*) Indicates minimum room size for a given exposure fire type/target damage temperature combination.

TD is damage temperature

TI is ignition temperature for cable

Table 4.3-5a: Ceiling Jet Layer Thickness and Critical Horizontal Separation Distances for Damage and Ignition (For UNCONFINED CEILING JETS - ie. where ratio of H/W <0.5)

H = height from fire source to ceiling
W = Width of Room (minimum room dimension)

Cabinet Fire Size (kW)	Number of Overhead Cable Trays	Combined Heat Release Rate (kW)	Plume Temperature at Ceiling (°F)[2]			Ceiling Jet Layer Thickness (ft)[3]			Minimum Horizontal Separation Distance for:[4] Damage (D ft)[5] Ignition I ft)[6]		
			Ceiling Height			Ceiling Height			Ceiling Height		
			13'	15'	20'	13'	15'	20'	13'	15'	20'
338	0(6.6')[1]	338	700	500	225	1.0	1.3	2.0	D=3.2 I=0	D=2.1 I=0	D=0 I=0
338	1(8')	558	1500	850	350	0.8	1.1	1.8	D=6 I=2	D=3.5 I=0	D=0 I=0
338	2(8')	1092	1600	1350	540	0.8	1.1	1.8	D=6.3 I=2.7	D=7.5 I=2.1	D=3 I=0
338	3(9')	1941	1600	1600	850	0.8	1.1	1.8	D=6.3 I=2.0	D=7.5 I=3.0	D=5.5 I=0
897	0(6.6')	897	1600	975	450	1.0	1.3	2.0	D=8.3 I=3.2	D=4.2 I=2.1	D=3.4 I=0

(1) Assumed elevation of base of fire
(2) Derived from FIVE 5E
(3) 0.15 x height from base of fire source to ceiling
(4) Derived from FIVE 6B
(5) Assume initial room temperature is 68°F, damage temperature is 482°F (523K)
(6) Assume initial room temperature is 68°F, ignition temperature is 932°F (773K)

Table 4.3-5b: Ceiling Jet Layer Thickness and Critical Horizontal Separation Distancesfor Damage and Ignition (For CONFINED CEILING JETS where ratio of 0.5 < H/W <1)

H = height from fire source to ceiling
W = Width of Room (minimum room dimension)

Cabinet Fire Size (kW)	Number of Overhead Cable Trays	Combined Heat Release Rate (kW)	Plume Temperature Rise at Ceiling (°F)[2]			Ceiling Jet Layer Thickness (ft)[3]			Minimum Horizontal Separation Distance for:[4] Damage (D ft)[5] Ignition I ft)[6]		
			Ceiling Height			Ceiling Height			Ceiling Height		
			13'	15'	20'	13'	15'	20'	13'	15'	20'
338	0 (6.6')[1]	338	700	500	225	1.0	1.3	2.0	D=6.4 I=0	D=11 I=0	D=0 I=0
338	1 (8')	558	1500	850	350	0.8	1.1	1.8	D=8.8 I=5	D=7.5 I=0	D=0 I=0
338	2 (8')	1092	1600	1350	540	0.8	1.1	1.8	D=11 I=5	D=10.5 I=7	D=12 I=0
338	3 (9')	1941	1600	1600	850	0.8	1.1	1.8	D=11 I=4	D=14 I=6	D=11 I=0
897	0 (6.6)	897	1600	975	450	1.0	1.3	2.0	D=14 I=6.4	D=8.4 I=8.4	D=13 I=0

(1) Assumed elevation of fire source
(2) Derived from FIVE 5E
(3) 0.15 x height from base of fire source to ceiling
(4) Derived from FIVE 6B
(5) Assume initial room temperature is 68°F, damage temperature is 482°F (523K)
(6) Assume initial room temperature is 68°F, ignition temperature is 932°F (773K)

Table 4.6-1

Results of Preliminary Detailed Analysis Phase

Fire Area Zone	Original Ignition Frequency	Adjusted Ignition Frequency	Method of Reduction	Conditional CDF	Core Damage Frequency	Compartment Screens Out
A/1	3.051E-02	N/A	N/A	6.36E-03	1.94E-04	N
A/2	3.05E-02	N/A	N/A	5.08E-04	1.55E-05	N
A/3	6.70E-03	6.54E-05	Fire modeling showed many sources did not damage safe shutdown equipment	4.82E-03	3.15E-07	Y
C/5	8.60E-03	1.32E-05	Fire modeling showed many sources did not damage safe shutdown equipment	8.58E-02	1.14E-06	Y
A/7	1.62E-02	N/A	N/A	3.39E-01	5.49E-03	N
D/9	1.40E-03	2.10E-05	Credit for automatic fire suppression prior to damage	9.43E-02	2.56E-07	Y
E/10	7.64E-03	2.60E-04	Credit for automatic fire suppression prior to damage and fire modeling showed some sources did not damage safe shutdown equipment	1.71E-03	4.44E-07	Y
A/11	1.40E-03	0.0	Fire modeling showed no fire sources did not damage safe shutdown equipment	9.43E-02	0.0	Y
A/15	5.42E-03	2.69E-03	Fire modeling showed some fire sources did not damage safe shutdown equipment	2.71E-04	7.28E-07	Y
A/16	6.29E-03	N/A	N/A	2.97E-01	2.34E-03	N
A/18	1.53E-03	4.72E-05	Fire modeling showed some fire sources did not damage safe shutdown equipment	8.1E-03	3.87E-03	Y
A/19	4.69E-03	N/A	N/A	4.80E-01	2.25E-03	N
A/20	1.17E-02	N/A	N/A	4.50E-01	5.27E-03	N
A/21	3.38E-03	1.35E-05	Fire modeling showed some fire sources did not damage equipment	1.40E-02	1.89×10^{-7}	Y
A/22	4.75E-03	N/A	N/A	4.78E-01	2.27E-03	N

Table 4.6-1

Results of Preliminary Detailed Analysis Phase

A/23	4.75E-03	1.59E-03	Credit for manual suppression	3.47E-04	5.50E-07	Y
G/25	4.20E-03	N/A	N/A	1.00	1.06E-01	N
G/26	6.97E-03	N/A	N/A	9.94E-03	6.93E-05	N
G/29	1.15E-02	N/A	N/A	2.75E-02	3.16E-04	N

Table 4.6-2a Fire Zone 1 Fire Scenarios

Scenario #	Fire Sources	IEF	Cable Trays Damaged	Conduit damaged	Direct SSD Equipment Damage	Appendix R Safe Shutdown Equipment Disabled	Credited Non-Appendix R Equipment Disabled
1-1 (BASE CASE)	EDG Control Panel EDGB Transients Welding	2.62E-03	None	Various conduits serving EDGB Ctl. Pnl.	EDGB Ctl.Pnl. EDGB EDGB Auxiliaries	EDGB-FOTP-A EDG-FOTP-B HVE-17 HVS-5	Offsite power to Bus E2 failed (bkr 52/28B)

Table 4.6-2b Fire Zone 2 Fire Scenarios

Scenario #	Fire Sources	IEF	Cable Trays Damaged	Conduit damaged	Direct SSD Equipment Damage	Appendix R Safe Shutdown Equipment Disabled	Credited Non-Appendix R Equipment Disabled
2-1 (BASE CASE)	EDGA Control Panel EDGA Transients Welding	2.62E-03	None	Various conduits serving EDGA Ctl. Pnl.	EDGA Ctl.Pnl. EDGA EDGA Auxiliaries	EDG-FOTP-A EDG-FOTP-B HVE-18 HVS-6	Offsite power to Bus E1 failed (bkr 52/18B)

Table 4.6-2c Fire Zone 7 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
7-1	Waste Evap. Equip. Panel Gas Stripper Panel A Gas Stripper Panel B	1.46E-03	R76 R78 R79 R84	None	None	EDG-B EDG-B FOTP HVE-17 HVS-5 HVH-6B SI-878B SIP-B (Power Cable)	Offsite Power to Bus E2
7-2	BA Evap. Equip. Panel A BA Evap. Equip. Panel B	9.74E-04	CR100-SA PR100-SA R76 R78 R79 R84	None	None	EDG-B EDG-B FOTP HVE-17 HVS-5 HVH-6B SI-878B SIP-B (Power Cable) Battery Charger A-1	Offsite Power to Bus E2
7-3	Transients Welding	2.87E-06	R85	None	None	MD AFW Pump A (Power Cable) TCV-1903A V2-16A V2-16B V2-20B	None

Table 4.6-2c Fire Zone 7 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
7-4	Station Air Compressor	4.72E-06	CR101-SA CR102-SB PR101-SA PR102-SB R10 R48 R49 R50 R75 R76 R77 R78 R79	DS040A DS041B DS048A DS048B	MCC-5 MCC-10	EDG-A, EDG-B EDG FOTP-A EDG FOTP-B HVE-17, HVE-18 HVS-5, HVS-6 HVH-5A, HVH-6A HVH-6B Battery Charger A-1 SWP A, B, C Chg Pump B, C SIP A, B AFW-V2-16A AFW-V2-16B AFW-V2-20B V2-14C CC-716A, CC-735 CC-749A CVC-381 LCV-115B, LCV-115C MCC-5, MCC-10 RHR-744A, RHR-750 RHR-759A SI-860A, SI-861A SI-862A, SI-863A SI-864A, SI-865A SI-865C, SI-866B SI-867A, SI-869 SI-870A, SI-878A SI-878B V6-12A, -12B, -12C V6-16B, V6-16C	Offsite Power to Bus E1 Offsite Power to Bus E2

Table 4.6-2c Fire Zone 7 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
7-5	Instr. Air Compressor A Instr. Air Compressor B	9.44E-06	CR100-SA CR101-SA CR102-SB PR100-SA PR101-SA PR102-SB R10 R49 R50 R75 R76 R78 R79	DS040A DS041B DS048A DS048B	MCC-5 MCC-10	EDG-A, EDG-B EDG FOTP-A EDG FOTP-B HVE-17, HVE-18 HVS-5, HVS-6 HVH-5A, HVH-6A HVH-6B Battery Charger A-1 SIP A, B AFW-V2-16A AFW-V2-16B AFW-V2-20B V2-14C CC-716A, CC-735 CC-749A CVC-381 MCC-5, MCC-10 RHR-744A, RHR-750 RHR-759A SI-860A, SI-861A SI-862A, SI-863A SI-864A, SI-865A SI-865C, SI-866B SI-867A, SI-869 SI-870A, SI-878A SI-878B V6-12A, V6-12B V6-16B, V6-16C	Offsite Power to Bus E1 Offsite Power to Bus E2

Table 4.6-2c Fire Zone 7 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
7-6	MCC-5	4.87E-04	None	None	MCC-5	Battery Charger A Battery Charger A-1 CC-716A, -735, -749A CVC-381 EDG-A EDG-A FOTP HVE-18 HVH-5A, -6A HVS-6 Instrument Bus 1 RHR-744A, -750 RHR-759A SI-860A, -861A, -862A SI-863A, -864A, -865A SI-865C, -866B, -867A SI-869, -870A, -878A V1-8A V6-12A, -12B, -12D MCC-10 (Includes HVE-8B AFW-V2-16A, -16B AFW-V2-20B V2-14A, -14C V6-16B, -16C)	None
7-7	MCC-10	4.87E-04	None	None	MCC-10	HVE-8B AFW-V2-16A, -16B AFW-V2-20B V2-14A, -14C V6-16B, -16C	None

Table 4.6-2d Fire Zone 16 Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
16-1	DC MCC-A DC MCC-B	9.74E-04	All	All	All in Room	All in Room	Offsite Power to Bus E1 Offsite Power to Bus E2
16-2	Battery Rack "A" Transients/Welding	5.50E-04	PR201-SA CR201-SA R65 R73	DS089 21732B1	Battery A	Battery Rack "A" Battery Charger "A" Battery Charger "A-1" HVH-6A HVE-8A	Offsite Power to Bus E1
16-3	Battery Rack "B" Transients/Welding	5.50E-04	PR200-SB CR200-SB	DS088 21732B1	Battery B	Battery Rack "B" Battery Charger "B" Battery Charger "B-1" HVH-6A HVE-8B	Offsite Power to Bus E2
16-4	Battery Charger "A"	1.09E-04	R65 R73	DS089	BC A HVE-8A	Battery Charger "A" HVE-8A	None
16-5	Battery Charger "A-1"	1.09E-04	PR205-SA R65 R73	DS089	BC A-1 MCC-A	DC MCC-A Battery Rack "A" Battery Charger "A" Battery Charger "A-1" HVE-8A	Offsite Power to Bus E1
16-6	Battery Charger "B"	1.09E-04	PR200-SB CR200-SB	DS088	BC B BC B-1 HVE-8B	Battery Charger "B" Battery Charger "B-1" HVE-8B	None
16-7	Battery Charger "B-1"	1.09E-04	PR200-SB CR200-SB	DS088 21732B1	BC B BC B-1	Battery Charger "B" Battery Charger "B-1" HVH-6A HVE-8B	None

Table 4.6-2e Fire Zone 19 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
19-2	MUX-1, -2, -3, -4, -5 Instrument Cabinet "A" RPI Rack Carrier Instrument Rack Lundel Rack ATWS Relay Cabinet "A" ATWS Relay Cabinet "B" Aux. Relay Panels A - H Aux. Relay Panels J - M	1.13E-05	All	All	Aux Relay Racks DC and GC	All in zone	Offsite Power to Bus E1 Offsite Power to Bus E2 Offsite Power to DS Bus Deepwell Pumps A, B, C AFW Auto Initiation
19-1	Transformer in Northwest Corner of Cable Spreading Room (For LP-28?)	8.65E-07	R39	None	None	Instrument Bus 1 Instrument Bus 3 Instrument Bus 8 SWP B SWP C	AFW Auto Initiation
19-3	Aux. Relay Panel A	1.38E-04	None	None	Aux. Relay Panel A	APCH-2, -3 CC-716A, 749A CVC-200A, -381 HVE-18 HVH-5A, -6A HVS-6 PZR HTR A, B, C RC-536 RHR-744A SI-860A, -864A, -867A SI-869, -870A, -866B V1-3A	AFW Auto Initiation
19-4	Aux. Relay Panel B	1.38E-04	None	None	Aux. Relay Panel B	480 V Bus 1, 3 480 V Bus 2A, 2B AFW-V2-20A, -20B RV1-1, -2, -3 SI-878A	Offsite Power to Bus E1 Offsite Power to Bus E2 Offsite Power to DS Bus Deepwell Pumps A, B, C AFW Auto Initiation

Table 4.6-2e Fire Zone 19 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
19-5	Aux. Relay Panel C	1.38E-04	None	None	Aux. Relay Panel C	CVC-200C LCV-460A, -460B MS-353A, -353B MS-353C PCV-456 RC-567, -569, -571 RV1-1, -2, -3 SDAFWP SI-861A, -865C SWP-A V6-12A, -16B	AFW Auto Initiation
19-6	Aux. Relay Panel D	1.38E-04	None	None	Aux. Relay Panel D	125 VDC Pnl A AFW-V2-16B CVC-200A, -200C CVC-303A, -303C CVC-310B, -311 PCV-456 SI-856B V1-3C V6-16C	Offsite Power to Bus E1 AFW Auto Initiation
19-7	Aux. Relay Panel E	1.38E-04	None	None	Aux. Relay Panel E	CC-735 FWP-A, -B LCV-115B	AFW Auto Initiation

Table 4.6-2e Fire Zone 19 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
19-8	Aux. Relay Panel F	1.38E-04	None	None	Aux. Relay Panel F	480 V Bus E1 AFW-FCV-1424, -6416 AFW-V2-16A, 16B APPCC-2 APRH-1 APSI-1 DG-A DG-A FOT Pump MDAFWP-A RHR-750, -759A SI-862A, -863A, -865A SWP -B V1-3B, -8A V2-14A V6-12B, -16B	Offsite Power to Bus E1 AFW Auto Initiation
19-9	Aux. Relay Panel G	1.38E-04	None	None	Aux. Relay Panel G	125 VDC Pnl B AFW-FCV-1425 CVC-200B, -303B CVC-310A, -387 MDAFWP-B PCV-455C RC-523 SDAFWP SI-856A, -878B	Offsite Power to Bus E2 Offsite Power to DS Bus AFW Auto Initiation
19-10	Aux. Relay Panel H	1.38E-04	None	None	Aux. Relay Panel H	APPCC-1 PCV-455C	AFW Auto Initiation

Table 4.6-2e Fire Zone 19 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
19-11	Aux. Relay Panel J	1.38E-04	None	None	Aux. Relay Panel J	480 V Bus E2 AFW-FCV-1425 APRH-2 APSI-2 DG-B DG-B FOT Pump MDAFWP-B RC-568, -570, 572 RHR-751, -759B SI-862B, -863B SWP -C V1-3A, -8B V2-14B V6-12C, -16A	Offsite Power to Bus E2 Offsite Power to DS Bus AFW Auto Initiation
19-12	Aux. Relay Panel K	1.38E-04	None	None	Aux. Relay Panel K	APCH-1 APPCC-3 CC-716B, -749B CVC-200B HVE-17 HVH-5B, -6B HVS-5 LCV-115C RC-535 RHR-744B SI-860B, -864B, -865B SI-866A, -867B V1-3B	AFW Auto Initiation
19-13	Aux. Relay Panel L	1.38E-04	None	None	Aux. Relay Panel L	4 kV Bus 1, 2, 3, 4	Offsite Power to Bus E1 Offsite Power to Bus E2 Offsite Power to DS Bus Deepwell Pumps A, B, C AFW Auto Initiation

Table 4.6-2e Fire Zone 19 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
19-14	Aux. Relay Panel M	1.38E-04	None	None	Aux. Relay Panel M	AFW-V2-16C APSI-3 FCV-626 PCV-455C SDAFWP SI-861B, -870B SWP-D TCV-1902A TCV-1903A, -1903B V1-3C, -8C V2-14C V6-12D, -16A, -16C	AFW Auto Initiation
19-15	ERFIS MUX 1	1.38E-04	None	None	ERFIS MUX 1	480 V Bus E1 (All) AFW-FCV-1424, -6416 AFW-V2-16A, -16B APPCC-2 APSI-1 CC-716A, -735 CVC-200A, -200C CVC-381 MDAFWP-A PCV-456 RHR-744A, -750 RHR-759A SI-860A, -861A, -862A SI-863A, -864A, -865A SI-865C, -866B, -867A SI-869, -870A SWP-A, -B V1-3A, -8A V2-14A, -14C	Offsite Power to Bus E1 AFW Auto Initiation

Table 4.6-2e Fire Zone 19 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
19-16	ERFIS MUX 2	1.38E-04	None	None	ERFIS MUX 2	480 V Bus E2 (All) AFW-FCV-1425 AFW-V2-16C APPCC-3 APSI-2, -3 CC-716B CVC-200B FCV-626 LCV-115C MDAFWP-B PCV-455C RHR-744B, -751 RHR-759B SI-860B, -861B, -862B SI-863B, -864B, -865B SI-866A, -867B, -870B SWP-C, -D V1-3B, -3C, -8B, -8C V2-14B	Offsite Power to Bus E2 Offsite Power to DS Bus AFW Auto Initiation
19-17	ERFIS MUX 3	1.38E-04	None	None	ERFIS MUX 3	APPCC-1 CVC-310A, 310B, -311 DG-A, -B FWP-A, -B LCV-115B LCV-460A, -460B MS-353A, -353B MS-353C	AFW Auto Initiation

Table 4.6-2f Fire Zone 20 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
20-1	Instrument Bus 1	2.38E-05	None	None	Inst Bus 1	Instrument Bus 1	None
20-2	Instrument Bus 2	2.38E-05	None	None	Inst Bus 2	Instrument Bus 2	None
20-3	Instrument Bus 3	2.38E-05	None	None	Inst Bus 3	Instrument Bus 3	None
20-4	Instrument Bus 4	2.38E-05	None	None	Inst Bus 4	Instrument Bus 4	None
20-5	Instrument Bus 6	2.38E-05	None	None	Inst Bus 6	Instrument Bus 6	None
20-6	Instrument Bus 7	2.38E-05	None	None	Inst Bus 7	Instrument Bus 7	None
20-7	Instrument Bus 8	2.38E-05	None	None	Inst Bus 8	Instrument Bus 8	None
20-8	Instrument Bus 9	2.38E-05	None	None	Inst Bus 9	Instrument Bus 9	None
20-9	Pressurizer Heater Control Panel	4.19E-07	R68	None	None	None	AFW Auto Actuation
20-10	Inverter A	4.19E-07	R65 R73 R74	None	Inverter A	Inverter A EDG A bkr 52-17B HVE-17 HVS-6 Instrument Bus 1 Instrument Bus 2 RHR-759A 125 VDC Panel A	AFW Auto Actuation
20-11	Inverter B	4.19E-07	None	None	Inverter B	Inverter B	AFW AUto Actuation

Table 4.6-2f Fire Zone 20 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
20-12	Aux Relay Rack Inverter	1.76E-06	R7 R39	DS224 DS225	Relay Rack 64	FCV-605 HCV-758 125 VDC Panel B 480 V Bus 3 480 V Bus E2 APCH-1 APPCC-3 APRH-2 CC-716B CC-735 CVC-200B CVC-381 EDG B HVH-5A LCV-115B LCV-115C LCV-460A LCV-460B MDAFWP B SDAFWP PCV-455C PCV-456 SI-865C SI-867B SI-870B SWP-C SWP-D V1-3A V1-3B V1-3C	AFW Auto Actuation Offsite Power to DS Bus
20-13	All Relay Racks	1.32E-05	--	--	Bus E1 Bus E2	All Fire Zone 20 (see quantitative screening analysis, table 2-2)	AFW Auto Actuation Offsite Power to DS Bus

Table 4.6-2f Fire Zone 20 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
20-14	480 V Emergency Bus E1 - AFSS Successful	1.71E-03	None	None	Bus E1	Bus E1	AFW Auto Actuation
20-15	480 V Emergency Bus E1 - AFSS Fails	1.21E-06	--	--	Bus E1 Bus E2	All Fire Zone 20 (see quantitative screening analysis, table 2-2)	AFW Auto Actuation Offsite Power to DS Bus
20-16	480 V Emergency Bus E2 - AFSS Successful	1.97E-03	None	None	Bus E2	Bus E2 bkr 52-17B	AFW Auto Actuation Offsite Power to DS Bus
20-17	480 V Emergency Bus E2 - AFSS Fails	1.39E-06	--	--	Bus E1 Bus E2	All Fire Zone 20 (see quantitative screening analysis, table 2-2)	AFW AUto Actuation Offsite Power to DS Bus
20-18	MCC-2 - AFSS Successful	4.73E-04	None	None	MCC-2	MCC-2	AFW Auto Actuation
20-19	MCC-2 - AFSS Fails	3.35E-07	--	--	MCC-2 Bus E1 Bus E2	All Fire Zone 20 (see quantitative analysis, table 2-2)	AFW Auto Actuation Offsite Power to DS Bus
20-20	MCC-6 - AFSS Successful	6.42E-04	None	None	MCC-6	MCC-6	AFW Auto Actuation
20-21	MCC-6 - AFSS Fails	4.55E-07	--	--	MCC-6 Bus E1 Bus E2	All Fire Zone 20 (see quantitative screening analysis, table 2-2)	AFW Auto Actuation Offsite Power to DS Bus
20-22	MCC-9 - AFSS Successful	1.18E-04	None	None	MCC-9	MCC-9	AFW Auto Actuation
20-23	MCC-9 - AFSS Fails	8.37E-08	--	--	MCC-9 Bus E1 Bus E2	All Fire Zone 20 (see quantitative screening analysis, table 2-2)	AFW Auto Actuation Offsite Power to DS Bus

Table 4.6-2g: Fire Zone 25 Fire Scenarios

Scenario #	Sources	IEF[1]	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
25-1	480V Bus 3/2B(1.21E-03) SST2C (1.24E-04) Trans/Welding (5.2E-05)	1.36E-03		R206 DS011 DS174 DS005 DS014 DS148	BUS2B BUS3 SST2C	Bkr 52/32B (DSDG to DS)	Bkr 52/15 (OSP to E2/DS/DWC) Bus 2B (OSP to DWA/DWB) Bkr 52/16B (OSP to E2/DWC) SST2C (OSP to DS/DWC) 480V Bus 3 (OSP to DWC)
25-2	480V Bus 1/2A (1.21-03) 480V Bus 4 (2.24E-04) SSTD (1.2E-04)	1.56E-03		DS178 DS189 DS148 DS140 DS141 DS144 DS145 DS199 DS198 DS476	DS Bus	DS/32B (DSDG to DS) DS BUS DS Controls	Bkr 52/32A (OSP to DS)
25-3	4.16kv Bus 4 (1.11E-03) 4.16kV Bus 3 (1.11E-0-3) Trans/Weld (7.8E-05)	2.30E-03	R206	DS014 DS005	4.16kV Bus 3	DS/32B (DSDG to DS)	Bkr 52/32A (OSP to DS) Bkr 52/15 (OSP to DS/E2/DWC) 4.16kV Bus 3 (OSP to E2/DS/DWC)
25-4	SST2B(1.24E-04) Trans/Weld (1.9E-05)	1.43E-04			SST2B		SST2B (OSP to DWA/DWB)

Table 4.6-2g: Fire Zone 25 Fire Scenarios

Scenario #	Sources	IEF[1]	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
25-5	4.16kV Bus 1 (1.11E-03) 4.16kV Bus 2 (1.11E-03) Trans/Weld (7.8E-05)	2.30E-03		DS199 DS198	4.16kV Bus 1 4.16kV Bus 2	Bkr 52/32B DS Control	4.16kV Bus 1 (OSP to DWA/DWB) 4.16kV Bus 2 (OSP to Bus E1)
25-6	SSTA (1.24E-04)	1.24E-04		DS178 DS210 DS140 DS141 DS144 DS145 DS189		Bkr 52/32B (DSDS to DS) DS Control	Bkr 52/32A (OSP to DS) SST2A (OSP to Bus E1)
25-7	MCC3(1.23E-04) Trans/Welding (2.1E-05)	1.44E-04		DS174	MCC3	Bkr 52/32B (DSDS to DS)	Bkr 52/32A (OSP to DS) MCC3 (OSP to DWA/DWB

Table 4.6-2h Fire Zone 26 Fire Scenarios

Scenario #	Sources	IEF	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
26-1	Large Yard Transformer, fire loss of offsite power, DS System unavailable	1.25E-03	N/A	DS Power Conduit	Smoke may hamper DS operations	DS System	Offsite Power to All Buses
26-2	Large Yard Transformer fire, loss of offsite power, DS operability degraded due to smoke	4.73E-04	N/A	N/A	Smoke may hamper DS operations	None	Offsite Power to All Buses
26-3	Start Up Transformer	2.86E-03	N/A	N/A	S-U transformer	None	Offsite Power to All Buses
26-4	Large Yard Tansformer fire, DS System unavailable	1.40E-03	N/A	N/A	Smoke may hamper DS operations	DS System	None
26-5	Any Yard Transformer Fire, Plant trip	1.14E-02	N/A	N/A	None	None	None

4.6-2h: Fire Zone 29 Fire Scenarios

Scenario #	Sources	IEF[1]	Cable Trays Damaged	Conduit Damaged	Direct Damage to SSD Equipment	Appendix R Safe Shutdown Equipment Damaged	Credited Non-Appendix R Equipment Disabled
29-1	Any Service Water Pump (Motor Fire)	1.50E-03	None	Conduit Associated with Pump	Ignition Source Pump	Ignition Source Pump	None
29-2	Service Water Pump A or Service Water Pump D (Oil Fire)	1.60E-04	None	Conduit Associated with Pumps	Pumps A and B or Pumps C and D	Pumps A and B or Pumps C and D	None
29-3	Service Water Pump B or Service Water Pump C (Oil Fire)	1.60E-04	None	Conduit Associated with Pumps	Pumps A, B and C or Pumps B, C and D	Pumps A, B and C or Pumps B, C and D	None

Table 4.6-3: PERTINENT DATA FROM SANDIA CABINET FIRE TESTS

EVENT		Test PCT 5	Test 24 [3]	Test 25 [4]
1.	Smoke first observed coming from cabinet	10:00	10:30	9:30
2.	Smoke detector gives alarm	n/a	n/a	10:00
3.	Ignition	15:33	15:40	15:40
4.	Significant Flame Spread	21:00	22:00	18:00
5.	MCR view Obscured	23:30	26:00	29:00
	Time Interval			
1.	Smoke being observed and ignition	5:33	5:10	6:10
2.	Ignitio and flame spread	5:27	6:20	2:20
3.	Flame spread and MCR being obscured	2:30	4:00	11:00

Table 4.6-4: Fire Zone 22 (Control Room) Fire Scenarios

Scenarios	Frequency	Cabinet(s) Damaged	MCR Controls Undamaged	Initiating Event	MCR Evacuated	Recovery Actions Outside Control Room
1	5.3E-05	RTGB "A"	AFW MDPs and TDP All AC and DC power supplied from offsite power SW Pumps SG level Indication	LOFW with stuck open PORV	No	Utilize DS power controls for CCW and Charging Pumps De-energize DC feeds to PORVs to close per DSP -002
2, 4, 6 & 8	5.6E-05	RTGB "A", "C" & "D"	None	LOSP with stuck open PORV	Yes	Utilize DS power and controls for all systems De-energize power supplies to prevent/reverse fire induced spurious actions per DSP-002
3	5.6E-05	RTGB "D"	SG level indication	LOSP	Yes	Utilize DS power and controls per DSP -002
5	3.3E-05	RTGB "C"	All safty related equipment control All indication except SG level	LOFW	No	Utilize DS SG level monitoring per DSP-002
7	9.3E-04	RTGB "B" or "E"	All equipment and indication	LOFW	No	None required

Scenarios	Frequency	Cabinet(s) Damaged	MCR Controls Undamaged	Initiating Event	MCR Evacuated	Recovery Actions Outside Control Room
9	4.4E-04	Any Other MCR panel	None initially All after 2 hours	LOFW	Yes	Utilize DS power and Control for all systems per DSP-002 De-energize power supplies to prevent/reverse fire induced spurious actions per DSP-002 Re-enter control room after 2 hours and recover all emergency shutdown systems and offsite power per DSP-002

Note 1: Although damage states 1, 5 and 7 do not require control room evacuation due adverse environmental conditions, (and therefore technically the entry condition into DSP-002 is not satisfied), it is assumed that operators will utilize DS components as required, to compensate for fire damage or random failures of safety related components. For scenarios 2, 3, 4, 5 and 6 damage is so extensive that operators are assumed to leave the control room regardless of the environmental conditions. For scenario 8 operators must leave due to severe smoke. However there is no significant damage to safety related equipment and operators may return once the fire has been extinguished and smoke has been cleared.

Note 2: DSP-002, steps 8-11 direct the operators to evaluate of the extent of fire damage and re-energization of undamaged equipment power damaged by the fire (including offsite power). For damage state 9 the ability to restore equipment is clear since no real damage has occurred. The restoration is therefore possible within a relatively short period of time. Therefore credit can be taken for restoration within say 2 hours, with a probability of failure of 0.1.

However, for the other scenarios which require the control room to be abandoned, damage to breaker controls for the E1 and E2 and BOP busses is postulated. Restoration of power would therefore require repair actions. At this time it is unclear what actions would be taken and what equipment (with the exception of the DS system), would be controllable from outside of th control room. Therefore in the initial quantification of these scenarios no credit will be given to restoration actions.

Table 4.6-5. Summary of Core Damage Frequency Results

Total CDF = 2.22E-4

Fire Zone	Scenario	Transient /Loca	Conditional CDF	Fire Scenario Frequency	Actual CDF	Scenario CDF
A/1	A1-1	TA1	1.28E-03	2.62E-03	3.37E-06	
		TQA1	2.13E-04		5.57E-07	3.92E-06
					TOTAL	3.92E-06
A/2	A2-1	TA2	2.03E-04	2.62E-03	5.32E-07	
		TQA2	1.23E-04		3.21E-07	8.53E-07
					TOTAL	8.53E-07
A/7	A7-1	TA71	1.01E-03	1.46E-03	1.48E-06	
		TQA71	1.79E-04		2.61E-07	1.74E-06
	A7-2	TA72	1.01E-03	9.74E-04	9.81E-07	
		TQA72	1.78E-04		1.74E-07	1.15E-06
	A7-3	TA73	1.77E-05	7.16E-06	1.26E-10	
		TQA73	9.87E-08		7.07E-13	1.27E-10
	A7-4	TA74	9.12E-02	4.72E-06	4.30E-07	
		TQA74	3.38E-02		1.60E-07	5.90E-07
	A7-5	TA75	9.13E-02	9.44E-06	8.61E-07	
		TQA75	1.34E-02		1.27E-07	9.88E-07
	A7-6	TA76	1.16E-2	4.87E-04	5.66E-06	
		TQA76	1.99E-03		9.67E-07	6.62E-06
	A7-7	TA77	2.01E-05	4.87E-04	9.78E-09	
		TQA77	6.92E-07		3.37E-10	1.01E-08
					TOTAL	1.11E-05
A16	A16-1	TA161	6.89E-02	9.74E-04	6.72E-05	
		TQA161	9.23E-03		8.99E-06	7.61E-05
	A16-2	TA162	1.09E-03	5.50E-04	5.99E-07	
		TQA162	1.40E-04		7.72E-08	6.76E-07
	A16-3	TA163	9.45E-04	5.50E-04	5.20E-07	
		TQA163	1.90E-04		1.05E-07	6.25E-07
	A16-4	TA164	7.01E-05	1.09E-04	7.64E-09	
		TQA164	2.29E-07		2.49E-11	7.66E-09
	A16-5	TA165	1.09E-03	1.09E-04	1.19E-07	
		TQA165	1.40E-04		1.52E-08	1.34E-07

Table 4.6-5. Summary of Core Damage Frequency Results

Total CDF = 2.22E-4

Fire Zone	Scenario	Transient /Loca	Conditional CDF	Fire Scenario Frequency	Actual CDF	Scenario CDF
	A16-6	TA166	7.03E-05	1.09E-04	7.66E-09	
		TQA166	2.29E-07		2.49E-11	7.68E-09
	A16-7	TA167	7.03E-05	1.09E-04	7.66E-09	
		TQA167	2.29E-07		2.49E-11	7.68E-09
					TOTAL	7.76E-05
A19	A19-1	TA191	2.58E-04	8.65E-07	2.23E-10	
		TQA191	8.30E-05		7.18E-11	2.95E-10
	A19-2	TA192	1.35E-01	1.13E-05	1.53E-06	
		TQA192	2.40E-01		2.71E-06	4.24E-06
	A19-3	TA193	5.96E-04	1.38E-04	8.23E-08	
		TQA193	1.73E-04		2.39E-08	1.06E-07
	A19-4	TA194	5.05E-03	1.38E-04	6.96E-07	
		TQA194	3.01E-03		4.15E-07	1.11E-06
	A19-5	TA195	2.63E-03	1.38E-04	3.64E-07	
		TQA195	8.54E-05		1.18E-08	3.75E-07
	A19-6	TA196	1.23E-02	1.38E-04	1.70E-06	
		TQA196	3.56E-03		4.92E-07	2.19E-06
	A19-7	TA197	2.18E-04	1.38E-04	3.01E-08	
		TQA197	5.25E-06		7.24E-10	3.09E-08
	A19-8	TA198	1.45E-03	1.38E-04	2.00E-07	
		TQA198	2.62E-04		3.62E-08	2.36E-07
	A19-9	TA199	3.33E-03	1.38E-04	4.60E-07	
		TQA199	1.96E-03		2.71E-07	7.31E-07
	A19-10	TA1910	1.43E-03	1.38E-04	1.97E-07	
		TQA1910	6.72E-07		9.27E-11	1.97E-07
	A19-11	TA1911	2.86E-03	1.38E-04	3.95E-07	
		TQA1911	4.96E-04		6.84E-08	4.63E-07
	A19-12	TA1912	5.96E-04	1.38E-04	8.23E-08	
		TQA1912	1.82E-03		2.51E-07	3.33E-07

Table 4.6-5. Summary of Core Damage Frequency Results

Total CDF = 2.22E-4

Fire Zone	Scenario	Transient /Loca	Conditional CDF	Fire Scenario Frequency	Actual CDF	Scenario CDF
	A19-13	TA1913	2.27E-03	1.38E-04	3.13E-07	
		TQA1913	3.30E-03		4.55E-07	7.68E-07
	A19-14	TA1914	1.51E-02	1.38E-04	2.08E-06	
		TQA1914	1.00E-04		1.38E-08	2.09E-06
	A19-15	TA1915	3.51E-03	1.38E-04	4.84E-07	
		TQA1915	2.89E-04		3.99E-08	5.24E-07
	A19-16	TA1916	2.14E-03	1.38E-04	2.95E-07	
		TQA1916	9.02E-03		1.24E-06	1.54E-06
	A19-17	TA1917	2.18E-04	1.38E-04	3.01E-08	
		TQA1917	6.72E-07		9.27E-11	3.02E-08
					TOTAL	1.50E-05
A20	A20-1	TA201	1.12E-05	2.38E-05	2.67E-10	
		TQA201	0.00E+00		0.00E+00	2.67E-10
	A20-2	TA202	4.58E-05	2.38E-05	1.09E-09	
		TQA202	0.00E+00		0.00E+00	1.09E-09
	A20-3	TA203	4.51E-05	2.38E-05	1.07E-09	
		TQA203	0.00E+00		0.00E+00	1.07E-09
	A20-4	TA204	1.12E-05	2.38E-05	2.67E-10	
		TQA204	0.00E+00		0.00E+00	2.67E-10
	A20-5	TA205	1.12E-05	2.38E-05	2.67E-10	
		TQA205	0.00E+00		0.00E+00	2.67E-10
	A20-6	TA206	1.12E-05	2.38E-05	2.67E-10	
		TQA206	0.00E+00		0.00E+00	2.67E-10
	A20-7	TA207	1.12E-05	2.38E-05	2.67E-10	
		TQA207	0.00E+00		0.00E+00	2.67E-10
	A20-8	TA208	1.12E-05	2.38E-05	2.67E-10	
		TQA208	0.00E+00		0.00E+00	2.67E-10
	A20-9	TA209	2.26E-04	4.19E-07	9.48E-11	
		TQA209	0.00E+00		0.00E+00	9.48E-11

Table 4.6-5. Summary of Core Damage Frequency Results

Total CDF = 2.22E-4

Fire Zone	Scenario	Transient /Loca	Conditional CDF	Fire Scenario Frequency	Actual CDF	Scenario CDF
	A20-10	TA2010	2.05E-03	4.19E-07	8.61E-10	
		TQA2010	9.12E-05		3.82E-11	8.99E-10
	A20-11	TA2011	2.92E-04	4.19E-07	1.22E-10	
		TQA2011	0.00E+00		0.00E+00	1.22E-10
	A20-12	TA12012	6.25E-03	1.76E-06	1.10E-08	
		TQA2012	9.20E+03		1.62E-08	2.72E-08
	A20-13	TA2013	1.27E-01	1.32E-05	1.68E-06	
		TQA2013	2.32E-01		3.06E-06	4.73E-06
	A20-14	TA2014	5.66E-04	1.71E-03	9.68E-07	
		TQA2014	1.20E-04		2.05E-07	1.17E-06
	A20-15	TA2015	1.27E-01	1.21E-06	1.54E-07	
		TQA2015	2.32E-01		2.80E-07	4.34E-07
	A20-16	TA2016	3.22E-03	1.97E-03	6.35E-06	
		TQA2016	3.71E-03		7.32E-06	1.37E-05
	A20-17	TA2017	1.27E-01	1.39E-06	1.76E-07	
		TQA2017	2.32E-01		3.22E-07	4.98E-07
	A20-18	TA12018	2.26E-04	4.73E-04	1.07E-07	
		TQA2018	0.00E+00		0.00E+00	1.07E-07
	A20-19	TA2019	1.27E-01	3.35E-07	4.25E-08	
		TQA2013	2.32E-01		7.76E-08	1.20E-07
	A20-20	TA2020	1.82E-03	6.42E-04	1.17E-06	
		TQA2020	2.58E-03		1.66E-06	2.83E-06
	20-21	TA2021	1.27E-01	4.55E-07	5.78E-08	
		TQA2021	2.32E-01		1.05E-07	1.63E-07
	A20-22	TA2022	2.54E-04	1.18E-04	2.99E-08	
		TQA2022	0.00E+00		0.00E+00	2.99E-08
	A20-23	TA2023	1.27E-01	8.37E-08	1.06E-08	
		TQA2023	2.32E-01		1.94E-08	3.00E-08
					TOTAL	2.38E-05

Table 4.6-5. Summary of Core Damage Frequency Results
Total CDF = 2.22E-4

Fire Zone	Scenario	Transient /Loca	Conditional CDF	Fire Scenario Frequency	Actual CDF	Scenario CDF
A22	A22-1	T3A221	1.07E-06	5.30E-05	5.67E-11	
		TQ5A221	6.78E-02		3.59E-06	3.59E-06
	A22-2	T3A222	6.99E-02	1.20E-05	8.38E-07	
		TQ5A222	3.03E-01		3.63E-06	4.47E-06
	A22-4,6&8	T	6.99E-02	4.40E-05	3.07E-06	
		TQ	3.03E-01		1.33E-05	1.64E-05
	A22-3	T3A223	7.00E-02	5.60E-05	3.92E-06	
		TQ5A223	2.83E-01		1.58E-05	1.98E-05
	A22-5	TA225	7.70E-06	3.30E-05	2.54E-10	
		TQA225	1.21E-07		3.99E-12	2.58E-10
	A22-7	TA227	0.00E+00	9.30E-04	0.00E+00	
		TQA227	9.87E-08		9.18E-11	9.18E-11
	A22-9	TA229	1.04E-02	4.40E-05	4.56E-07	
		TQA229	5.73E-04		2.52E-08	4.82E-07
					TOTAL	4.47E-05
G/25	G25-1	T	1.96E-03	1.36E-03	2.67E-06	
		TQ	7.29E-05		9.91E-08	2.77E-06
	G25-3	T	1.50E-04	2.30E-03	3.45E-07	
		TQ	9.96E-05		2.29E-07	5.74E-07
	G25-4	T	2.12E-05	1.43E-04	3.03E-09	
		TQ	9.87E-08		1.41E-11	3.04E-09
	G25-5	T	1.54E-04	2.30E-03	3.55E-07	
		TQ	6.41E-05		1.48E-07	5.02E-07
	G25-6	T	1.05E-05	1.24E-04	1.30E-09	
		TQ	6.72E-07		8.33E-11	1.38E-09
	G25-7	T	1.05E-05	1.44E-04	1.51E-09	
		TQ	6.72E-07		9.68E-11	1.60E-09
					TOTAL	3.85E-06
G/26	G26-1	TG261	7.77E-05	1.25E-03	9.71E-08	
		TQG261	1.93E-02		2.41E-05	2.42E-05

Table 4.6-5. Summary of Core Damage Frequency Results
Total CDF = 2.22E-4

Fire Zone	Scenario	Transient /Loca	Conditional CDF	Fire Scenario Frequency	Actual CDF	Scenario CDF
	G26-2	TG262	1.28E-03	4.73E-04	6.08E-07	
		TQG262	4.35E-03		2.06E-06	2.67-06
	G26-3	TG263	9.23E-04	2.86E-03	2.64E-06	
		TQG263	2.49E-03		7.12E-06	9.76E-06
	G26-4&5	TG264	2.99E-05	1.28E-02	3.82E-07	
		TQG264	1.51E-07		1.93E-09	3.84E-07
					TOTAL	3.70E-05
G/29	G29-1A	TG291A	1.67E-07	3.75E-04	6.27E-11	
		TQG291A	1.21E-07		4.53E-11	1.08E-10
	G29-1B	TG291B	1.67E-07	3.75E-04	6.27E-11	
		TQG2921B	1.21E-07		4.53E-11	1.08E-10
	G29-1C	TG291C	6.19E-07	3.75E-04	2.32E-10	
		TQG291C	1.01E-06		3.80E-10	6.13E-10
	G29-1D	TG291D	6.19E-07	3.75E-04	2.32E-10	
		TQG291D	1.01E-06		3.80E-10	6.13E-10
	G29-2A	TG292A	9.21E-05	8.00E-05	7.37E-09	
		TQG262A	2.02E-04		1.62E-08	2.36E-08
	G29-2D	TG292D	3.76E-06	8.00E-05	3.01E-10	
		TQG292D	5.83E-06		4.66E-10	7.67E-10
	G29-3B	TG293B	8.12E-03	8.00E-05	6.49E-07	
		TQG263B	1.92E-02		1.54E-06	2.19E-06
	G29-3C	TG293C	7.97E-03	8.00E-05	6.37E-07	
		TQG293C	2.70E-02		1.52E-06	2.16E-06
					TOTAL	4.37E-06

Figure 4.3-1: Critical Separation Distance for Tertiary Damage and Ignition

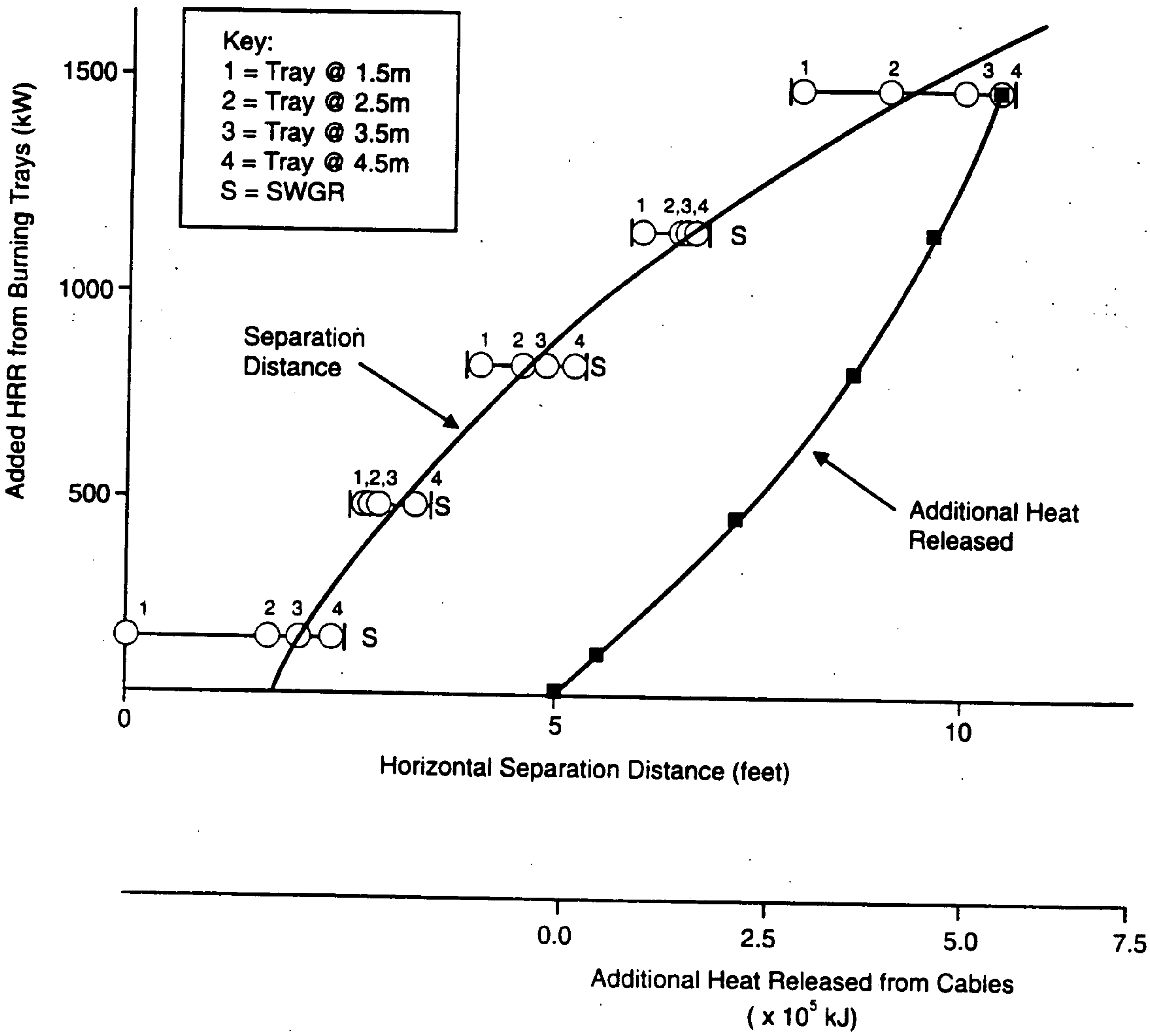


Figure 4.3-2: Minimum Volume Required to Prevent Damage or Ignition Due to Hot Gas Layers

Mininmum volume to prevent hot gas layer temperature exceeding damage or ignition temperatures is given by:

$$V_L = \frac{Q_f\ (1-X_l)}{\ln[(dT_L / 293) + 1) \times 368.4}$$

(derived from EPRI-TR 100443 [6, Appendix K]

Therefore:

For TD = 523K

$$V_L = \frac{Q_f\ (1 - X_l)}{213.45}$$

For TD = 433K

$$V_L = \frac{Q_f\ (1 - X_l)}{143.9}$$

For TD = 338K

$$V_L = \frac{Q_f\ (1 - X_l)}{52.6}$$

For TI = 773K

$$V_L = \frac{Q_f\ (1 - X_l)}{357.4}$$

Where: Qf is the heat released during the fire (kJ)

XL is the heat loss factor

= 0.85 for short duration events (5 minutes)
= 0.94 for events longer than 5 minutes where HGL fills the compartment

Figure 4.3-3: Flow Chart Showing Simplified Fire Modeling Approach

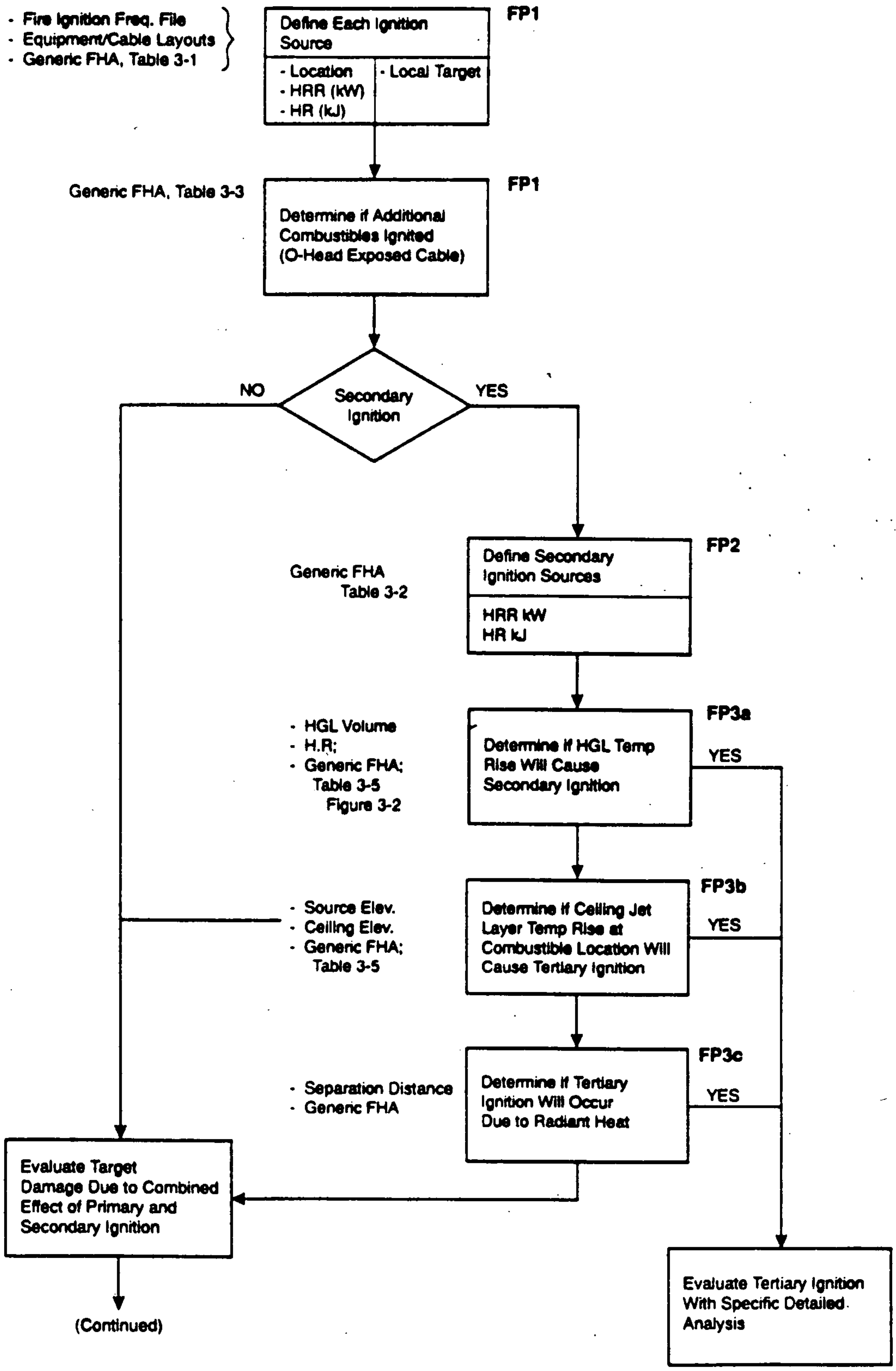
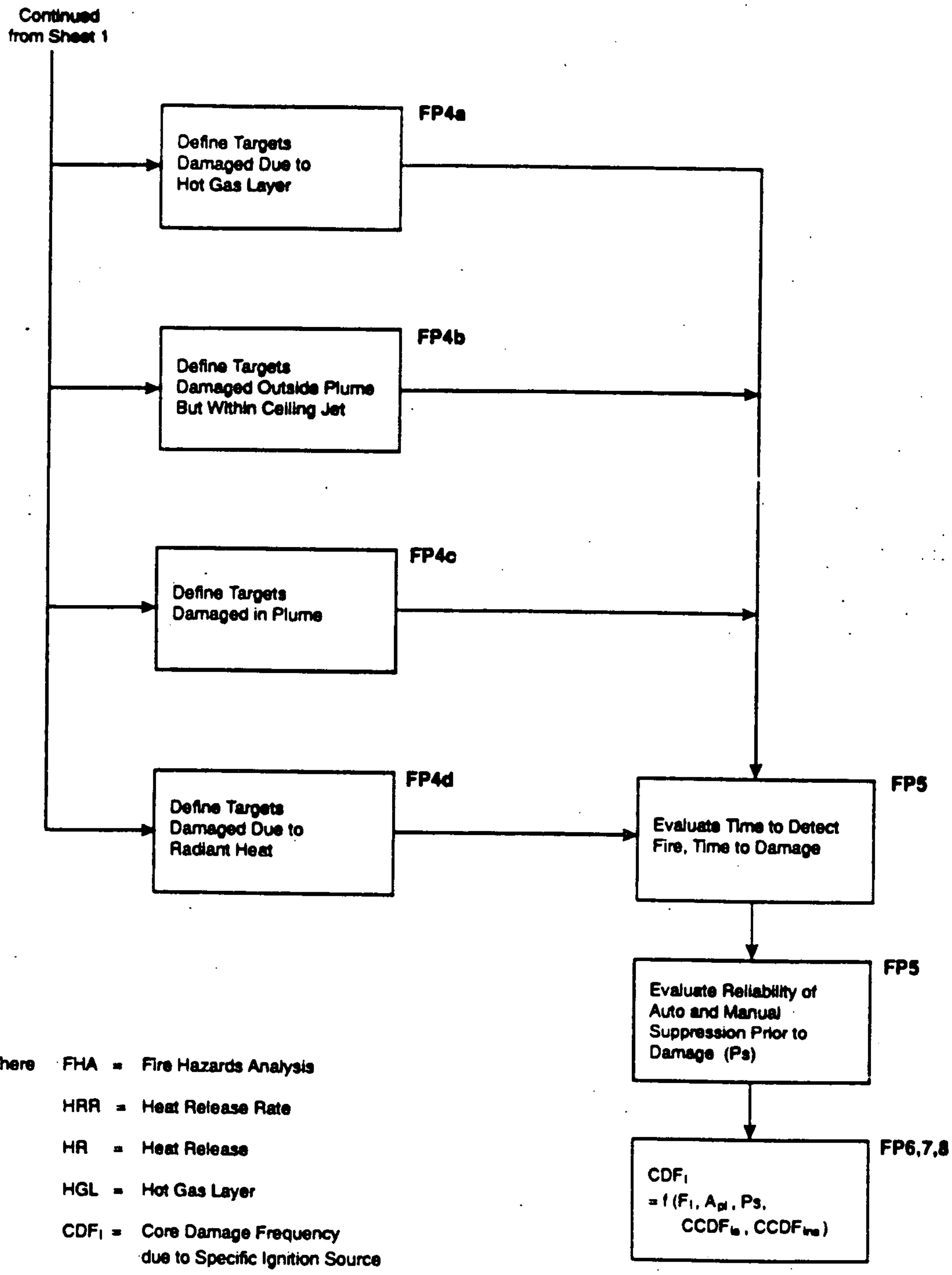


Figure 4.3-3: Flow Chart Showing Simplified Fire Modeling Approach (Continued)



Where FHA = Fire Hazards Analysis

HRR = Heat Release Rate

HR = Heat Release

HGL = Hot Gas Layer

CDF_I = Core Damage Frequency due to Specific Ignition Source

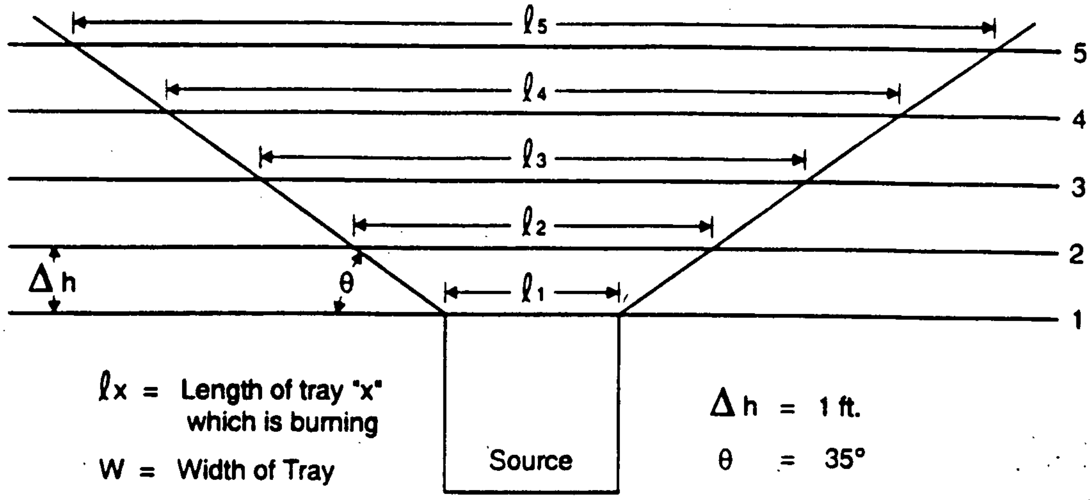
F_I = Frequency of Ignition

A_{pl} = Location Weighting Factor for Transients

$CCDF_{is}$ = Conditional CDF GivenFire Damage with Suppression

$CCDF_{ins}$ = Conditional CDF Given Fire Damage with no Suppression

Figure 4.3-4: Cable Tray Configuration Used For Secondary Combustible Model



$\ell_1 = W \qquad\qquad \Delta_1 = \ell_1 \times W = W^2$

$\ell_2 = \ell_1 + 2\left(\frac{\Delta h}{\tan\theta}\right) = W + 2.856 \text{ ft.} \qquad \Delta_2 = \ell_2 \times W = W^2 + 2.856\,W$

$\ell_3 = \ell_1 + 2\left(\frac{2\Delta h}{\tan\theta}\right) = W + 5.712 \text{ ft.} \qquad \Delta_3 = \ell_3 \times W = W^2 + 5.712\,W$

$\ell_4 = \ell_1 + 2\left(\frac{3\Delta h}{\tan\theta}\right) = W + 8.569 \text{ ft.} \qquad \Delta_4 = \ell_4 \times W = W^2 + 8.569\,W$

$\ell_5 = \ell_1 + 2\left(\frac{4\Delta h}{\tan\theta}\right) = W + 11.425 \text{ ft.} \qquad \Delta_5 = \ell_5 \times W = W^2 + 11.425\,W$

HRR per Tray

	W = 1 ft	W = 1.5 ft	W = 2 ft
A1 HRR_1	1.00 ft^2 55 kw	2.25 ft^2 124 kw	4.00 ft^2 220 kw
A2 HRR_2	3.86 ft^2 212 kw	6.53 ft^2 359 kw	9.71 ft^2 534 kw
A3 HRR_3	6.71 ft^2 369 kw	10.82 ft^2 595 kw	15.43 ft^2 849 kw
A4 HRR_4	9.57 ft^2 526 kw	15.10 ft^2 831 kw	21.14 ft^2 1163 kw
A5 HRR_5	12.42 ft^2 683 kw	19.39 ft^2 1066 kw	26.85 ft^2 1477 kw

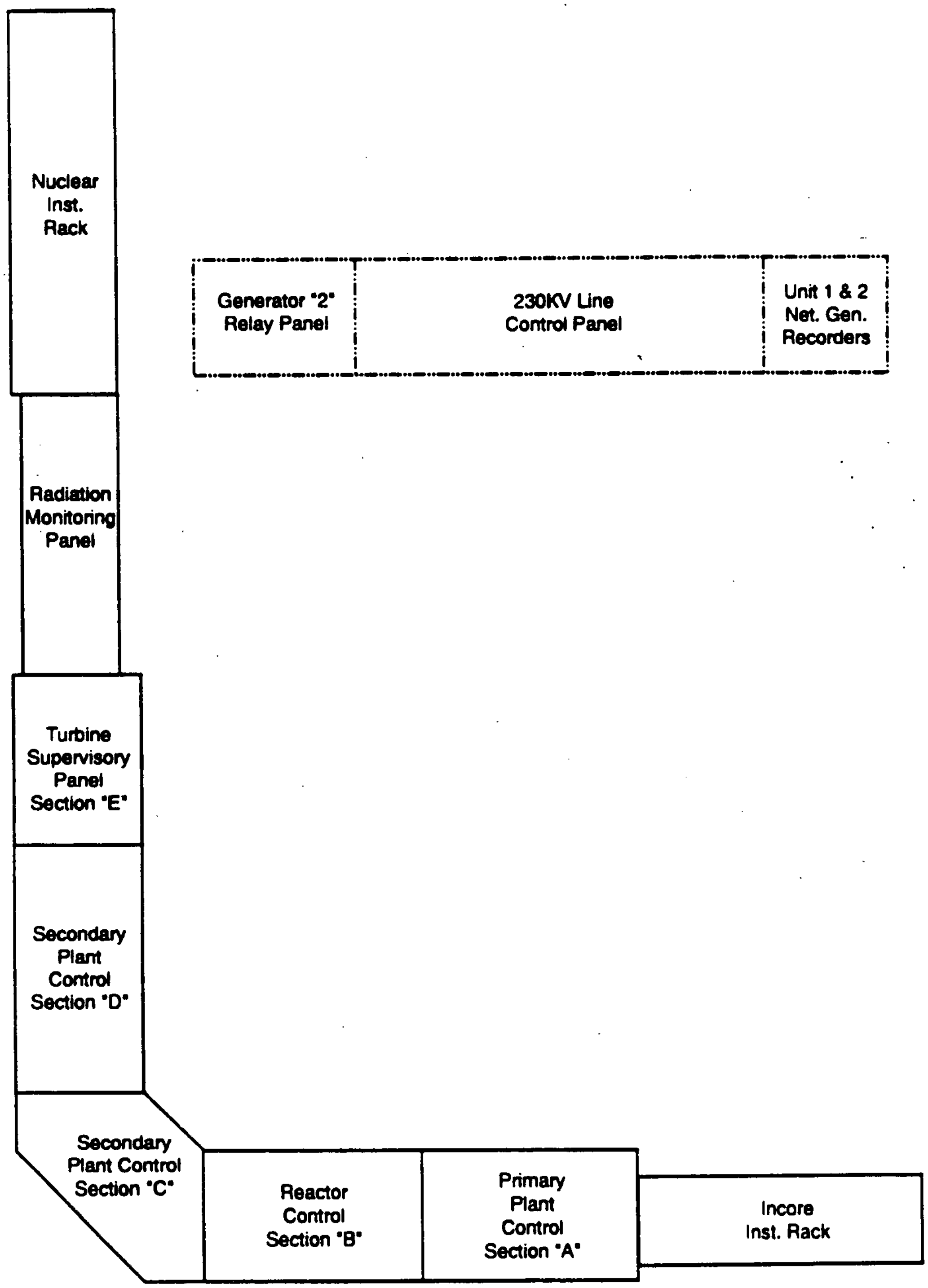
$HRR = \frac{d}{dA}(HRR) \times A$

$\frac{d}{dA}(HRR) \approx 52 \text{ BTU/s ft}^1$

$\approx 55 \text{ kw/ft}^2$

$\frac{d}{dA}(HRR)$ from FIVE for PE/PVC

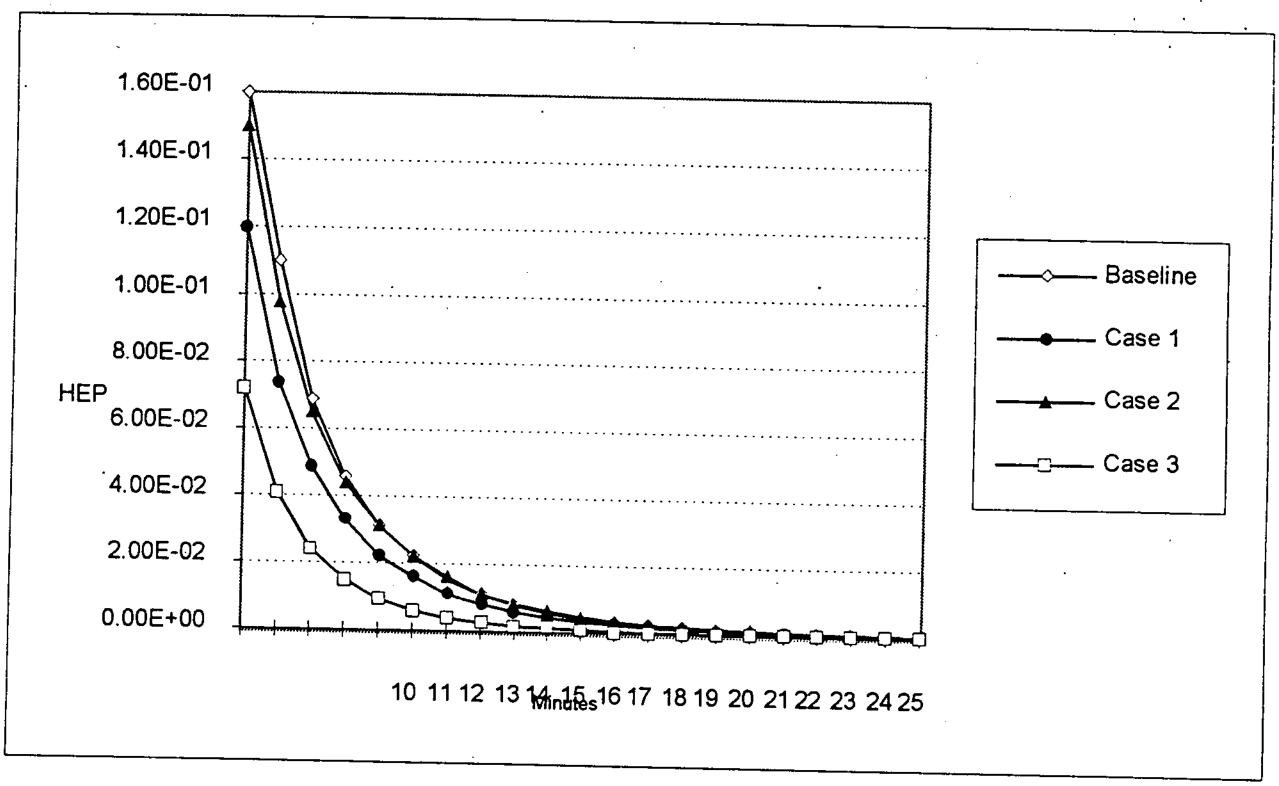
Figure 4.6-1. Control Room Layout



RTGB and LINE PANEL KEY PLAN

Figure 4.6-2. Control Room Fire Manual Suppression Time Reliability Curve

	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25
Baseline	1.60E-01	1.10E-01	6.90E-02	4.60E-02	3.10E-02	2.20E-02	1.50E-02	1.10E-02	8.00E-03	5.90E-03	4.30E-03	3.30E-03	2.50E-03	1.90E-03	1.50E-03	1.10E-03	8.80E-04	6.90E-04	5.50E-04	4.40E-04	3.50E-04
Case 1	1.20E-01	7.40E-02	4.90E-02	3.30E-02	2.20E-02	1.60E-02	1.10E-02	8.20E-03	6.00E-03	4.50E-03	3.40E-03	2.60E-03	2.00E-03	1.50E-03	1.20E-03	9.50E-04	7.60E-04	6.00E-04	4.90E-04	3.90E-04	3.20E-04
Case 2	1.50E-01	9.80E-02	6.50E-02	4.40E-02	3.10E-02	2.20E-02	1.60E-02	1.10E-02	8.40E-03	6.30E-03	4.80E-03	3.60E-03	2.80E-03	2.20E-03	1.70E-03	1.40E-03	1.10E-03	8.60E-04	7.00E-04	5.60E-04	4.60E-04
Case 3	7.20E-02	4.10E-02	2.40E-02	1.50E-02	9.30E-03	6.00E-03	3.90E-03	2.60E-03	1.80E-03	1.30E-03	8.90E-04	6.30E-04	4.60E-04	3.30E-04	2.50E-04	1.80E-04	1.40E-04	1.10E-04	8.10E-05	6.20E-05	4.80E-05



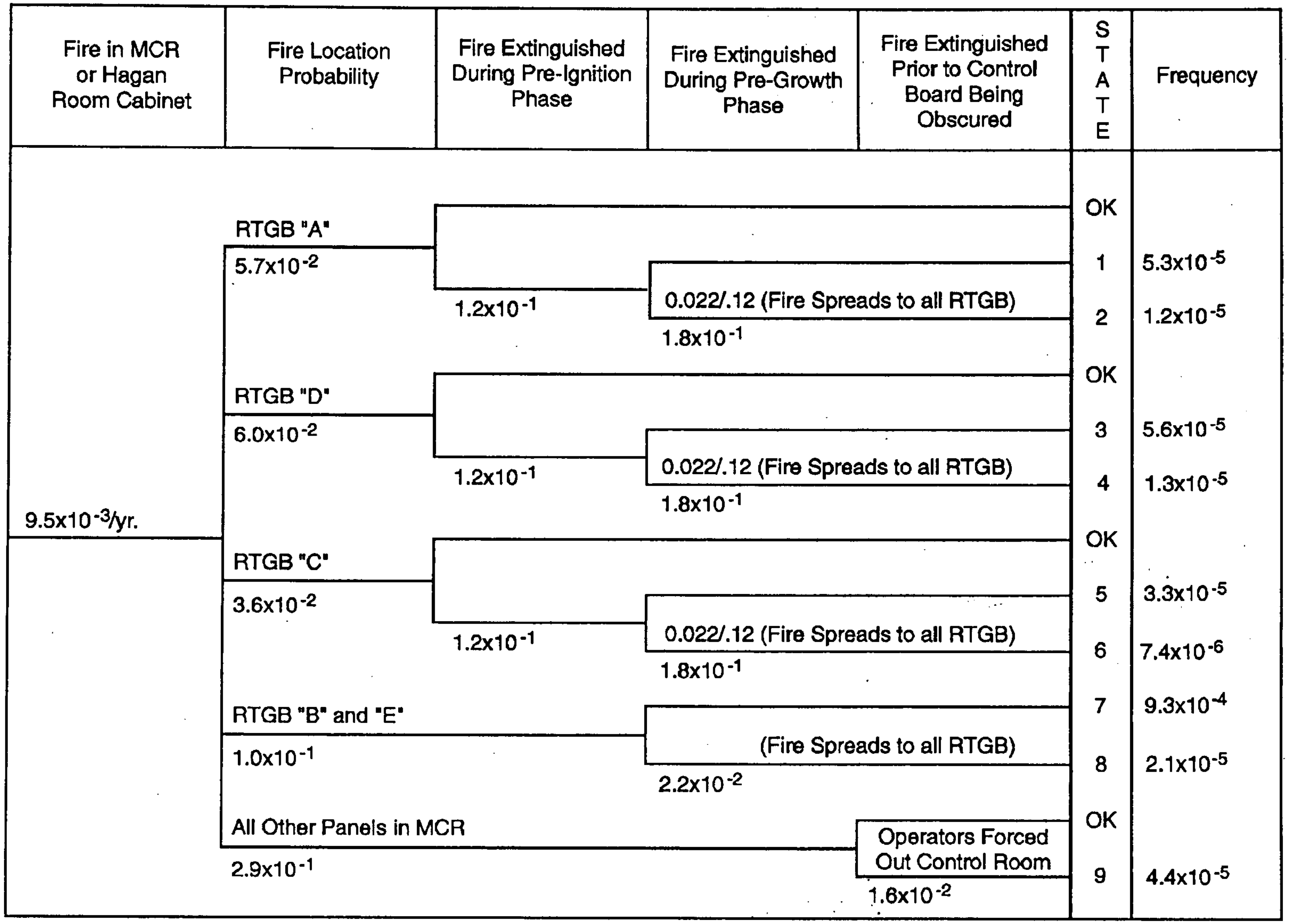


Figure 4.6-3. Control Room Fire Damage Event Tree

SECTION 5

OTHER EXTERNAL EVENTS

5.0 INTRODUCTION

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 (NRC, 1991), 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 documents the analysis of other external hazards for the HBRSEP. The approach used follows the method described in NUREG/CR-4839 (NRC, 1992) by performing an initial screening analysis, followed by bounding or detailed analyses as necessary.

The next section gives a brief description of HBRSEP, extracted from the UFSAR (CP&L, UFSAR). The following section describes the initial screening approach, and presents the conclusions of the screening analysis. Sections 5.3, 5.4, 5.5, and 5.6 discuss high winds, floods, transportation and nearby facility accidents, and failure of the Lake Robinson dam respectively. Conclusions are given in Section 5.7, and References in Section 5.8.

5.1 GENERIC PLANT DESCRIPTION

5.1.1 Site Description

The site is in northeastern South Carolina, 56 miles ENE of Columbia, the state capital. The location is about 25 miles NW of Florence, and about 35 miles NNE of Sumter, S.C. Coordinates of the site are latitude 34° 24.2'N and longitude 80° 09.5'W. It is located on the southwestern corner of Lake Robinson which is impounded to furnish cooling water for power plants at the site. The exclusion distance and low population distances are 1400 ft. and 4.5 miles respectively. Exclusion distance is the distance from the reactor to the closest point on the boundary of the exclusion area defined in 10 CFR 100. The low population distance is the distance from the reactor to the boundary of the low population zone defined in 10 CFR 100. The total site area including Lake Robinson is more than 5,000 acres. Farming is the predominant activity in the sparsely populated immediate environs of the plant site. The site surface soil is sandy and surface water drains to the lake. The region is gently rolling and is not subject to severe persistent inversions. Tornadoes occur in the region but have not affected the site. While many hurricanes affect the southeastern United States, no hurricane storm tracks were reported in the near vicinity of the site during the period between 1900 and the beginning of commercial operation of the plant. However, hurricane Hugo passed within 40 miles of the plant site in September 1989, but no weather stations within 50 miles of the plant reported sustained hurricane force winds associated with Hugo.

The nuclear unit at the site, H.B. Robinson Steam Electric Plant Unit No. 2, is located adjacent to a coal-fired steam power plant, H.B. Robinson Steam Electric Plant Unit No. 1.

The major structures of Unit 2 are the Reactor Containment, Auxiliary Building, Turbine Building, the Intake Structure, the Radwaste Facility, and the Fuel Handling Building. A general plan of the building arrangements is shown in Figure 1.2.2-1 of the UFSAR, reproduced here as Figure 5.1-1.

5.1.2 Identification of Structures, Systems and Components Susceptible to External Events

A principal concern in a study of external events 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 shutdown the reactor. The equipment necessary to achieve safe shutdown to hot standby is addressed in the Individual Plant Examination (CP&L, 1992). The majority of that equipment is safety related and, as such, is protected by the major structures, namely the Containment Building, the Auxiliary Building, and the Control Room. These are seismic class I structures, and are also designed to withstand tornado impact. However, there are important components which are not protected by these structures and are, therefore, potentially vulnerable to external influences. These include:

the switchyard,
the main transformers,
the power conversion system which is housed in the open turbine building,
the condensate storage tank,
the dedicated shutdown diesel, which is housed in its own (non-safety related) structure,
the refueling water storage tank,
the service water pumps (located on the intake structure),
the fire pumps (located on the intake structure),
the turbine-driven auxiliary feedwater pump (located in the turbine building), and
diesel fuel storage tanks and fuel oil transfer pumps.

In addition to potential damage to this equipment, there is a concern as to whether there exists the possibility for damage to the Containment Building and Auxiliary Building sufficient to damage the equipment they contain. This damage may be as a result of a direct impulsive force, or by ingress of harmful agents through penetrations in the structures. The most obvious of the latter are water, due to flooding, or toxic or flammable gases leading to control room habitability problems or fires.

5.2 SCREENING OF EXTERNAL HAZARDS

5.2.1 Description of Approach

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

The PRA Procedures Guide (NRC, 1983) provides an exhaustive list of potential external hazards which provides the starting point for the analysis. An extensive review of information on the site region and plant design is necessary to identify all external events that are applicable using the screening criteria below. For this purpose, 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 (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. 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 one of the following is applicable.

1. 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 (Kennedy, 1983) that safety-related structures designed for earthquake and tornado loadings in UBC Zone 1 can safely withstand a 3.0 psi static pressure from explosions. Hence, if the PRA analyst demonstrates 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.

2. 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, the PRA analyst may exclude an event 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 by the PRA analyst as not significantly influencing the total risk.

3. 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.

4. 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.

5. 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 criteria as summarized in Table 5.2-1. Not included in Table 5.3-1 are the three hazards specifically identified in NUREG-1407 as requiring site specific evaluations, and internal floods, which are already addressed in the IPE.

In recognition of the recommendations of NUREG-1407, a plant walkdown was performed to confirm the conclusions of the paper study.

5.2.2 Results of Screening Analysis

Based on information in the UFSAR and on the basis of the walkdown, it was determined that the conclusions of NUREG-1407 (that there are no known plant-unique other external events that pose any significant threat of severe accident within the context of the screening approach) are valid for the H.B. Robinson site, with the exception that the reservoir dam failure could result in loss of the ultimate heat sink, and provides a potential unique hazard. This is discussed further in Section 5.7 of this report.

Table 5.2.1
Screening of External Events for H. B. Robinson Unit 2

Event	Applicable Screening Criteria
Avalanche	3
Biological Events	5
Coastal Erosion	3
Drought	2,5
Fog	4
Forest Fire	3
Frost	1,4
Hail	1,4
High Tide, High Lake Level, or River Stage	4
High Summer Temperatures	5
Ice Cover	3,4,5
Landslide	3
Lightning	4
Low Lake or River Water Level	4
Low Winter Temperature	1,5
Meteorite	2
River Diversion	3,5
Sandstorm	3
Seiche	3
Snow	1
Soil Shrink-Swell Consolidation	1
Storm Surge	3
Tsunami	3

Screening of External Events for H. B. Robinson Unit 2
(Continued)

Event	Applicable Screening Criteria
Toxic Gas	4
Turbine Generated Missiles	2
Volcanic Activity	3
Waves	3

5.3 TORNADOS AND HIGH WINDS

5.3.1 Introduction

The purpose of this section of the report is to present the assessment of the likelihood of core damage arising from the impact of high winds at HBRSEP. The following sections provide: a brief description of the plant design with respect to high wind loads; an assessment of the wind hazard at the site; an identification of those components that may be used to safely shut-down the plant and which are not protected against high wind loads by being enclosed in wind resistant structures; and finally an assessment of the core damage frequency.

5.3.2 Design of HBRSEP for Wind Loadings

The design of the plant with respect to wind loads is described in Section 3.3 of the UFSAR. Essentially, structures on the plant site are designed to a 30 psf basic wind loading which the UFSAR states is conservative for the location of the site. The design wind speed corresponds to a gusted wind velocity of 108 mph.

The design against tornados is discussed in Section 3.3.2 of the UFSAR, from which the following is extracted.

> The basic philosophy for passage of a tornado directly across the site was to design against damage to critical systems, accept limited damage to the remainder of the plant, and shut the plant down if necessary for repairs.
>
> The facility was designed such that a tornado will not interfere with the plant's capability to cope with long term recovery aspects associated with the design basis accident or prevent safe shutdown of the plant. Additionally, the facility was designed such that a tornado will not effect vital structures, systems, or components so as to cause release of radioactivity to the environment.
>
> The containment structure and other buildings and structures that house vital equipment are capable of withstanding the passage of a tornado without loss of function. The walls of these buildings which are constructed of concrete a minimum of 8 inches thick provide considerable protection for the equipment inside from debris thrown about by the tornado.
>
> The Reactor Auxiliary Building is a reinforced concrete Class I structure with several doorways located in its exterior walls. The passage of a tornado across the building would create a pressure differential, causing the higher pressure in the building to blow out the doors. This would create a large vent area which would result in the rapid equalization of the pressure between the interior and exterior of the building.

Using a conservative approach to assure that the building will safely withstand the direct passage of a tornado, the building is designed for such an occurrence with no venting.

The Reactor Auxiliary Building, which includes the Control Room, is designed to withstand, without failures, a 300 mph wind loading coincident with an atmospheric pressure drop of 3 psi due to a tornado.

The intake structure is inherently safe from tornado loads as it is mostly beneath the lake water level. It was not specifically designed for the tornado loads. However the pressure drop (3 psi) would have no effect on it as it is open so that venting would occur and the water level inside would respond to the drop in pressure, thereby equalizing the pressure. The wind load would act on an insignificant portion of the structure. The structure is designed for the Class I earthquake criteria.

The Class I portion of the turbine structure houses the steam driven auxiliary feed pump which provides backup to redundant motor driven auxiliary feed pumps located in the tornado proof Reactor Auxiliary Building.

The design for the primary water storage tank and the diesel fuel oil tank includes consideration for tornado wind loads (i.e. 300 mph wind and 3 psi negative pressure).

Section 3.3.2.3 of the UFSAR states:

All components necessary for safe operation which are located outdoors and exposed to damage from tornado debris are parts of redundant systems and as such have sufficient backup to provide reasonable assurance that no loss-of-function of the systems will result because of tornado damage.

The redundancy and location of vital equipment is as follows:

1. Emergency steam generator feed is provided by a steam-driven pump backed up by two motor-driven pumps. Both motor driven pumps are inside buildings. In the event the steam lines supplying steam to the turbine-driven pump are damaged, the motor driven pumps, powered by the emergency diesel generators located in Auxiliary Building, can be used.

2. The four service water pumps are located in three separate bays in the intake structure, the middle bay containing two pumps. The pumps are sufficiently isolated to make it unlikely that a missile could damage more than one pump. The walls separating the bays and the deck above the piping are two and one half

foot thick reinforced concrete. Thus it is highly unlikely that a missile could get to the pumps.

3. The Condensate Storage Tank could be pierced by a missile. However, missile impact would have to occur at the bottom of the tank to cause total loss of water. Also, the service water system or the well water systems could be used to supply water to the secondary side of the Steam Generators to remove decay heat.

4. If a tornado or tornado debris destroys the outside electrical power supply, the unit could be tripped and either of the two emergency diesel generators, located in the Auxiliary Building, would supply sufficient power to place and maintain the plant in the safe shutdown condition. Also, a dedicated shutdown diesel is available for plant shutdown.

5. Piping and electrical connections from the Auxiliary Building to the Containment are each in two separate concrete enclosures following different routes. Other vital piping and equipment is located in below-grade trenches or pits with concrete covers.

Therefore it is concluded that the plant can withstand the effects of the design tornado without endangering the health and safety of the public.

5.3.3 Wind Hazard

5.3.3.1 Tornado Wind Hazard

The Robinson plant is located in South Carolina, and is in Region B of Twisdale and Dunn's tornado risk regionalization scheme (Twisdale, 1983), for which the observed regional occurrence rate is quoted as 4.76×10^{-4}/sq mile/year. The updated FSAR quotes an incidence rate of 1.95×10^{-3}/sq mile/year.

Data obtained from the National Severe Storms Forecasting Center (NSSFC) gives an estimate of the occurrence rate , based on occurrences in the area within a 125 NM radius about the site, of 1.4×10^{-4}/sq mile/year for winds in excess of 73 m.p.h. Data that was obtained in May 1983 from the NSSFC gave a frequency of 6.5×10^{-5}/sq mile/year. The NSSFC data is also analyzed for a 2 degree square centered on the Robinson site. The corresponding occurrence rates are 1.85×10^{-4}/sq mile/year, and 6.7×10^{-4}/sq mile/year. The difference between the two estimates from the NSSFC data is partially a result of more complete reporting of tornados in recent years, but a more significant cause of this increase is discussed below. The more recent data is adopted as the basis for frequency estimates.

In the output obtained from the NSSFC data, the complementary distribution function for frequency is given in terms of the Fujita (F classification). The results are given in the table below for the more conservative results, those for the two degree square centered on the site, for both the 1983 and the 1994 data. For the most recent data, the complementary distribution function is almost flat, which differs from typical tornado hazard curves found in the literature (see for example (Reinhold, 1982)). To understand the reason for this, the data was analyzed in more detail as discussed below.

F Classification	Range of Wind Speed (mph)	frequency of tornado with F value $\geq F_i$	
		1993 data	1983 data
F_1	73-112	1.85×10^{-4}	6.7×10^{-5}
F_2	112-157	1.73×10^{-4}	5.7×10^{-5}
F_3	157-206	1.42×10^{-4}	2.6×10^{-5}
F_4	206-260	1.3×10^{-4}	1.4×10^{-5}
F_5	260-318	0	0

Taking only the last 10 years of data for tornados greater than F1, the total path area of the 110 tornado occurrences is 299.7 square miles in a 125 nm radius circle about the HBRSEP site.

This gives a frequency of striking a point as

$$f = 299.7 \times \frac{1}{\pi\ 125^2 \times F^2} \times \frac{1}{10\ years}$$

where F is the ratio of nautical miles to miles = 1.15. Therefore,

$$f = 4.6 \times 10^{-4}/\text{year}.$$

Of the total path area covered by tornados, 179.92 square miles was contributed by a swarm of fifteen tornados that occurred on March 28, 1984. The distribution of these 15 tornados among F categories was 2 in category F_1, 4 in F_2, 3 in F_3, and 6 in F_4. This distribution is unusually skewed towards the high F numbers. The largest tornado, an F_4, had a path area of 67.34 sq mile and was by far the largest tornado in the whole data set. On removing this swarm, the resulting strike frequency becomes 1.8×10^{-4}, which is relatively low.

This calculation shows the potentially distorting effect of rare, but extreme, events. However, there is no basis for censoring the data to exclude this swarm, but we note that, for the higher F classification tornados at least, the frequency is probably on the high side.

There have been no occurrences of F5 tornados, therefore the frequency was taken from (Twisdale, 1978 and 1981) and estimated at 4×10^{-7}/year.

5.3.3.2 Non-Tornadic Winds

Data was obtained from the National Climatic Data Center in Asheville, NC on extreme wind speeds for the last 20 years at Columbia, South Carolina. This data is included in the analysis file (Ref. 5-8). This was assumed to be representative of the Robinson site and the data was fit to a Type I Extreme Value distribution using the method of moments.

$$F(v) = \exp\left[-e^{-\alpha(v-u)}\right]$$

with ` = .1444 and u = 50.74.

The anemometer at the weather station is at a height of 20 ft. Using Figure 2.3.1-1 of the UFSAR, the wind velocities were adjusted for the 50 ft. level. This gives the estimates of the frequency of exceedance shown below for both the 20 ft. and 50 ft. levels:

Wind speed (20 ft.) v (mph)	**Wind speed (50 ft.) v (mph)**	**Frequency of Exceedance F(v)**
50	56.5	.672
75	84.75	.03
80	90.4	.0145
100	113	.00082
125	141.25	.00002

5.3.3.3 Wind Hazard Curve

The hazard curve obtained by combining both hazard causes is shown in Figure 5.3-1. Above about 110 m.p.h. it can be seen that tornados dominate at the lower elevations. Since most structures are designed to 108 m.p.h., tornados are assumed to be the most significant threat at the lower levels. However, at the higher elevations, as discussed later, non-tornadic winds may be important.

5.3.4 Equipment Potentially Vulnerable to High Winds

The most likely initiating event, given high winds at the site, is a loss of offsite power, which could be prolonged as a result of structural damage to transmission lines or the switchyard. Therefore, following a loss of offsite power, it is necessary to first of all maintain emergency AC power (diesel generators). This requires all of the following:

a) one of the 2 emergency diesel generators,

b) since each day tank has a capacity for only 91 minutes of diesel generator running time (CP&L, 1992) the diesel fuel oil storage tank and the fuel oil transfer system is also required, and
c) one service water pump, and the whole service water system at least as far as the discharge piping.

In addition, to provide decay heat removal, the Auxiliary Feedwater System is required. This requires the CST to remain intact. The turbine driven auxiliary feedwater pump is in the turbine building, but the motor driven pumps are in the Auxiliary Building and are thereby protected. Even if the AFW function were lost, feed and bleed is an alternative, which in turn requires an intact RWST. (It is assumed that, under tornado conditions, no credit can be taken for the safe shutdown diesel, or for the deep well pumps or the fire water system as a make up to AFW supply.) The RWST has a limited capacity however, and would require switch over to recirculation in a few hours, which in turn requires the service water system to be operable.

The failure of any one of the following single systems will result in core damage when substituted into the IPE model, given prolonged loss of offsite power as an initiating event.

a) the onsite emergency AC power system (i.e., both emergrncy diesel generators),
b) the diesel fuel oil storage and transfer system, and
c) the service water system.

There are other combinations of failures, i.e., higher order cut sets, corresponding to multiple wind caused failures, or wind related failures combined with random equipment failures, that could result in core damage. The frequency of such events is judged to be much lower than that of the scenario identified in section 5.3.5.1 which can result from a single missile strike. One such combination for example is the simultaneous failure of the CST and the RWST as a result of missile strikes. The RWST and the CST are, however, in different locations and on opposite sides of the auxiliary building, and each is protected to some degree by surrounding structures. The conditional probabilities of missile strikes on each of these targets can be considered to be independent and since the conditional probability of a missile hitting a single target is judged to small, the probability of hitting two or more targets is much smaller than that of hitting one.

Two walkdowns were conducted to identify those components of the above mentioned systems which are not contained within buildings designed for tornadic effects. During each walkdown, a search was made for the potential for failures of non-safety related equipment which could lead to failures of safety related equipment. This was done to address the concern of Information Notice IN 93-53, Supplement 1 (NRC, 1993).

5.3.4.1 Diesel Generators

The two emergency diesel generators are contained in the Auxiliary building and are thus protected. However, the air intakes and exhaust systems are attached to the roof. There are engineering calculations to demonstrate that these components can withstand the design basis tornadic winds. There is, in addition, a dedicated shutdown diesel generator but it is housed in a structure that is not designed for tornados and hence no credit can be taken for it for the higher wind speeds.

5.3.4.2 Diesel Fuel Oil Storage and Transfer System

The diesel fuel oil storage tank is situated in the yard to the NE of the containment building. It is a seismic class I structure and as such should, as is stated in the UFSAR, withstand the design basis tornado wind pressure. The fuel oil transfer pumps are exposed and adjacent to the tank in close proximity to each other. The fuel oil transfer lines are underground except where they emerge and enter the diesel generator building at its NE corner. There is an exposed length of small diameter pipe of about 15 feet for one line, and 35 feet for the other. While these pipes cannot be claimed to be inherently robust, the target they represent is very small.

5.3.4.3 Service Water System

The service water pumps are located on the intake structure. The pump motors are located inside a metal sided structure (with a mesh roof), whose purpose is to control access to the pumps, rather than to provide physical protection. The structure does not appear to be substantial. The pump motors are anchored with four bolts, and are about 6 feet tall. To the West of the metal sided structure housing the pumps are the three large circulating pumps which are substantial enough to act as shields from low flying missiles. To the East by about 10 feet are the housings for the traveling screens which again provide some, though limited, protection. There is section of service water (discharge) piping that emerges from the SE corner of the auxiliary building near the turbine structure and is above ground for a total of about 50 feet before going underground again. This, however, is a very large pipe that is not vulnerable to being crushed and crimped, which is the most important failure mode, as it would effectively stop service water flow.

5.3.4.4 Condensate Storage Tank

The CST is protected on three sides as it lies between the turbine building and the condensate polishing building. It is exposed to the South East.

5.3.4.5 Ventilation Systems

The cooling units for the Hagen rack room are located on the auxiliary building roof. Loss of ventilation to the Hagen rack room will not impact plant shutdown based on a station blackout coping study (CP&L, SBO).

None of these structures is vulnerable to direct wind pressure, but they may be vulnerable to missile damage.

5.3.5 Estimate of Core Damage Frequency

As discussed in the previous sections, the components of interest are judged not to be susceptible to direct wind pressure, but instead, the impact of wind generated missiles is considered to be the most significant hazard. As noted during the plant walkdowns, there are very few significant sources of missiles. Furthermore, Operations Management Manual Procedure OMM-021 instructs the plant staff to make sure that potential missiles are removed or tied down given warning of a hurricane or tornado. Therefore, the only significant sources of missiles are the sheet metal sided or trailer type structures and possibly some automobiles.

5.3.5.1 Tornados

The trailers are all at the lower elevations and their disposition is such that any resulting missiles are unlikely to impact the important safety related equipment discussed above. This is based on the observation that most tornados track in a SW to NE direction and that missiles are generally entrained with the path and transported along the tornado's direction of travel (Twisdale, 1978 and 1981). The automobiles most likely to be moved would be in the SE area of the plant site, the closest target being the service water pumps, which are shielded by the circulating water pumps.

The most significant sources of missiles are the elevated sheet metal sided structures, namely the fuel building and the upper sections of Unit 1, the coal powered station. According to the UFSAR, the fuel building siding can be expected to "blow off" at about 125 m.p.h. Tornados with maximum wind speeds in excess of this can be expected, based on the hazard curve, to hit the site with a frequency of about 1.5×10^{-4} per year. However, it is well known that the wind velocity varies within the tornado path, and that only a fraction of the path sees the wind speeds corresponding to that of the F classification. Hence it is necessary to estimate the frequency with which the local tornado state at the structures exceeds this velocity. Using Table 15c of Reference 5-9, and the frequency of a tornado strike on the site of 1.5×10^{-4} per year, the frequency of a local tornado state of 125 m.p.h. and higher is estimated to be on the order of 1.4×10^{-5} per year. This is assumed to be the frequency with which missiles are generated as a result of damage to the metal sided structures.

To complete the assessment, it is necessary to estimate the likelihood that tornado generated missiles strike the vulnerable plant areas. This is done judgementally based on the following arguments. The NSSFC data shows that the majority of tornados track in the SW to NE direction. This is also supported by a histogram (Figure 1-4 in (Twisdale, 1978)). EPRI-NP-769 (Twisdale, 1978) also shows that the majority of missiles are entrained within the 73 m.p.h. boundary. EPRI-NP-2005 (Twisdale, 1981) gives examples of trajectories of missiles generated by simulations using the TORMIS code, and some from actual observations. They show that missiles are generally ejected in the direction of travel of the tornado and primarily to the right of the path. Because of this, it is judged unlikely that missiles generated by a break-up of Unit 1 would track in the direction of the service water pumps. In addition, the diesel generator fuel oil lines, and the H_2 line on the external wall of the diesel generator building are sheltered, as is the service water piping on the NE corner of the turbine building. Similarly, the Hagen rack room coolers are sheltered by the turbine building.

The most vulnerable components are the diesel generator fuel oil pumps since they are unprotected and are in the same location, and to the NE of the fuel building, which is postulated to be the primary source of missiles. The angle subtended by the target area is however extremely small, and the number of potential missiles (the aluminum sheet siding) is not large and on the order of 100. Therefore, a conditional probability of less than 0.1 of striking the target may be justified by observing that, if the missiles are randomly distributed over the width of the tornado path, which has an average value of .063 miles or 332 feet, and the target has no more than a width of 3 feet, the chance of the missile hitting the target is 3/332. This is a purely geometric argument, independent of the number of missiles. This assessment takes no account of the distribution of distance traveled by the missiles which could be argued to reduce the conditional probability further.

Therefore, the core damage frequency from this scenario is estimated as being on the order of 1×10^{-6}/yr.

5.3.5.2 Non-Tornadic Winds

For non-tornadic winds, the principal concern is still missiles. Because the most significant missile sources are the elevated sheet metal clad structures, it can be argued that the frequency of generating missiles is on the order of 1×10^{-4}/yr. The major target of concern is still the diesel fuel oil transfer pumps as they are completely unprotected from high trajectory missiles. Missiles rolling near the ground should be stopped by the concrete wall surrounding the fuel tank. The service water pumps are protected to the side by the circulating water pumps, and from above by the security structure, which while not substantial should absorb some of the momentum of the potential missiles. However, the principal mitigating factor for both these targets is that they are very small, and particularly with non-tornadic winds there is no clearly preferred direction, so that missiles are likely to be evenly distributed radially about the site. In addition, plant operational procedures require the site to be cleared of debris that could become missiles when a hurricane is expected. Thus the number of unrestrained missiles is small. Based on these arguments, a conditional probability of a missile striking a target of the size of the fuel oil transfer pumps of about 10^{-2} is considered reasonable. Thus again a core damage frequency of 1×10^{-6}/yr is estimated as resulting from non-tornadic winds.

Information Notice 93-53, (NRC, 1993), raised a concern about the potential for non-safety related structures falling into safety related equipment during severe storms. As stated in Section 5.3.4, walkdowns were performed for the severe wind scenarios. These walkdowns identified no issues of concern.

5.3.6 Conclusion of Analysis of Risks from High Winds

The frequency of core damage, from high winds is estimated to be on the order of 2×10^{-6} per year, with equal contributions from tornados and non-tornadic winds. The frequency is judged to be low because of the small size of the targets and the relatively small number of potential missiles. Based on sensitivity studies, the uncertainty range on this frequency is representable by an error factor of 10.

The estimates presented above are considered to be conservative in that no credit has been taken for the possibility of supplying fuel from the Unit 1 fuel tanks, or even from a tanker. The fuel oil transfer pumps can be isolated downstream of any postulated break caused by missiles, and there are several places downstream at which fuel oil could be fed into the system. While the IPE suggested that there is only enough fuel for 91 minutes, this was calculated using a design basis load. Thus there could be a much longer time available for establishing an alternate supply. Such contingencies could be addressed in SAMGs.

5.4 EXTERNAL FLOODS

The topography of the site is such that it ensures that the large scale flooding event from excessive rainfall is virtually impossible, and there are no upstream dams whose failure could cause flooding. The dam creating Lake Robinson has a high water level about 40 feet above the natural grade level, and the maximum lake level is 3 feet below plant grade. Therefore, a more significant concern is flash flooding. There are essentially three issues: ponding on the roofs of critical structures; inflow into the auxiliary building; and backflow through storm drains.

5.4.1 Roof Ponding

Most of the flat roof sections of the control room and the auxiliary building, while they are surrounded by about six inch parapets, have gaps in the parapet for access ladders that allow drainage. There is a small roof section at elevation 254' outside the control room, and a larger section at elevation 262', which have unbroken parapets, and although they are provided with roof drains, those on the 262' elevation were blocked when the walkdown was performed.

The live load allowed for in the design of the auxiliary building roof is 100 psf however, which corresponds to about 19 inches of water, and thus it is concluded that, even if roof ponding were to occur, the height of the parapets is insufficient to hold back enough water to approach the design load.

5.4.2 Water Ingress into the Auxiliary Building

There are several access ways into the auxiliary building which are not water tight. However, the internal flooding analysis (CP&L, 1992) has shown that, for the flooding to cause potential damage, the water level in the auxiliary building has to reach up to the 1 foot mark. Given that the access ways will provide a partial barrier to intrusion, a buildup of more than a foot of water inside the building would require a water level of more than one foot high to be sustained outside the lowest of the access ways, for a considerable time.

The site topography is such that the catchment area of rainfall that flows towards the site is relatively small, the drainage area widens out towards the lake, and there is a three foot drop in grade from the plant to the lake. The 6 hour probable maximum precipitation corresponds to an average of about 5 inches per hour (NOAA, 1978), which strongly suggests that a sustained level of 1 foot or more at the auxiliary building is extremely unlikely, and it is concluded that ingress of water is not a significant hazard.

5.4.3 Backflow Through Storm Drains

From the same arguments as above, for water to accumulate to a depth greater than one foot, via backflow through the drains, the water level outside the building has to be sustained at that level, and that has been argued above to be unlikely.

5.4.4 Conclusions of Analysis of Risk from External Flooding

It is concluded that as a result of design, and of the drainage characteristics of the site, the Robinson plant is not vulnerable to the effects of external flooding, either as a result of prolonged precipitation or of sudden downpours.

5. 5 TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

5.5.1 Aircraft Impact

There are no commercial airways close enough to the plant to be of concern. The Hartsville airport is a private airport which services only single or twin engine aircraft. Because of the orientation of the runway, the relatively low number of airplane operations, and the small size of critical targets, the likelihood of an aircraft crash having a significant impact on the plant is low.

While the Charlotte sectional aeronautical chart (NOAA, 1993) indicates military routes very close to the plant, the military has been requested and has agreed not to use these routes.

5.5.2 Transportation Accidents

The results of a survey conducted in 1983 (NUS, 1984) showed that the toxicity limits of the chemicals then being transported along Highway SC 151, which passes the plant about 0.5 miles to the West, would not be exceeded in the control room. However, a concern was raised by the implementation of a new pickling process at Talley Metals Technology which uses Hydrofluoric acid, a highly volatile and toxic material. An accident to a truck transporting this chemical along SC 151 could be cause for concern. However, the HF is transported in aqueous solution and in that form it is not considered by DOT to be an inhalation hazard.

The railroad spur that runs close to the plant does not carry any through traffic, but is used only to deliver coal to HBR-1, and to take low level radioactive waste and spent fuel away from the plant. This usage does not constitute a hazard for the safe operation of HBRSEP.

5.5.3 Fixed Facility Accidents

Discussions with the local chamber of commerce confirmed that there has been no major development in the area of Hartsville over the last ten years. While, in the UFSAR, both formaldehyde and anhydrous ammonia were identified as being the limiting chemicals from the point of view of the control room habitability, use of the former is being phased out and use of the latter has been discontinued.

A survey of local industries revealed that Talley Metals Industries, located only 1.5 miles west of the plant, has begun using hydrofluoric acid in a new metal pickling process. However, the hydrofluoric acid is to be transported, stored, and used in aqueous solution, and as such does not pose an inhalation hazard should there be an accidental release. Inhalation hazards are of particular concern because there is no automatic detection and isolation capability at Robinson.

5.5.4 Gas Pipeline Accidents

There are several pipelines that are in the vicinity of the plant. An LPG line feeds an internal combustion turbine plant 1.33 miles NNW of the HBR site. By a comparison with a calculation performed for Shearon Harris Nuclear Power Plant, Unit No. 1, it has been determined that this LPG line causes no hazard to the plant either from explosions or from the formation of a combustible cloud. An additional factor of importance is that there is a region of higher ground between the HBRSEP and the combustion turbine plant which acts both as a shield from any blast, and to prevent the heavier-than-air propane drifting towards HBRSEP.

Of more potential concern is the natural gas pipeline that enters the site parallel to the discharge canal. The 8" pipeline terminates at a metering and regulation station which reduces the pressure from 700 to 225 psig and distributes the gas to the onsite IC turbine and the Unit 1 boilers. Natural gas is no longer used for the Unit 1 auxiliary boilers and that distribution line is valved off. Apart from the valve station, which is about 600 feet NE of the critical structures, the pipeline is beneath ground. Line breaks would be alarmed at the Carolina Piping Company's dispatch center and the line isolated manually in about fifteen minutes according to gas company officials. Since natural gas has to be confined to form an explosive mixture, the main threat to the plant is the formation of flammable mixtures at the critical structures. Because there is so little exposed pipeline, and the wind blows to the SW only about ten percent of the time, it is estimated that the frequency of a flammable mixture occurring at the auxiliary building is less than 1E-7 per year. The intakes for the auxiliary building themselves are in the narrow passageway formed by the auxiliary building and the radwaste building. The intake to the control room is to the south of the radwaste building which should make the frequency of a flammable mixture lower than that for the auxiliary building intakes. Because a ruptured gas main is unlikely to go unnoticed, and because of the possibility to take action, such as isolating the ventilation system, the likelihood of core damage resulting from this occurrence is judged to be negligible.

5.5.5 Hydrogen Trailer and Other Onsite Compressed Gas Storage

Hydrogen for the volume control tank is supplied from a trailer containing several pressurized cylinders which is located in the yard to the North West of the auxiliary building by about 100-150 feet. The trailer is tied to anchors by six metal straps. Two possible scenarios can be envisaged: a leak could result in a flame jet, or it could lead to the leaking container or even the trailer becoming a missile. Because the trailer is in the open, a build up of an explosive concentration is not considered possible. Assuming that leakage will most likely occur from the top ends of the cylinders, the orientation of the trailer is such that neither possibility would impact the auxiliary building. There is therefore no identifiable initiating event associated with this accident.

On April 30, 1995, a CO_2 cylinder, stored in the Unit 1 Compressed Gas Shed, ruptured. This cylinder was a 5 pound compressed gas pilot control cylinder for the emergency diesel generator CO_2 system. The cylinder was stored in a cabinet made of welded angle iron and expanded metal. It came loose from its confinement, hit a hydrogen cylinder and knocked it loose from the header to which it was attached, and started a hydrogen fire. On investigation it was found that the cylinder had been grossly overfilled and, in addition, there were three rupture discs installed in place of the one required by design. The implication of the investigation is that the event occurred due to the lack of proper controls for filling cylinders.

The implication for the IPEEE is that cylinder explosions can occur, and can produce projectiles. However, since the only targets of concern are far removed from the Unit 1 Gas Shed, from a geometric argument, coupled with a low frequency of the event, and the fact that there are no components whose failure would cause an initiating event, such events are not considered to be significant. However, CO_2 bottles are also used in the fire protection system. There are racks of cylinders outside the diesel generator rooms. Again though, there is no reason to suppose that there would be an initiating event even if the cylinders were to penetrate the re-inforced concrete walls of adjacent rooms.

CP&L is planning to institute revisions to procedures to avoid the use of overfilled cylinders, and to review all the fire protection system cylinders that are in service to ensure that they are within the maximum allowable loading limit. In addition there is a plan to purchase two new 5 pound compressed gas pilot control cylinders for the "A" and "B" diesel generator CO_2 system to replace the current equipment. With these improvements, the likelihood of a recurrence of such an event is greatly decreased and is of very low frequency.

5.6 DAM FAILURES

Lake Robinson provides both the cooling water and service water supply for the HBRSEP. A complete failure of the dam is therefore equivalent to a complete loss of service water. The loss of service water as an initiating event is discussed in the IPE where it is estimated as having a frequency of 1.1×10^{-3}/yr. There are two recovery actions which are essential to preventing core damage, namely maintaining secondary side cooling with the turbine driven AFW pump in its self cooling mode and with makeup to the CST from the deep well pumps, and providing cooling for the charging pump seals from the fire water system. However, the lake is also the source of water for the fire water pumps, thus one of the essential actions discussed above would not be available in case of a sufficiently catastrophic dam failure. However, there would be enough time available to take the actions described in Operating Procedure OP-801 to run hoses from a Darlington County fire hydrant, or to take suction from the discharge canal.

The statistics on dam failures are difficult to obtain, but one source (Baldewitz, 1984) quotes a failure rate of 2.6×10^{-5}/yr for the period 1870-1980. The same source quotes overtopping and first-fill failures (i.e., failures during the initial filling of the reservoirs) as being the most common. The dam has been in place for more than thirty years and is inspected every five years. The 1990 review report indicates no precursors to any of the common failure modes of earth dams (Van der Leeden, 1990), and has many of the features considered to be appropriate measures against common causes of dam failure. There are procedures in place to maintain the lake level such that overtopping is not a concern, and visual inspections are carried out at one to two month intervals, and remedial work preformed as necessary.

Therefore, the following factors reduce the probability and potential consequences of failure of the Lake Robinson Dam.

- Offsite power remains available, at least for some extended period of time, so power is available for alternate equipment and operator actions.

- The dam is expected to have a lower failure frequency than the generic estimate because:

  1) The dam was constructed to high standards based on its safety importance,

  2) The dam is inspected often and maintained to high standards,

  3) The major dam failure mechanisms, such as overtopping, have a low potential for occurrence based on the design, procedures, and level monitoring, and

  4) The dam has been in place for thirty (30) years, so "first fill" failures do not contribute.

Furthermore, the generic frequency of dam failure is much less than the loss of service water initiator already analyzed. Therefore, based on these considerations, it is judged that core damage resulting from a catastrophic dam failure is a low frequency event, and is not a significant concern for HBRSEP.

5.7 CONCLUSIONS OF ANALYSIS OF RISK FROM OTHER EXTERNAL EVENTS

The analysis of other external events identified some unique plant-specific features that warranted additional analysis; specifically, a rupture of the natural gas pipeline, failure of the dam and the potential susceptability of the diesel fuel oil transfer pumps to wind generated missiles. However, neither of these events is judged to significantly impact the core damage frequency. The core damage frequency from high winds is estimated to be $2x10^{-6}$ per year. This potential scenario could result from missile damage to the diesel fuel oil transfer pumps.

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Figure 5.1-1

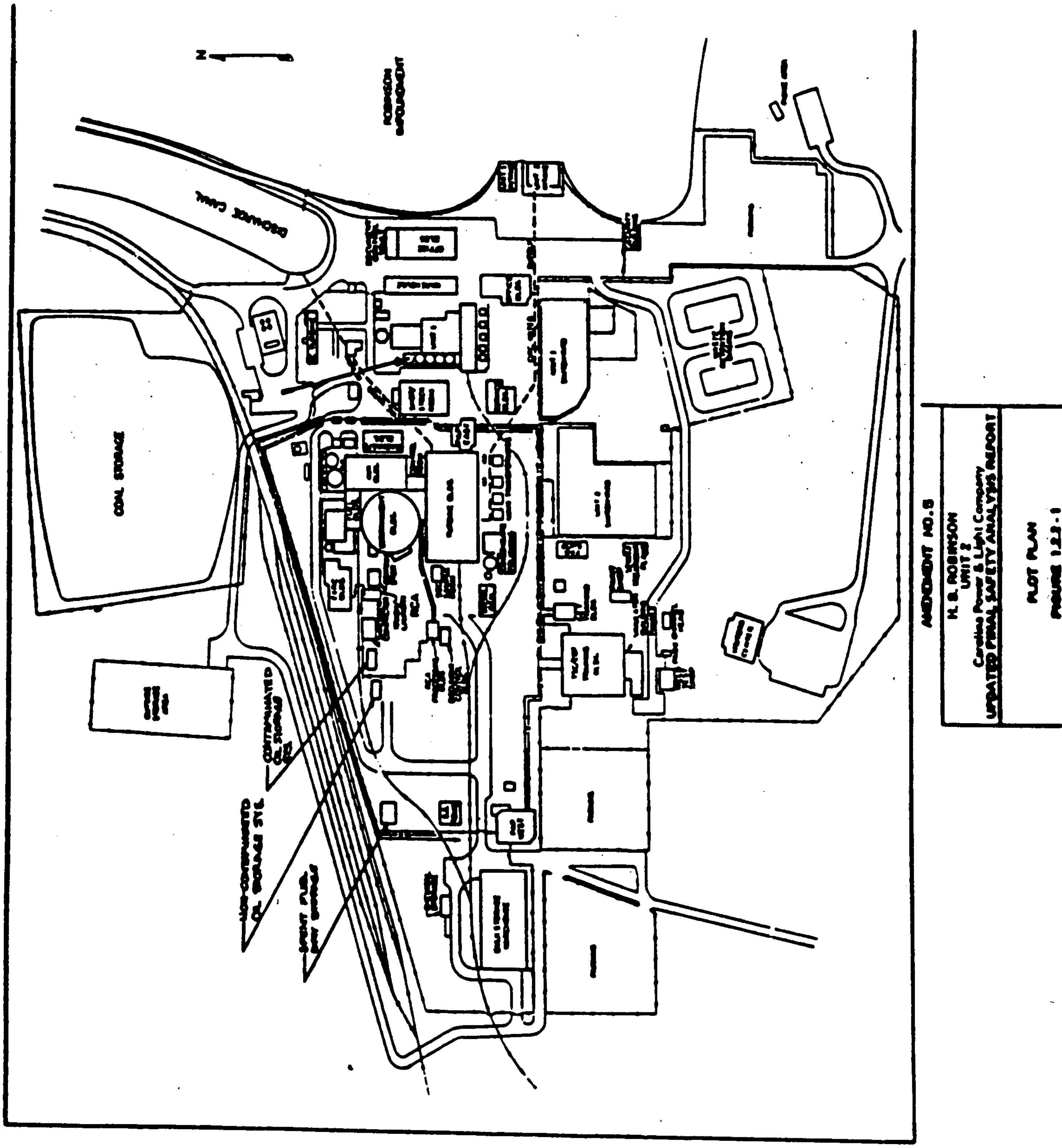
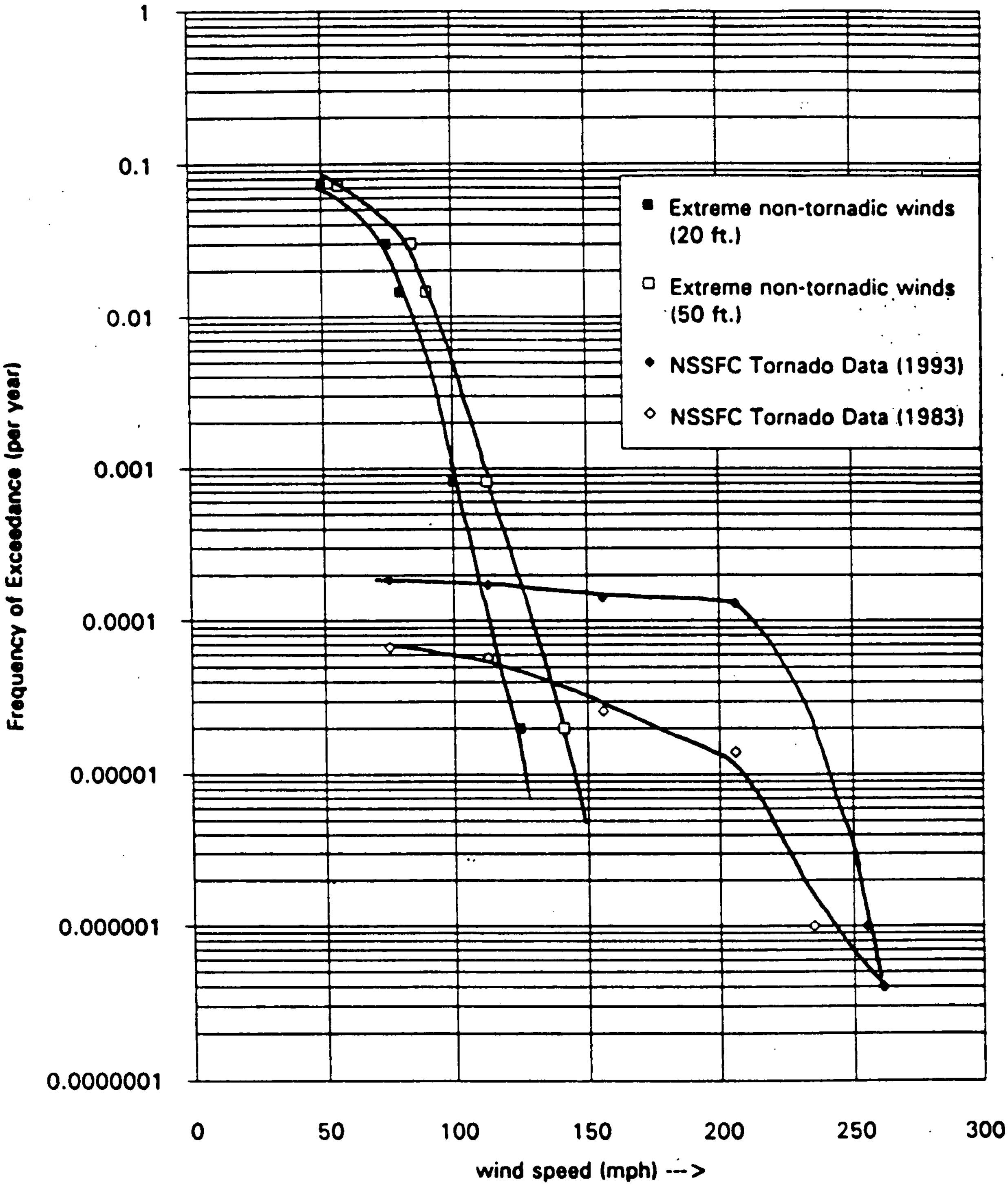

COAL STORAGE
DISCHARGE CANAL
ROBINSON IMPOUNDMENT
CONTAMINATED OIL STORAGE STG.
NON-CONTAMINATED OIL STORAGE STG.
SPENT FUEL DRY STORAGE
AMENDMENT NO. 5
H. B. ROBINSON
UNIT 2
Carolina Power & Light Company
UPDATED FINAL SAFETY ANALYSIS REPORT
PLOT PLAN

Figure 5.3-1

Wind Hazard Curves



SECTION 6

LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 IPEEE PROGRAM ORGANIZATION

Management of the overall IPEEE project at CP&L was provided by Rudy Oliver, Manager - PWR Safety Analysis, who is responsible for all PRA related work performed for the two PWR plants owned and operated by CP&L.

Responsibility for the technical aspects of the project was divided by technical area. Ron Knott and Steve Bostian were responsible for the seismic analysis, and Issa Zarzar and Neil Johnson were responsible for the fire analysis and the analysis of Other External Events.

6.1.1 Seismic Analysis

The seismic review was performed through the efforts of a CP&L project engineer and a site project manager. The project engineer facilitated the completion of engineering activities while the project manager provided for effective plant interface. The project engineer and project manager worked closely to coordinate site walkdowns, implement repairs and plan for modifications. CP&L's consultants for engineering activities were as follows:

- EQE International, Inc. (EQE),
- Science Applications International Corporation (SAIC), and
- Vectra Technologies, Inc.

The seismic analysis required various organization structures depending on the task being performed. A summary of the various responsibilities by task is addressed below.

<u>Safe Shutdown Equipment List (SSEL) Development:</u>

The SSEL development was facilitated by a preliminary walkthrough by SAIC, CP&L and EQE personnel to search for potential low seismic capacity components. The supporting information for SSEL development was completed by Michele Laur (SAIC) and reviewed by CP&L.

<u>Seismic Walkdowns and Reviews:</u>

A seismic review team (SRT) was assembled following the guidance provided in EPRI NP-6041 drawing on the experience and expertise of EQE and CP&L personnel.

Each walkdown team included a minimum of two SRT members who had completed the Seismic Qualification Utility Group (SQUG) Walkdown Screening and Seismic Evaluation training course, as well as EPRI's add-on training for IPEEE. Joint walkdown teams generally consisted of at least one EQE Engineer and at least one CP&L Engineer. Component screening and HCLPF analysis candidate selection was performed jointly between CP&L and EQE. HCLPF calculations were performed by EQE and reviewed by CP&L.

Relay Evaluation:

The relay evaluations were primarily performed by SAIC. However, early involvement was provided by operations personnel. The relay consequence review was performed by Joel Lewis, a CP&L employee at HBRSEP.

Peer Review:

The seismic peer review was performed by Vectra Technologies, Inc. CP&L and EQE supported the peer reviews by participating in the site walkdown review and providing responses for reviewer questions.

6.1.2 Fires and Other External Events

The analysis was, for the most part, performed by NUS at their Gaithersburg office. In order to ensure that CP&L personnel are fully conversant with the IPEEE methods and are in a position to fully integrate the knowledge gained from performing the work into operating procedures, training programs and appropriate hardware changes, a cognizant CP&L engineer was appointed to be the point of contact throughout the study. CP&L engineers performed a review of each of the separate analyses that make up the study. These areas were:

- Qualitative fire area screening analyses,
- Fire frequency analyses,
- Deterministic fire modeling assumptions,
- Fire induced accident sequence analyses,
- Human reliability/recovery action analyses, and
- Other external events analyses.

In addition, CP&L engineers performed the quantification of the conditional core damage probabilities for the various plant damage states that were identified during the course of performing the fire analysis.

6.2 COMPOSITION OF THE INDEPENDENT REVIEW TEAM

A review team considered the final results of the IPEEE analysis in order to assess potential vulnerabilities, evaluate alternatives to address them and recommend actions to resolve severe accident issues using the NUMARC closure guidelines. The composition of the team is shown below. The depth of experience of the plant operations, training and nuclear engineering personnel assigned to this team ensured adequate understanding and appropriate disposition of the issues raised by the IPEEE.

Corporate Support

Fred A. Emerson, Director - Regulatory Affairs, NEI and INPO
Neil Johnson, Engineer - Risk Assessment
Ron Knott, Principal Engineer - Nuclear Engineering
Steve Laur, Project Engineer - Risk Assessment
Rudy E. Oliver, Manager - PWR Safety Analysis

HBRSEP Staff

Don Dyksterhouse, Robinson Engineering Support
Al Garrou, Robinson Regulatory Affairs
Richard Hightower, Robinson Engineering Support
Jan Kozyra, Robinson Regulatory Affairs
Bill Stover, Operations
Jim Townsend, Robinson Engineering Support

6.3 AREAS OF REVIEW AND MAJOR COMMENTS

The IPEEE analysis was reviewed by plant and corporate personnel individually and in meetings, teleconferences, and video conferences. The review considered the methodology, the analysis assumptions, and the results. Scenarios that were estimated to represent a core damage frequency of 1 x 10^{-6} per year or greater were reviewed in detail.

The major review comments are discussed in the following sections. All of these comments have been addressed; resolution of these comments is discussed in Section 6.4.

6.3.1 Seismic Analysis

The seismic portion of the review focused on two technical issues: Liquefaction of site soil and the potential for ductile iron motor operated valve yokes to fail. The liquefaction HCLPF was determined by EQE to be .3g, which is not considered a vulnerability. It was noted that there may be some very localized areas that would experience liquefaction at smaller g values. The low bearing capacity of the site soil was considered in the plant design. Therefore, pile foundations were used for safety-related structures. Isolated liquefaction in the lower strata was judged not to adversely impact safety-related buildings. Tanks mounted on surface foundations were determined not to be affected by liquefaction.

Failure of MOVs with ductile iron yokes was identified during the seismic analysis. MOV 750 and 751 are the only valves that are required for cold shutdown that fall into this category. An analysis was performed to evaluate the potential failure of these valves in two scenarios. First, failure of the yoke could result in the inability to open one or both of the valves. The other scenario evaluated was the possibility of a seismically induced ISLOCA. This would require both of the valves to fail, connecting the low-pressure RHR piping to the RCS.

There are a number of housekeeping changes and minor modifications that were addressed as either part of the IPEEE or the A-46 study. During the review, it was noted that the IPEEE submittal should include a summary of these for completeness. It was also decided that the summary should refer to the schedule for completing the A-46 fixes.

It was also noted in the review that no "low-ruggedness" relays were identified at HBRSEP. It was decided that this plant strength should be documented in the IPEEE submittal.

6.3.2 Fire Analysis

This review of the HBRSEP IPEEE for fires generated general comments on the fire analysis and specific comments for those scenarios identified as having a core damage frequency of greater than 1 x 10^{-6} per year. These are discussed below.

General Comments

It was noted during the review that the assumptions underlying Appendix R and the IPEEE fire analysis are different. For example, the IPEEE analysis considered random failures in addition to equipment failed as the result of the fire itself; Appendix R does not require this. It was decided that Section 4 would be clarified to include a summary of the differences in methodology between the deterministically driven Appendix R analyses and the probabilistic IPEEE fire analyses.

The reviewers noted that the resolution of IE Notice 94-12 should be documented in the IPEEE submittal. Resolution of this IE Notice will be documented in Section 3.

The other major general comment was that the plant and its procedures were not static but changed over time. It was pointed out that the IPEEE analysis represents a "snap-shot" as of the end of refueling outage 15. The plant configuration and procedures as of that time were used for the IPEEE fire analyses. This will be documented.

Significant Fire Scenarios

The fire scenarios that were determined to be significant (CDF above 1 x 10^{-6} per year) were reviewed in detail. As a result of the review comments and additional information gathered during the review process, some of these scenarios were re-analyzed. The review comments are summarized in Table 6-1. Key points in the review are included with under each scenario.

6.3.3 Other External Events Analysis

The review of this area of the IPEEE analysis generated several general comments and some specifically focused on the scenarios that were estimated to have a core damage frequency above 1 x 10^{-6} per year. The general comments are presented first in the discussion below.

The major comment was that the RWST did not appear to be addressed in the analysis. The RWST was mentioned in the write-up, however. The reason for not analyzing the RWST was that other equipment was deemed to be higher contributors to core damage than failure of the RWST, which must occur in conjunction with another failure in order to lead to core damage. Since the higher probability scenarios were estimated to be less than 1 x 10^{-6} per year, the RWST should not present a vulnerability with respect to external events. It was decided that the analyst would verify that the scenarios considered were indeed the higher probability ones and that the IPEEE submittal would be clarified on the subject of the RWST.

Another general comment was that a probabilistic analysis had been used in the past to justify that door SD-45 (entrance to control room HVAC equipment room) did not need to be protected from missiles. It was decided that a determination would be made whether this was consistent with the IPEEE analysis.

The only external event scenarios that were above 1 x 10^{-6} per year involved the diesel generator fuel oil transfer pumps. There are two scenarios (high winds and tornado) that could result in damage to these pumps from missiles. The reviewers questioned whether adequate credit for cross-tying from the unit 1 diesel fuel oil tanks was taken in the analysis. The possibility of providing procedural direction to cover the event of damage to these pumps was discussed. The fuel oil transfer pumps could be isolated downstream of any postulated break caused by missiles. Several points downstream of the fuel oil transfer pumps are available for supplying oil to the EDG day tanks, including one near the EDG that could be filled by an oil truck. If one or more of the four unit 1 diesel fuel tanks are available, they can supply the unit 2 EDGs downstream of the postulated break as well. These are contingencies that could be addressed by severe accident management guidelines or in plant procedures.

6.4 RESOLUTION OF COMMENTS

Resolution of the review comments is addressed in this section. Note that any actions planned or anticipated to address insights or potential vulnerabilities, including any plant modifications, procedure changes, or future evaluation, are covered in Section 7 of this report.

6.4.1 Seismic Analysis

The potential failure of MOVs with ductile iron yokes was addressed by an analysis of the impact of seismically induced failure of MOVs 750 and 751. Failure of these valves to open was determined not to be a concern, since the plant can stay in hot shutdown for at least 72 hours. This provides adequate time to effect repairs on the valves. The core damage frequency due to seismically induced ISLOCA was qualitatively evaluated (Section 3.1.8) and determined to not be a vulnerability.

The IPEEE submittal was revised to include a summary of the A-46 changes and the schedule for completing them in Section 3. The revision also noted that the seismic review identified no relays at HBRSEP that were not seismically rugged.

6.4.2 Fire Analysis

The IPEEE submittal document was clarified to include a summary of the differences in methodology between the deterministically driven Appendix R analyses and the probabilistic IPEEE fire analyses. Resolution of IE Notice 94-12 was appropriately referenced in the submittal, Section 4, since this IE Notice deals with fire as well as seismic issues. The date for plant and procedure configuration used in the fire analysis, which corresponds to the end of HBRSEP refueling outage 15, was added to Section 4 of the report.

The comment regarding use of the "safety switches" to provide control power to 4kV breakers is addressed as follows. This action is already proceduralized for loss of DC in EPP-26. Adding a reference to this procedure in the dedicated shutdown procedures is being considered. This is discussed further in Section 7.

Resolution of the comments provided for the fire scenarios is presented in Table 6-1. It was noted that any scenarios that contributed greater than 1×10^{-6} to the annual core damage frequency could be evaluated for potential inclusion in the severe accident management guidelines when they are developed.

6.4.3 Other External Events Analysis

The analyst verified that the evaluation of other external events properly addressed the higher probability scenarios and that the RWST was a less important contributor than the ones that screened out. The IPEEE report was modified to include a discussion of the RWST. The probabilistic analysis of door SD-45 was reviewed and there was no conflict between that analysis and the IPEEE. The scenarios involving the diesel generator fuel oil transfer pumps are being evaluated for potential inclusion in the severe accident management guidelines; see Section 7.

Table 6-1
Review of Significant Fire Scenarios

Scenario	Review Comments / Discussion	Resolution of Comments
Fire in EDG Control (damage confined to panel) (1-1)	Bus E2 itself would not be damaged in this scenario. A procedure change could be written to provide for recovery of bus E-2 by pulling fuses and locally operating breakers.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.
Waste Evaporator and Gas Stripper panel fires damage overhead trays: lose Offsite Power to Bus E2 (7-1)	Bus E2 itself would not be damaged in this scenario. A procedure change would be written to provide for recovery of Bus E-2 by pulling fuses and locally operating breakers.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.
BA Evap. Equip. Panel A; BA Evap. Equip. Panel B: lose Offsite Power to Bus E2 (7-2)	Bus E2 itself would not be damaged in this scenario. A procedure change could be written to provide for recovery of bus E-2 by pulling fuses and locally operating breakers.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.
MCC 5 fire confined to the panel (7-6)	Only certain cubicles will result in a non-recoverable loss of the associated bus. In a recent actual fire event, the affected breaker burned and the fire penetrated the cubicle above. The bus tripped due to the fire. However, the bus itself was not damaged and was able to be re-energized in a short period of time following the event. Only certain fire locations in the cabinet would result in a non-recoverable failure of the bus. For the remaining locations, it could be argued that only the affected breaker and all immediately adjacent cubicles would be lost. Recovery of power to the bus and the remaining loads by the operators could be assumed within a reasonable time following the event for those cases. Therefore, the reported number is conservative. Note that a detailed analysis taking credit for this insight would require considerable additional effort without any real benefit in terms of plant safety.	The IPEEE report was revised to include a note that the analysis was conservative; however, the analysis was not revised.

Table 6-1
Review of Significant Fire Scenarios

Scenario	Review Comments / Discussion	Resolution of Comments
Fire in MCC-A or MCC-B, with failure of manual suppression, results in loss of train "A" and "B" DC power (16-1)	Credit for manual suppression was not given because of the rapid fire growth. It is possible that some credit could be given if there is substantially less combustible loading in these cabinets than was in the Sandia fire tests. Sealing the conduit penetrations would greatly reduce this CDF, since a fire in either cabinet would self-extinguish.	The combustible loading for MCC-A or -B that would ensure the battery room temperature due to a fire does not exceed that required to fail the battery chargers was determined by analysis to be 1.33×10^5 BTUs, equivalent to approximately 16 lbs of cable insulation. The actual combustible loading in these cabinets was estimated to exceed this amount, so that the original fire CDF remains unchanged.
Battery Rack "A" fire damages overhead cable tray and conduit (16-2)	This scenario takes about 10 minutes to develop. Manual suppression (based on actual drills) will commence from 6 to 13 minutes. Credit for manual suppression could reduce this value by half.	This scenario was re-analyzed, giving credit for manual suppression. As a result, this scenario screened out.
Battery Rack "B" fire damages overhead cable tray and conduit (16-3)	This scenario takes about 10 minutes to develop. Manual suppression (based on actual drills) will commence from 6 to 13 minutes. Credit for manual suppression could reduce this value by half.	This scenario was re-analyzed, giving credit for manual suppression. As a result, this scenario screened out.
Fire in open/ventilated cabinet, Halon system and manual suppression fails, fire damage to all functions served by cables in zone. (19-2)	The analysis took credit for automatic suppression. Drill times for manual suppression would allow some credit for manual suppression. The alarm function of the detectors are truly redundant.	This scenario was re-analyzed, giving credit for manual suppression.

Table 6-1
Review of Significant Fire Scenarios

Scenario	Review Comments / Discussion	Resolution of Comments
Aux. Relay Panel B fire confined to cabinet (19-4)	This scenario credits Halon suppression to prevent the fire from damaging overhead cables or other cabinets.This is confined to the panel itself, so no suppression could be credited. Use of the safety switches to restore control power and allow restoration of offsite power to the DS bus would reduce this scenario to below 1E-6.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.
Fire in Auxiliary Relay Panel "D" (19-6)	Similar to 19-4.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.
Aux Relay Panel M fire confined to cabinet (19-14)	Similar to 19-4.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.
ERFIS MUX 2 Panel fire confined to cabinet (19-16)	The question was raised as to why an ERFIS MUX cabinet could be significant.	It was clarified that the ERFIS MUX cabinet contains cable termination that could fail as a result of the fire.
Fire in Bus E2 with successful AFSS (20-17)	Only certain cubicles will result in a non-recoverable loss of the associated bus. In a recent actual fire event, the affected breaker burned and the fire penetrated the cubicle above. The bus tripped due to the fire. However, the bus itself was not damaged and was able to be re-energized in a short period of time following the event. Only certain fire locations in the cabinet would result in a non-recoverable failure of the bus. For the remaining locations, it could be argued that only the affected breaker and all immediately adjacent cubicles would be lost. Recovery of power to the bus and the remaining loads by the operators could be assumed within a reasonable time following the event for those cases. Therefore, the reported number is conservative. Note that a detailed analysis taking credit for this insight would require considerable additional effort without any real benefit in terms of plant safety.	The IPEEE report was revised to include a note that the analysis was conservative; however, the analysis was not revised.

Table 6-1
Review of Significant Fire Scenarios

Scenario	Review Comments / Discussion	Resolution of Comments
Fire in MCC 6 confined to cabinet (20-21)	This area has auto suppression, but that does not help for fire inside the cabinet. Also, note that only certain cubicles will result in a non-recoverable loss of the associated bus. In a recent actual fire event, the affected breaker burned and the fire penetrated the cubicle above. The bus tripped due to the fire. However, the bus itself was not damaged and was able to be re-energized in a short period of time following the event. Only certain fire locations in the cabinet would result in a non-recoverable failure of the bus. For the remaining locations, it could be argued that only the affected breaker and all immediately adjacent cubicles would be lost. Recovery of power to the bus and the remaining loads by the operators could be assumed within a reasonable time following the event for those cases. Therefore, the reported number is conservative. Note that a detailed analysis taking credit for this insight would require considerable additional effort without any real benefit in terms of plant safety.	The IPEEE report was revised to include a note that the analysis was conservative; however, the analysis was not revised.
RTGB "A" cabinet fire confined within the cabinet - no control room evacuation (22-1)	The only practicable way to reduce this CDF would be to install new, state-of-the-art fire detection systems (incipient fire detection), which are very costly. Note that the control room fires experienced in the industry have all been rapidly extinguished. There is no cost-beneficial way to improve the CDF.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.
RTGB "A" cabinet fire with no suppression prior to propagation to the other RTGB cabinets - control room evacuation assumed due to extent of damage. (22-2)	The only practicable way to reduce this CDF would be to install new, state-of-the-art fire detection systems (incipient fire detection), which are very costly. Note that the control room fires experienced in the industry have all been rapidly extinguished. There is no cost-beneficial way to improve the CDF.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.

Table 6-1
Review of Significant Fire Scenarios

Scenario	Review Comments / Discussion	Resolution of Comments
RTGB "D" cabinet fire suppressed within the cabinet - control room evacuation assumed due to extent of damage (22-3)	The only practicable way to reduce this CDF would be to install new, state-of-the-art fire detection systems (incipient fire detection), which are very costly. Note that the control room fires experienced in the industry have all been rapidly extinguished. There is no cost-beneficial way to improve the CDF.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.
RTGB, "B", "C", "D" or "E" cabinet fire with no suppression prior to propagation to the other RTGB cabinets - control room evacuation assumed due to extent of damage (22-4, -6, -8)	The only practicable way to reduce this CDF would be to install new, state-of-the-art fire detection systems (incipient fire detection), which are very costly. Note that the control room fires experienced in the industry have all been rapidly extinguished. There is no cost-beneficial way to improve the CDF.	Evaluate for potential inclusion in the severe accident management guidelines when the guidelines are developed.

Table 6-1
Review of Significant Fire Scenarios

Scenario	Review Comments / Discussion	Resolution of Comments
480 V Bus 3/2B fire damages overhead cable tray and conduit (25-1)	Only certain cubicles will result in a non-recoverable loss of the associated bus. In a recent actual fire event, the affected breaker burned and the fire penetrated the cubicle above. The bus tripped due to the fire. However, the bus itself was not damaged and was able to be re-energized in a short period of time following the event. Only certain fire locations in the cabinet would result in a non-recoverable failure of the bus. For the remaining locations, it could be argued that only the affected breaker and all immediately adjacent cubicles would be lost. Recovery of power to the bus and the remaining loads by the operators could be assumed within a reasonable time following the event for those cases. Therefore, the reported number is conservative. Note that a detailed analysis taking credit for this insight would require considerable additional effort without any real benefit in terms of plant safety. Also, it was noted that drill times for manual suppression range from 4 - 14 minutes; this is enough to justify credit for manual suppression. However, the analysis would be complicated because the impact of those loads damaged in the cabinet itself (other targets saved because of suppression) would still have to be evaluate.	The IPEEE report was revised to include a note that the analysis was conservative; however, the analysis was not revised.

Table 6-1
Review of Significant Fire Scenarios

Scenario	Review Comments / Discussion	Resolution of Comments
Explosive yard transformer fire results in fire propagating to the turbine building with LOSP and loss of DS system (26-1, 26-2)	An explosive transformer fire in one phase of the main transformer (closest to the auxiliary transformer), in the auxiliary transformer, or in the startup transformer was postulated to cause loss of offsite power and thick smoke at the dedicated shutdown panels in the 4KV switchgear room. Additionally, the explosive fires in the auxiliary or main transformer were assumed to damage the dedicated shutdown power cabling that runs in conduit along the outside of the turbine building about 21 feet from the fire. Based upon expert judgment by the HBRSEP fire protection engineer, damage to the dedicated shutdown cabling is not likely. Transformers do not explode in such a way that spews burning oil at great distances. An explosive failure is usually the result of degradation over time followed by instantaneous arcing and failure. The degradation over time results in production of hydrogen and carbon monoxide inside the oil container; the arcing ignites these flammable gases and ruptures the oil container. The oil container ruptures in a splitting or tearing fashion as opposed to an explosive one. The oil is ignited on its way out by the arcing or from the hydrogen and carbon monoxide explosion itself. The oil may then spread or be washed by deluge water. It was noted that the transformers are N2 blanketed (since about 1991 time frame) and that the insurance company has required increased transformer surveillance. The result of these measures may be a trend towards fewer transformer failures as time goes on. The gravel-filled pit below the auxiliary and startup transformers would capture some portion of the oil; since this pit is not pumped, it is not clear how much volume would be available to contain the oil. The slope of the ground is away from the DS cabling; oil would run toward a storm drain away from this target. The storm drain then slopes toward a settling pond, with no intervening drains or openings.	Based on the review comments, scenarios involving an explosive transformer fire were re-evaluated. Scenario 26-1 became two scenarios: 26-1 and 26-2. The original 26-2 was renumbered 26-3. It was discussed that an action could be added to the fire pre-plans for fires in this area to direct water on the DS bus conduit to keep it cool; alternately, a radiant heat shield to mitigate the damage to the DS conduit could be evaluated.

Table 6-1
Review of Significant Fire Scenarios

Scenario	Review Comments / Discussion	Resolution of Comments
(26-1, 26-2 continued)	The dedicated shutdown cabling is inside conduit. It runs relatively low along the building for the portion of its length closest to the postulated transformer fire. Because of the distance of the cable from the transformers, its low height, and the typical shape of the fire plume, it was remarked during the review that it should be unlikely that the DS cable would be damaged. However, further analysis of the radiant heat from a burning auxiliary or startup transformer indicated the possibility of damaging the DS cabling inside the conduit. It was noted that steps could be added to fire fighting procedures to isolate the deluge water supply to prevent transport of burning oil and to direct cooling of the DS cabling by the fire brigade. It was also suggested that the pit could be monitored and pumped out periodically as necessary. It was questioned whether fire fighters would be able to get close enough to direct water on the DS bus conduit. The HBRSEP fire protection engineer stated that the fire fighters could cool the DS conduit in the event of a yard fire, even if they had to take a vantage point outside of the security fence on the unit 1 side (slightly east and south of the cable). They could deploy either an unmanned master stream device or a manned nozzle location that could get water on the side of the turbine building and on the conduit itself. He said that several hundreds of gallons of water a minute could be directed to the wall and conduit to provide cooling.device or a manned nozzle location that could get water on the side of the turbine building and on the conduit itself. He said that hundreds several of gallons of water a minute could be directed to the wall and conduit to provide cooling.	

Table 6-1
Review of Significant Fire Scenarios

Scenario	Review Comments / Discussion	Resolution of Comments
Start Up Transformer fire results in LOSP (26-3)	The plant experienced an actual loss of the startup transformer several years ago. The plant started immediately to back feed through the main transformer to supply offsite power to the site (both EDGs and the DS diesel were running and carrying loads). Back feeding is proceduralized. Credit can be taken for back feed of the main transformer within 8 hours to recover the "fail to run" diesel events in the cutsets for this scenario.	Credit for restoration of offsite power by back feeding through the main transformer was used in the re-analysis of this scenario.
Oil fire associated with SW pump B or C (29-3)	There is about 6 gallons of oil in each SW pump; this is enough to fail both pumps given a fire (as assumed in the analysis). There is a security camera that would allow earlier detection than credited in the analysis, however.	The presence of the security camera and the fact that it would provide detection capability was documented in the IPEEE report. The scenario was not re-evaluated.

SECTION 7

PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

7.1 PLANT IMPROVEMENTS

As described in Section 6.3, a multi-disciplinary team was established to evaluate the IPE results and suggest areas for potential improvements. The results of the evaluation are summarized in Section 8.2.1.

CP&L will continue the evaluation of identified IPEEE scenarios to determine whether any cost-effective measures exist which could reduce the contribution of those sequences to the IPEEE annual core damage frequency. Measures to be evaluated will encompass potential procedural changes, minor hardware changes, and consideration during development of the H. B. Robinson Steam Electric Plant (HBRSEP) Severe Accident Management Guidelines. CP&L will submit the results of these evaluations and any planned actions in a supplement to this submittal by November 30, 1995.

7.2 UNIQUE PLANT FEATURES IMPORTANT TO SAFETY

As noted in previous sections, the HBRSEP IPEEE provided insights concerning important plant features. As discussed in Section 7.1, efforts are underway to determine improvements for some areas; other areas will be considered for inclusion in the severe accident management guidelines. A summary of IPEEE insights and important plant features is provided below.

7.2.1 Seismic Analysis

The seismic analysis revealed no vulnerabilities for HBRSEP from seismic events. Several insights were gained as a result of the analysis, including:

- Liquefaction lenses may occur in isolated areas but do not impact safety related structures,
- Several seismic interactions exist which will be corrected by the installation of additional bracing, and
- The susceptibility of ductile iron valve yokes to failure from a seismic event was noted for two motor-operated valves: RHR 750 and 751.

7.2.2 Fire Analysis

The fire analysis re-enforced the importance of the dedicated shutdown diesel in mitigating the effects of postulated fires at the plant. The single battery room and the physical configuration of the yard transformers each had a significant effect on the analyses.

Most sequences leading to core damage as a result of a fire were found to be caused by reactor coolant pump seal LOCAs.

A number of sequences were identified that provide insights that will be considered for inclusion in the HBRSEP severe accident management guidelines when they are developed.

7.2.3 Other External Events Analysis

The potential for damage to the diesel generator fuel oil pumps from wind-generated missiles was an insight gained as a result of the IPEEE. There are unique site features that should allow recovery of fuel oil to the EDGs, including the availability of a number of cross-connection points downstream of the postulated failure location and the ability to directly pump fuel oil near the EDG day tank from a diesel fuel truck.

SECTION 8

SUMMARY AND CONCLUSIONS

8.1 SUMMARY

8.1.1 Overview of IPEEE

Carolina Power and Light (CP&L) Company has completed an examination of the potential for events external to the plant to cause core damage accidents at its H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. This report describes the results of the examination and documents compliance with the Nuclear Regulatory Commission's Generic Letter 88-20, Supplement 4 "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" (NRC, 1991). This analysis complements the analysis presented in the Individual Plant Examination (IPE) (CP&L, 1992), which addressed internal initiating events. By the performance of this project, CP&L has achieved the four primary objectives of the IPEEE, which were, for initiating events resulting from events external to the plant systems:

> To develop an appreciation of severe accident behavior,
>
> To understand the most likely severe accident sequences that could occur at the plant under full-power conditions,
>
> To gain a qualitative understanding of the overall likelihood of core damage and fission product release, and
>
> If necessary, to reduce the overall likelihood of core damage and fission product release by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

It should be noted that the results of this study are not directly comparable with those of the IPE. The methodology used to perform the IPE was based on a systems analysis approach that has achieved an accepted degree of maturity. The analysis of external initiating events, by contrast, has not reached the same degree of maturity. For example, some of the potentially damaging external initiating events have very low frequencies that cannot be estimated using actuarial data without considerable extrapolation, so that the frequency estimates are subject to a large uncertainty. Many of the events can occur with a range of severity, with the damage potential being a function of that severity. Analyzing the impact of such events can be very complex and time consuming. Because of this, the methods that have been developed to analyze the impact of external initiating events are essentially screening analyses, designed either to identify the most significant contributors, while minimizing the need for detailed analyses, or to identify specific weaknesses without explicitly estimating risk.

The method chosen to analyze the impact of seismic events, the Seismic Margin method, is the latter type of analysis. There is no estimation of core damage frequency. Instead, the analysis is an assessment of whether the plant has sufficient margin over and above the design basis to withstand what is known as the Review Level Earthquake (RLE). The analysis of the Other External Events for HBRSEP is for the most part a confirmation that the plant, even though not built to the requirements of the Standard Review Plan (NRC, 1975) criteria, does in fact comply with their intent, and again does not require that core damage frequency be calculated.

The PRA approach adopted for the fire analysis does, on the other hand, result in the evaluation of the core damage frequencies from a set of fire initiated scenarios. However, even in this case, the core damage frequency is not evaluated in the same way as for internal initiating events. The analysis is based on a screening approach, in which fire areas were screened from further consideration when a conservative analysis showed that the frequency of core damage was less than 1.0E-06. However, since for areas that are screened, the analysis is not further refined, the degree of conservatism is not estimated. Therefore, it would be inaccurate to sum the screening core damage frequencies to obtain the overall core damage frequency. Instead, the analysis has been used to identify the scenarios that have the highest likelihood of leading to core damage.

There is an additional difference between the approaches used for the IPE and the fire PRA performed for the IPEEE. The sequences in the IPE were grouped by functional type for screening and for comparison with the Severe Accident Issue Closure Guidelines (NUMARC, 1992). In the fire analysis, sequences were grouped by fire location because it is the vulnerable locations that are of interest.

8.1.2 Results

8.1.2.1 Seismic Margins Assessment

Results of the seismic margins assessment are grouped in three categories as follows:

- Housekeeping/Maintenance Issues

Thirty-three items were identified as outliers requiring minor maintenance that could be repaired by a work ticket. Items that can be repaired by work tickets are typically those items whose conditions have only slightly degraded from the original design intent and can be fixed by using existing plant drawings, procedures, or guidelines. The repair can be implemented by maintenance or construction without any engineering input or review. It usually involves a replacement of like hardware, torquing of bolts, etc. These items are listed in Tables 3-1 and 3-2.

- Repairs/Modifications

Twenty-two items were identified as outliers and the Seismic Review Team determined that additional calculations would potentially not resolve the outlier issues. They concluded that the twenty-two items would best be resolved by the implementation of physical plant modifications. Modifications provide the vehicle to change components using engineering review and input. These components are listed in Table 3-3.

- Raceway Repairs/Modifications

Sixteen issues involving electrical raceway installations were identified as requiring work ticket/maintenance or modifications attention in order to restore the reported item to an acceptable condition.

All 789 relays on the HBRSEP essential relay list have been accepted by either capacity screening or system consequence screening. Twenty items were evaluated using the High Capacity for Low Probability of Failure methodology. These items are identified in Table 5-3 of Appendix A to this report.

8.1.2.2 Fires

In total, twenty-three scenarios that have contributions to core damage frequency greater than 1.0E-6 were identified. They are summarized in Table 8-1. There are five scenarios with contributions to CDF greater than 1.0E-5. They are:

- Scenario 16-1. This is a fire originating in battery room A-16 in MCC-A or MCC-B, with failure of manual suppression, leading to a loss of train A and B DC power. This scenario is significant because the MCCs have several open conduits, and the battery room is small. Therefore, damage to the redundant MCC due to the formation of a hot gas layer is possible in a short time, such that manual supression may not be possible.

- Scenario 20-16. This is a fire originating in the emergency switchgear room that leads to loss of the E2 480 V bus. This fire is significant because offsite power is lost and it has a high initiating event frequency.

- Scenario 22-3. This is a fire in RTGB cabinet D that is suppressed within the cabinet. This is significant because it is a control room fire and requires evacuation.

- Scenario 22-4. This is a fire in RTGB B, C, D or E that propagates to other RTGB cabinets. This is significant because it is a control room fire and requires evacuation.

- Scenario 26-1. This scenario results from an explosive transformer fire in the switchyard that results in a loss of offsite power and the dedicated shutdown diesel generator. The transformers of concern, because of their proximity to a conduit associated with the DS diesel which is routed on the outside of the turbine building, are the auxiliary and start up transformers.

8.1.2.3 Other External Events

One scenario greater than 1E-6 was identified. However, the prolonged operation of the emergency diesel generators, which would in all probability be required following the occurrence of extreme winds in the vicinity of the plant, could be compromised by the fact that the fuel oil transfer pumps are unprotected from misssiles. The day tanks for the diesel generators are of very limited capacity, allowing only about 90 minutes of operation. The frequency of scenarios leading to the simultaneous loss of offsite power and damage to the fuel oil transfer pumps was estimated to be on the order of 2.0E-6 per year.

8.2 CONCLUSIONS

The IPEEE scenarios involving seismic events, fires, and other external events are discussed in Section 8.1. This section summarizes the issues that were identified for further evaluation and summarizes resolution of other issues that are subsumed by performance of the IPEEE.

8.2.1 Issues Identified for Further Evaluation

The results of the IPEEE were evaluated using the guidance presented in the Severe Accident Issue Management Closure Guidelines (NUMARC, 1991). The applicable thresholds from those guidelines used for this evaluation are summarized below. Issues that exceeded these thresholds were considered candidates further evaluation for the purpose of this IPEEE.

1. Threshold: Containment bypass sequences representing a contribution to annual containment bypass frequency of greater than 1×10^{-7} but less than 1×10^{-6}.

   Guideline: Ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure;

2. Threshold: Non-containment bypass sequences representing a contribution to annual CDF of greater than 1×10^{-5} but less than 1×10^{-4} or 20% to 50% of total CDF.

   Guideline: Either 1) Find a cost effective treatment in EOPs or other plant procedure or minor hardware change with emphasis on prevention of core damage, or, 2) ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure; and,

3. Threshold: Non-containment bypass sequences representing a contribution to annual CDF of greater than 1×10^{-6} but less than 1×10^{-5}.

Guideline: Ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.

No seismic vulnerabilities were identified. However, failure of the ductile iron yokes of valves RHR-750 and -751 was determined to be a potential interfacing systems LOCA precursor. This precursor frequency is considered to be less than 1×10^{-6}. Since the potential consequences of an interfacing systems LOCA are significant, this scenario was considered as falling into category (1) above.

Fire scenarios that contributed greater than 1×10^{-6} to the annual core damage frequency are presented in Table 8-1. No scenarios were identified that were greater than 1×10^{-4}. The fire scenarios that contribute greater than 1×10^{-5} to the annual core damage frequency were discussed in Section 8.1.2.2 and belong to category (2) above. The remaining fire scenarios in Table 8-1 contribute between 1×10^{-6} and 1×10^{-5} to the annual core damage frequency fall into category (3) above.

Only one scenario from other external events was identified as above the guideline threshold. The damage to the diesel fuel oil transfer pumps from high winds (tornadic and non-tornadic) was estimated to contribute approximately 2×10^{-6} to the annual core damage frequency. This scenario falls into category (2) above.

The scenarios that fall into the three categories were identified and will be evaluated as appropriate. The Severe Accident Issue Management Closure Guidelines (NUMARC, 1991) will be considered during this evaluation. Section 7 of this report provides CP&L's anticipated actions with respect to these scenarios.

8.2.2 Summary of Resolution of Subsumed Issues

The Eastern U.S. Seismicity Issue is resolved by the seismic part of the IPEEE. Since CP&L exercised the seismic margins option, the resolution was achieved by an appropriate choice of review level earthquake. GI-131 deals with the seismically induced failure of the flux mapping transfer cart that would lead directly to the rupture of instrumentation tubes at the seal table. Since this is applicable to Westinghouse plants only, it is applicable to HBRSEP. It has been addressed in the IPEEE. USI A-46 has subsumed USI A-17, "Seismic Interactions in Nuclear Power Plants". HBRSEP is an A-46 plant, and USI A-17 was addressed through the seismic walkdown that was performed to meet the requirements of the IPEEE. The seismic-fire interaction issues raised in Information Notice 94-12 were addressed in the SMA and are discussed in Section 8 of Appendix B.

The Fire Risk Scoping Study (FRSS) Issues, NUREG/CR-5088, were examined through comparison to standardized checklist questions and through specifically tailored plant walkdowns according to the FIVE Methodology. The FRSS issues are discussed in Section 4.8. The issue of seismic-fire interactions has been addressed and is discussed in Section 3.1.6.

The revised "Design Probable Maximum Precipitation (PMP)" criteria were assessed within the Other External Events Task as requested in Generic Letter 89-22, Supplement 4. The conclusions are presented in Section 5.4.

Information Notice 93-53, Supplement 1 requested that the IPEEE address the lessons learned from the effects of Hurricane Andrew on the Turkey Point Nuclear Generating Station (NRC, 1993). This was addressed during the performance of a walkdown that was conducted to confirm the conclusions of the review of the plant design with respect to Other External Events, as discussed in Section 5.

8.3 REFERENCES

(CP&L, 1993), Carolina Power and Light Company, "H. B. Robinson Steam Electric Plant, Unit No. 2, Individual Plant Examination Submittal", August 1992.

(NRC, 1975), USNRC, "Standard Review Plan for Review of Safety Analysis Report for Nuclear Power Plants", NUREG-75/187, December 1975.

(NRC, 1991), USNRC, Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)", April, 1991.

(NUMARC, 1991), NUMARC, "Severe Accident Issue Closure Guidelines", 91-044, 1991.

TABLE 8-1 SUMMARY OF ALL FIRE SCENARIOS WITH CONTRIBUTION TO CDF > 1.0E-06/YEAR

Fire Zone	Scenario #	Description	Frequency	Total CDF (/yr)	Conservatisms Identified in the Model
A/1 EDG B Room	1-1	Fire in EDG Control (damage confined to panel)	2.62E-03	3.92E-06	No credit was given to recovery of offsite power to bus E-2. If this action to recover offsite to bus E-2 by pulling control fuses and operating breakers manually was proceduralized, the total contribution from this zone could be reduced to < 1E-6/yr.
A/7 Auxiliary Bldg. Hall	7-1	Waste Evaporator and Gas Stripper panel fires damage overhead trays	1.46E-03	1.74E-06	Same as for A1-1
	7-2	Boric Acid Equipment Panel A & B panel fires damage overhead trays	9.74E-04	1.15E-06	Same as for A1-1
	7-6	MCC 5 fire confined to the panel	4.87E-04	6.62E-06	Analysis assumes any fire will result in complete loss of MCC. Experience indicates that fire damage will in fact be confined to a loss of the breaker which initiates the fire and possible breakers in adjacent cubicles. Once fire is extinguished the bus can usually be re-energized
A-16 Battery Room	16-1	Fire in MCC-A or MCC-B, with failure of manual suppression, results in loss of train "A" and "B" DC power	9.74E-04	7.61E-05	If fire seals were provided for MCC conduit penetrations, thus minimizing potential size of fire and limiting damage to individual cabinets, the CDF contribution from the battery room could be reduced by an order of magnitude
A/19 Cable Spreading Room	19-2	Fire in open/ventilated cabinet, Halon system and manual suppression fails, fire damage to all functions served by cables in zone.	1.13E-05	4.24E-06	
	19-4	Aux Relay Panel B fire confined to cabinet.	1.38E-04	1.11E-06	
	19-6	Aux Relay Panel D fire confined to cabinet	1.38E-04	2.19E-06	

TABLE 8-1 SUMMARY OF ALL FIRE SCENARIOS WITH CONTRIBUTION TO CDF > 1.0E-06/YEAR

Fire Zone	Scenario #	Description	Frequency	Total CDF (/yr)	Conservatisms Identified in the Model
	19-14	Aux Relay Panel M fire confined to cabinet	1.38E-04	2.09E-06	
	19-16	ERFIS MUX 2 Panel fire confined to cabinet	1.38E-04	1.54E-06	
A/20 Emergency Switchgear Room	20-13	Fire in any Auxiliary Relay Racks with failure of AFSS	1.32E-05	4.73E-06	
	20-14	Fire in 480v Bus E1 confined to cabinet with successful AFSS	1.71E-03	1.17E-06	Analysis assumes any fire will result in complete loss of MCC. Experience indicates that fire damage will in fact be confined to a loss of the breaker which initiates the fire and possible breakers in adjacent cubicles. Once fire is extinguished the bus can usually be re-energized
	20-16	Fire in 480v Bus E2 confined to cabinet with successful AFSS	1.97E-03	1.37E-05	Same as for scenario 20-14
	20-20	Fire in MCC 6 confined to cabinet	6.42E-04	2.83E-06	Same as for scenario 20-14
A/22 Main Control Room	22-1	RTGB "A" cabinet fire confined within the cabinet - no control room evacuation	5.30E-05	3.59E-06	
	22-2	RTGB "A" cabinet fire with no suppression prior to propagation to the other RTGB cabinets - control room evacuation assumed due to extent of damage.	1.20E-05	4.47E-06	
	22-4, 22-6, 22-8	RTGB, "B", "C", "D" or "E" cabinet fire with no suppression prior to propagation to the other RTGB cabinets - control room evacuation assumed due to extent of damage	4.40E-05	1.64E-05	

TABLE 8-1 SUMMARY OF ALL FIRE SCENARIOS WITH CONTRIBUTION TO CDF > 1.0E-06/YEAR

Fire Zone	Scenario #	Description	Frequency	Total CDF (/yr)	Conservatisms Identified in the Model
	22-3	RTGB "D" cabinet fire suppressed within the cabinet - control room evacuation assumed due to extent of damage	5.60E-05	1.98E-05	
G/25 Turbine Building	25-1	480 V Bus3/2B fire damages overhead cable tray and conduit	1.36E-03	2.77E-06	Analysis assumes all fires will result in complete loss of bus and damage to overhead cable. Experience indicates that fire damage will in fact be confined to loss of the breaker which initiates the fire and possible loss of breakers in adjacent cubicles. Once fire is extinguished bus can usually be re-energized.
G/26 Transformer Yard	26-1	Explosive or other non-suppressed fire in Auxiliary of Start Up Transformer with LOSP and loss of DS system	1.25E-03	2.42E-05	
	26-2	Explosive or other non-suppressed fire in phase 3 main transformer Start Up Transformer results in LOSP with degradation of DS operability due to smoke	4.73E-04	2.67E-06	
	26-3	Start Up transformer fire	2.86E-03	9.76E-06	
G/29 Service Water Pump Area	29-3	Oil fire associated with SW pump B or C	1.60E-04	4.35E-06	Security camera provides means of fire detection not credited in analysis.

APPENDIX A
TO

THE H. B. ROBINSON STEAM ELECTRIC PLANT
UNIT NO. 2

INDIVIDUAL PLANT EXAMINATION FOR
EXTERNAL EVENTS SUBMITTAL

SEISMIC IPEEE

SEISMIC IPEEE
H. B. ROBINSON STEAM ELECTRIC PLANT
UNIT NO. 2

June 1995
Revision 0

Prepared By:

Carolina Power & Light Company
and
EQE International, Inc.

Prepared For:

CAROLINA POWER & LIGHT COMPANY
411 Fayetteville Street
Raleigh, NC 27602

EQE Project Number: 52212

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TABLE OF REVISIONS

Revision No.	Date	Description of Revision
0	June 23, 1995	Original Issue

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLES

FIGURES

TABLE OF CONTENTS (Continued)

APPENDICES

1. INTRODUCTION AND METHODOLOGY SELECTION

In the Commission policy statement on severe accidents in nuclear power plants issued in 1985, the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on any regulatory requirements for these plants. However, the Commission recognized that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements. In 1988 the Commission requested that each licensee conduct an individual plant examination (IPE) for internally initiated events including internal flooding. Many PRAs indicated that, in some instances, the risk from external events could contribute significantly to core damage.

In July 1990, following public comments and a workshop, the Commission issued Supplement 4 to Generic Letter 88-20 (Reference 1) requesting that each licensee conduct an individual plant examination of external events (IPEEE). The general objectives of the IPEEE are similar to that of the IPE - that is, for each licensee (1) to develop an appreciation of severe accident behavior, (2) to understand the most likely severe accident sequences that could occur at its plant under full-power operating conditions, (3) to gain a qualitative understanding of the overall likelihood of core damage and fission product releases, and (4) if necessary, to reduce the overall likelihood of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

The staff has concluded that five external events need to be included specifically in the IPEEE: seismic events, internal fires, high winds, floods, and transportation and nearby facility accidents. This report addresses seismic events.

Acceptable methodologies for performing the seismic IPEEE are summarized in NUREG-1407 (Reference 2). This evaluation may be conducted by performing a seismic PRA or a Seismic Margins Assessment (SMA). The SMA methodology was designed to demonstrate sufficient margin over the Safe Shutdown Earthquake (SSE) to ensure plant safety and to find any "weak links" that might limit the plant shutdown capacity to safely withstand a seismic event larger than the SSE or lead to seismically induced core damage. The SMA may in turn be performed using the methodology developed by Lawrence Livermore National Laboratories (LLNL), or by Electric Power Research Institute (EPRI). Carolina Power & Light (CP&L) has opted to perform an SMA using the EPRI methodology (Reference 3).

Robinson was placed in the full-scope category for margin assessment. The basic information used was the 1989 Lawrence Livermore National Laboratory seismic hazard estimates for nuclear power plant locations in the eastern United States (Reference 4) and the EPRI hazard study (Reference 5).

New seismic hazard data were published in October, 1993 that demonstrates that the seismic hazard at existing eastern United States nuclear power plants is much less than what the NRC staff originally believed (Reference 6). The data demonstrate that the annual probability of Robinson exceeding the 0.2g design basis earthquake based on the 1989 LLNL mean hazard curves is roughly equal to the annual probability of exceeding the 0.3g review level earthquake (RLE) based on the 1989 LLNL hazard curves.

Based on the revised hazard curves, the NRC issued draft Supplement 5 to Generic Letter 88-20 (Reference 7) to ease seismic IPEEE requirements for plants that were placed in the focused-scope category. The supplement does not significantly impact full-scope plants such as Robinson. Therefore, the Robinson seismic IPEEE program follows the EPRI seismic margins methodology in accordance with Generic Letter 88-20 (Reference 1) and NUREG 1407 (Reference 2) for a full-scope plant, without exception.

Detailed plant walkdowns are considered the most cost-effective and beneficial aspect of the SMA program. Combined A-46 and IPEEE walkdowns were performed by teams of CP&L and consultant Seismic Review Teams (SRTs) in accordance with the Seismic Qualification Group (SQUG) Generic Implementation Procedure (GIP) (Reference 8), with enhancements based on EPRI NP-6041. Pre-walkdown activities included prescreening of success path components with available data entered into EHOST, a microcomputer database developed by EQE International. Walkdowns were performed using pen-based computers and the program EWALK. EQE proprietary software EWALK is compatible with EHOST to facilitate efficient data collection and subsequent data management.

These walkdowns identified issues that will result in cost effective improvements as a result of the SMA program. Analyses to determine high confidence of low probability of failure (HCLPF) capacity of selected success path elements confirmed that the equipment and subsystems HCLPF is generally greater than or equal to the 0.3g Review Level Earthquake (RLE).

2. REVIEW OF PLANT INFORMATION

A brief description of the general plant, ground response spectra, structures, equipment, and distribution systems is presented below. Information presented in this section is contained in existing plant licensing documents including the Updated Final Safety Analysis Report (UFSAR). The purpose of this section is to provide a review of the plant design.

2.1 GENERAL PLANT DESCRIPTION

The Nuclear Steam Supply System (NSSS) for the unit is a pressurized light water moderated and cooled reactor (PWR) consisting of three closed reactor coolant loops connected in parallel to the reactor vessel each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to the hot leg of one of the loops. The NSSS, along with the design and fabrication of the initial fuel core, is supplied by Westinghouse Electric Corporation.

The Containment is a steel lined reinforced concrete structure in the form of a vertical right cylinder with a hemispherical dome and a flat base with a recess beneath the reactor vessel. The containment is designed by Ebasco Services Incorporated, the architect/engineer for Robinson.

The unit is designed for a licensed power output of 2300 Megawatts thermal (Mwt). All steam and power conversion equipment, including the turbine generator, was designed to permit a generation of 769 Megawatts electric (Mwe) gross.

2.1.1 Site Location and Description

The site is in northeastern South Carolina. The Robinson Plant is located in northwest Darlington County, South Carolina, approximately 3 miles west-northwest of Hartsville, South Carolina.

The plant is located in the Coastal Plain physiographic province, approximately 15 miles southeast of the Piedmont province. Topography of the region is characterized by rolling sand hills interspersed with water courses. Surface water drains to Lake Robinson. The region is not subject to severe persistent inversions.

Robinson is located on the southwestern corner of Lake Robinson, a cooling impoundment of Black Creek, to furnish cooling water for power plants at the site. Figure 2-1 shows the Robinson plant site boundary and exclusion zone.

Farming is the predominant activity in the sparsely populated immediate environs of the plant site.

2.2 HYDROLOGY

The main surface water feature in the vicinity of the site is Lake Robinson. The primary purpose for which Lake Robinson was constructed by CP&L is industrial cooling. Secondary uses such as recreation are not restricted by CP&L as long as the primary function is not impaired by those activities.

Downstream from Lake Robinson, on the northern edge of Hartsville, is another smaller impoundment of Black Creek called Prestwood Lake. It serves the Sonoco Products Company, located adjacent to the lake.

2.3 GEOLOGY

2.3.1 Geology of the Region

The site selected for RNP 2 is located in the Coastal Plain physiographic province approximately 15 miles southeast of the Piedmont province.

In South Carolina, the Coastal Plain is composed of largely unconsolidated sediments which overlie a slightly sloping surface of crystalline rock. This crystalline basement is exposed further to the west in the Piedmont province. The boundary between the Piedmont and the Coastal Plain is known as the Fall Line. The Fall Line can be traced from as far north as New Jersey to as far south as Alabama. It is oriented roughly northeast-southwest. The basement crystallines dip to the southeast from 10 to 40 ft per mile. These dipping crystalline rocks in the Piedmont and below the Coastal Plain are composed largely of granite, gneiss, phyllite, and schist. These crystallines are of pre-Cambrian and early Paleozoic age with subordinate sandstones and intrusive diorites of Triassic age.

Triassic sediments have been faulted into the ancient crystallines. In some instances, these sediments were later intruded by dikes, also during the Triassic period. Faulted Triassic basins are evident in the Piedmont province and deep wells have located Triassic rocks in widely divergent areas beneath the Coastal Plain. These faulted Triassic basins appear to be oriented roughly parallel with the trend of the Blue Ridge Mountains to the West, a line generally paralleling the Fall Line.

Overlying the pre-Cambrian, Paleozoic, and Triassic rocks, are the sediments of the Coastal Plain. These sediments are composed of sands, gravels, clays, shales, and limestones which range in age from Cretaceous to Pleistocene. The Coastal Plain itself is divided into the upper Coastal Plain and the lower Coastal Plain by what has been termed the Orangeburg Scarp, an erosional feature representing a shoreline formed during Miocene times. It is well developed in the vicinity of Orangeburg, South Carolina, but only weakly expressed in the northern part of the State near the site.

The Orangeburg Scarp has a gentle slope and is generally more than a mile in width. It is mantled by coalescing alluvial fans which mask the contact of the Tuscaloosa (Middendorf) Formation with the younger Black Creek Formation below the Orangeburg Scarp.

2.3.2 Geology in the Site Vicinity

The Robinson site lies nearly adjacent to the Orangeburg Scarp in the outcrop zone of the oldest late Cretaceous sediments.

The Coastal Plain sediments in the area of the site were formed at the same time as the Tuscaloosa Formation, but locally are known as the Middendorf Formation. The contact of the Middendorf Formation with the younger Black Creek beds to the east is an irregular one. The closest contact to the site occurs roughly eight miles southeast of the site.

In the Fall Line, some 15 miles northwest of the site, the pre-Cambrian basement rocks are exposed at the surface. A considerable thickness of saprolite (in situ weathered granitic rock) covers much of the area of the Piedmont and therefore may extend under the Middendorf in the Coastal Plain area near the site. The saprolite is a tough reddish clayey material, quite impermeable.

Down-faulted Triassic basins occur in the Piedmont adjacent to the Fall Zone and also below the upper and lower Coastal Plain sediments. Deep wells have encountered Triassic rock near Aiken, Sumter, and Florence, and what is probably Triassic rock at Summerville and Dillon. A magnetometer survey inferred a northeast-trending basin in the vicinity of Florence and Dillon.

The Coastal Plain sediments in the area include only late Cretaceous formations. The Middendorf Formation is largely of fluvial origin while the Black Creek Formation is of marine origin. To the southwest the Cretaceous sediments underlie tertiary sands, shales, marls, and limestones. However, only the Middendorf and Black Creek Formations are found in the vicinity of the site.

The Middendorf Formation consists of light-colored feldspathic and slightly micaceous quartz sand interbedded with red, purple, gray, and brown silty and sandy clay. Some of the sand layers have been cemented, resulting in poorly indurated sandstones and occasional laminated mudstones. The formation is irregularly bedded and cross-bedding is common. Stringers of small clay balls are encountered in some strata. However, most of the sands are relatively free of silt and clay. The formation is characteristic of intensive alluviation.

The dip of the Middendorf is southeasterly at about 15 to 20 ft per mile in the Upper Coastal Plain, increasing to about 20 ft per mile beneath the Lower Coastal Plain.

Overlying the Middendorf is the Black Creek Formation, consisting primarily of phosphatic and glauconitic sands, interbedded with hard gray and black clay locally indurated to shale. Within the formation there is carbonized wood, pyrite, and marcasite. Generally, the sands are light gray to yellow in color and contain flakes of mica and some glauconite. The shales possess a flaky fissility. The formation is characteristic of marine deposition in a quiet lagunal environment.

2.3.3 Geology at the Site

The surficial materials at the Robinson site are recent sands or soils developed from the Middendorf.

Because of the high quartz content of the sands and the climatic environment, the surficial soils may not weather sufficiently to differ considerably from the parent material. Thus, it is nearly impossible to distinguish the recent alluvial soils from the parent Middendorf sand since both the alluvial and weathered soils are derived from the Middendorf. Only their manner of placement would be different. From an engineering standpoint, the difference is minor. The subsurface materials encountered in the test holes drilled at the site are completely consistent with recent alluvium and Middendorf Formations encountered throughout the vicinity. Discontinuities within the strata are sedimentary and no structural deformation is apparent in the Middendorf Formation in the site area.

The Middendorf is about 400 ft thick and overlies an eroded, slightly sloping surface of Piedmont crystallines that may be somewhat weathered near the surface.

Triassic basins are known in the area, however, it is believed that the likelihood of a Triassic basin at the site is quite small. The basement rock at the site is considered to be Piedmont crystalline since the results of the seismic surveys indicate a high velocity material at a depth consistent with the depth of Piedmont crystallines encountered in wells in the area.

In general, the upper alluvial sands and gravels are moderately compact. Layers of compressible material occur in the upper 30 to 50 ft. Because of the quantity of fines in the sand and gravel, it could not be considered free-draining material. The underlying Middendorf contains generally compact relatively incompressible sands and firm to hard clayey soils. Several strata of cemented sandstone were encountered in the borings at depths of roughly 90 to 100 ft.

From a geological standpoint, the Middendorf is considered to be an unconsolidated formation. From an engineering point of view, however, the materials are firm and compact and would provide good foundation support for the plant. The materials range in texture from a hard or compact soil to a soft rock.

2.4 SEISMICITY

2.4.1 Regional Seismicity

The largest earthquake in this region occurred at Charleston in August, 1886. Charleston is approximately 120 miles south of the site. This shock had an intensity of about Modified Mercalli IX at the epicenter and it is estimated that this shock had a Magnitude of 6 1/2 to 7 with epicentral acceleration of 0.25g to 0.30g. However, damage was confined to a relatively small area and no permanent scars remain to give testimony to the shock. Aftershocks of the main earthquake had intensities ranging up to Modified Mercalli VII.

Another shock (Modified Mercalli VII) occurred in the Charleston area in 1912. Succeeding shocks from 1914 to the present appear to have decreased in intensity and in the affected area. The last shock in 1960 (Modified Mercalli V) was felt over only 3500 square miles.

An earthquake of Intensity Modified Mercalli VII-VIII occurred in Union County, South Carolina, on New Year's Day in 1913. This is the second largest shock in the Carolinas, and its epicenter lies about 90 miles from the site.

In 1959, an earthquake of Intensity Modified Mercalli V-VI occurred about 15 miles from the site in the vicinity of McBee. No permanent effects of this shock are noted in the literature or in a geologic reconnaissance, although it is presumed to have been felt at the site. It is estimated that this shock had a Magnitude no greater than 4.5 with an epicentral acceleration of well under .10g.

Except for the aforementioned trend of epicenters paralleling the Blue Ridge, there is no apparent trend of other epicenters in the region. Most of the smaller historical shocks were reported in scattered population centers. The seismicity of the region is generally moderate. Of those shocks that do occur, only two earthquakes with epicenters outside of the Charleston areas have had intensities exceeding VI.

2.4.2 Local Seismicity

Only one earthquake of intensity V or greater has ever been recorded within 50 miles of the site. This shock occurred on October 26, 1959, near McBee, Chesterfield County, South Carolina, with an Intensity of Modified Mercalli VI. The epicenter was located about 15 miles from the site. The estimated intensity at the site was about V.

The epicenters of two other shocks are located within 100 miles of the site. The epicenter of the 1913 earthquake in Union County (Modified Mercalli VII-VIII) was approximately 90 miles from the site and the epicenter of the 1945 Lake Murray shock (Modified Mercalli VI) was approximately 70 miles distant. Damage was slight in both epicentral areas and nonexistent at the site.

The Charleston earthquakes occurred about 120 miles from the site. Damage was confined to the epicentral area, and it is unlikely that intensity at the site exceeded VI for the largest Charleston shock.

While the aforementioned shocks were probably felt in the locality of the site, no damaging effects were experienced. The amplitude of ground motion at the site would not cause damage to any reasonably well-built structure.

The sediments underlying the site are quite thick and apparently undisturbed. The surface of the buried crystallines is an ancient eroded one, and active faults are unknown in the vicinity of the site.

2.4.3 Geologic Structure and Tectonic Activity

From a seismic point of view, the most important geological considerations are:

a) The type, structure, and physical properties of the foundation, soils, and rock

b) The existence and location of both active and inactive faults

A detailed description of regional, area, and local geology is presented in the UFSAR. In summary, more than 400 ft of unconsolidated Coastal Plain sediments composed largely of sands with some clay and indurated layers overlie the crystalline basement. The surface of the basement rock slopes from the outcrop zone approximately 15 miles northwest of the site toward the Atlantic Coast. The basement surface is estimated at more than 3000 ft below sea level to the southwest at Charleston, South Carolina.

To evaluate the elastic properties of the foundation materials at the site, field measurements of the velocity of compressional wave propagation were made. The recorded velocities indicate unconsolidated Coastal Plain sediments overlying crystalline basement at a calculated depth of some 460 ft below the existing plant. The measured velocities do not indicate the presence of Triassic rocks.

A study of the possibility of the existence of faults was made during the geologic study of the area. No active faulting was apparent.

No faulting is apparent in the unconsolidated sediments of the Coastal Plain. The underlying basement rocks are effectively masked by more than 400 ft of sediments at the site and cannot be directly observed below the Fall Zone. However, faulting in the basement complex is known from exposures above the Fall Zone and cores from scattered borings drilled through the Coastal Plain sediments.

Faulting of the Triassic Period is evident along the edge of the Deep River Basin, which extends from the vicinity of Durham, North Carolina, into South Carolina near Chesterfield. The precise location of the fault border near Chesterfield is unknown because of the cover of Coastal Plain sediments.

Other Triassic basins are known to exist below the Coastal Plain. Deep borings at the Savannah River Nuclear Facility near Barnwell, South Carolina, and in Florence, South Carolina, have penetrated Triassic rocks.

Suspected Triassic rocks have been encountered below Summerville and Sumter. A magnetometer survey inferred a basin below Florence and Dillon, paralleling the trend of the Deep River Basin. Triassic basins in this area are downfaulted grabens, and, therefore, bounded by faults.

Another major fault in the region is the Blue Ridge Scarp. This scarp forms the southeastern boundary of the Appalachian province. However, it is more than 120 miles to the northwest and not likely to significantly affect the site.

A definite alignment of earthquake epicenters can be seen parallel to the Blue Ridge Scarp in the mountains of western North Carolina.

2.4.4 Safe Shutdown Earthquake

Comparison between the Robinson site and certain areas in California indicate a similarity in the depth and type of overburden material. For this reason Dr. G. W. Housner of the California Institute of Technology recommended the use of his average California response spectra to define the earthquakes. These spectra are shown on Figures 2-2 and 2-3 for various degrees of damping. However, recommendations were; design for maximum horizontal ground acceleration of 0.1g with a vertical component of 2/3 of the horizontal acceleration and hypothetical earthquake maximum horizontal ground acceleration of 0.20g.

To provide an adequate margin of safety, a maximum earthquake ground acceleration of 0.2g was selected for the hypothetical earthquake. It is important to note that even if an earthquake comparable to the Charleston shock were to occur 35 miles from the site, the ground acceleration would not exceed 0.2g.

2.5 GROUND RESPONSE SPECTRA

Two earthquake motions were considered in the dynamic analysis of all Seismic Category I structures, systems, subsystems, and equipment. These two earthquake motions are the operating basis earthquake (OBE) and the Safe Shutdown Earthquake (SSE) or Design Basis Earthquake (DBE). The design value of the maximum horizontal ground acceleration is 0.2g for the safe shutdown earthquake and 0.1g for the operating basis earthquake. A vertical component of two-thirds of the magnitude of the horizontal component was applied simultaneously.

The OBE peak acceleration is based on a Richter scale Magnitude 4.5 earthquake with an epicentral distance of less than ten miles from the site. The probable ground acceleration from this earthquake would be .07 to .09g. However, .10g was selected conservatively. The SSE peak acceleration of .20g is based on a magnitude 7.0 earthquake comparable to the 1886 Charleston shock occurring 35 miles from the site. The accelerations from that hypothetical earthquake would not exceed .20g. These ground motion horizontal and vertical spectra shapes and peak acceleration are Housner's average California response spectra. He based his selection on the similarities between the depth and type of overburden material at the Robinson site and certain areas in California.

2.5.1 Damping Values

Damping values for Class I components and structures were given from Westinghouse to Ebasco in 1967 as a percentage of critical. These damping values are as follows:

Containment Structure	
Design Earthquake -	2.0
Hypothetical Earthquake -	5.0
Concrete Support structure of Reactor Vessel -	2.0
Welded Structural Steel Assemblies -	1.0
Bolted or riveted Steel Assemblies -	2.5
Vital Piping Assemblies -	0.5
Concrete Structures above Ground -	5.0

2.5.2 Floor Accelerations

As a result of the analyses of the building structures by Ebasco in 1970, the maximum absolute OBE of 2% damping and SSE of 5% damping accelerations at various elevations within each building were derived. These maximum floor accelerations were derived by standard response spectra techniques and were based on the design Housner ground response spectra, structural frequencies, mode shapes, and participation factors.

In early 1970, Westinghouse generated SSE 1/2% damped horizontal floor response spectra. Similar floor response spectra were never generated for OBE conditions. The DBE horizontal floor response spectra were generated for specific elevations by exciting the multi-degree of freedom dynamic model of the building with a normalized acceleration time history forcing function, which gives a ground response spectra as large as the design Housner response spectrum acceleration for 2% and 5% damping. The time history accelerations at each mass point of the building models were then used to construct a 1/2% damped floor response spectrum for a one degree of freedom system. This represents the maximum response of the equipment located at the mass point which represents the building elevation under consideration. Westinghouse tabulated the peak values of the DBE 1/2% damped horizontal response spectrum curves.

The vertical response spectra curves were taken as 2/3 the 1/2% damped horizontal ground spectra.

The unbroadened Westinghouse spectra curves were later digitized by Ebasco. Ebasco also provided additional enveloped response spectra curves for use in the analysis of piping systems attached to various elevations or different areas of buildings. These curves were also based on the original Westinghouse unbroadened curves. The digitized response spectra were used in the IE Bulletin 79-14 (Reference 33) piping and pipe support reanalysis effort and were transmitted to CP&L in a February 26, 1985 letter.

2.5.3 Modeling Techniques for the Robinson Response Spectra

The Category I structures were reviewed based on four foundation models. The models that were considered are as follows:

- Fixed Base
- Rotational Spring with Stiffness Computed from Pile Test Data
- Rotational Spring and Translational Spring Computed from Lateral Pile Test Data
- Rotational Spring and Translational Spring Computed from the Properties of the Soil Mass Assuming No Contribution From the Pile Group

The second case involving the rotational spring with stiffness computed from pile test data produced the most conservative design.

All Category I structures were modeled as cantilever stick models attached to a ground spring with mass nodes located at various locations on the cantilever.

The containment structure was modeled with a mass at grade (Elev. 226), at 63 feet above grade (Elev. 289), 126 feet above grade (Elev. 352), and at 175 Feet above grade (Elev. 401).

The Containment Inner Structure was modeled with a mass at 6.5 feet above grade (Elev. 232.5), at 22 feet above grade (Elev. 248), 46.5 feet above grade (Elev. 272.5), and 80.5 feet above grade (Elev. 306.5).

The Reactor Auxiliary Building was modeled with a mass at the grade elevation (Elev. 226), at 20 feet above grade (Elev. 246), and at 36 feet above grade (Elev. 262).

The Spent Fuel Pit was modeled with a mass at the grade elevation (Elev. 226), a mass at 12.5 feet above grade (Elev. 238.5) , a mass at 25 feet above grade (Elev. 251), a mass at 37.5 feet above grade (Elev. 263.5), a mass at 50 feet above grade (Elev. 276), a mass at 77 feet above grade (Elev. 303), and a mass at 98 feet above grade (Elev. 324).

The Class I Turbine Generator Building was modeled with a mass at grade (Elev. 226), a mass at 18.5 feet above grade (Elev. 244.5), a mass at 39 feet above grade (Elev. 265), and a mass at 56.5 feet above grade (Elev. 282.5).

Lastly, the intake structure was modeled with a mass at Elevation 186 (40 feet below ground level), a mass at Elevation 206 (20 feet below ground level), and a mass at ground elevation 226.

2.6 STRUCTURES

2.6.1 Containment Building

The reactor containment structure is a steel lined concrete shell in the form of a vertical right cylinder with a hemispherical dome and a flat base supported by means of piles. The containment structure is designed for an accident pressure based upon the pressure transients as shown in Section 15.6 of the UFSAR. The containment structure is designed to contain radioactive material which might be released from the core following a loss-of-coolant accident.

The structure consists of side walls measuring 126 ft from the liner on the base to the springline of the dome and an inside diameter of 130 ft. The containment free volume is 1,950,000 ft^3. The side walls of the cylinder and dome are 3 ft 6 in. and 2 ft 6 in. thick, respectively. The inside radius of the dome is equal to the inside radius of the cylinder, i.e., the discontinuity at the springline due to the change in thickness is on the outer surface. The base consists of a 10 ft thick structural concrete slab. The base liner is installed on top of the structural slab and covered with two feet of concrete.

The basic structural elements considered in the design of the containment structure are the piles, base slab, side walls, and dome acting as essentially one structure under all loading conditions. The bottom plates of the liner are laid loose on the foundation slab and are anchored only at the hangways for the crane wall and primary shield. In the vertical walls and dome, the liner is anchored to the concrete shell by means of "KSM" shaped anchor studs fusion welded to the liner plate so that it forms an integral part of the entire composite structure under all loadings. The cylindrical portion of the liner is insulated. The dome of the containment is reinforced concrete. The cylinder walls are concrete-reinforced circumferentially and prestressed vertically. The base slab is reinforced concrete.

The base slab is 144 ft diameter circular reinforced concrete slab 10 ft in thickness. At the center it is penetrated by the reactor sump which extends 16 ft below the slab and is designed to hang from the slab. The base slab is designed to be supported by 923 steel pipe piles which supply restraint to it both vertically and horizontally and are anchored to it where required to provide restraint for uplift. The base slab is reinforced with a radial, circumferential pattern on the top surface and a rectangular grid of reinforcing steel on the bottom which fits between the piles.

The containment is supported on pile foundations. The large depth of relatively low bearing strength soil occurring at the surface was a major factor in the selection of piles to support the containment. Piling safely carries the structural loads through the surface soils and transmits them to the dense soils underlying the area.

The containment liner is designed to serve as a leakproof membrane and is not relied upon for the structural integrity of the containment except for resisting tangential shears in the dome. It is anchored to the concrete by means of "KSM" shaped steel studs. The liner is not anchored to the concrete base slab hence does not act compositely with it. It was laid loose on the base slab and the butt weld backing strips were set in grooves in the base slab. After welding, the distortions in the liner were considered too great and a neat cement grout was flowed beneath it to fill the voids. A bond breaker, form oil, was flowed first on the base slab to prevent the liner from acting compositely with the slab.

2.6.2 Internal Structures

The reinforced concrete containment structure encloses the concrete structures and structural components which comprise the containment internal structures. The containment internal structures provide support for the NSSS equipment during all operational phases and, in the unlikely event of an accident, act to mitigate the consequences of the accident by protecting safety-related equipment and other engineered safety features from the effects induced by the accident.

The concrete internal structures, which consist of the primary and secondary shield walls and other concrete enclosures, form compartments within which the entire RCS is located. The main components are the concrete primary shield wall, which encloses the reactor cavity, the semi-circular concrete secondary shield walls, which forms the steam generator and reactor coolant pump compartments, the reinforced concrete walls and floors, the fuel storage area, refueling pool and reactor internals laydown area, the concrete enclosure walls around the pressurizer, the containment steel floors, stairs and platforms, reactor vessel supports, steam generator supports, and reactor coolant pump supports. These structures are reviewed in more depth below.

The primary and secondary shield walls are constructed of concrete and surround the Reactor Coolant System. These walls provide shielding to permit access into the containment during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the containment liner plates.

The steam generators are supported on a structural system consisting of four connected columns all welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction to accommodate piping expansion with a system of "Lubrite" plates, hydraulic snubbers, guides, and stops. The "Lubrite" plates, hydraulic snubbers, guides, and stops are designed as damped support to resist the action of the seismic and pipe break loads. Sliding shoes at the top of the support structures permits radial thermal growth of the steam generators during heatup.

Steam generator lateral bracing is provided near the upper tube sheet elevation to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing is provided at a lower elevation to resist pipe rupture loads.

The reactor vessel support structure consists of a circular box section ring girder fabricated of carbon steel plates. The bottom flange of the girder is in continuous contact (except for opening for neutron detectors) with a non-yielding concrete foundation.

The reactor vessel has three supports located at alternate nozzles. Each support bears on a support shoe which is fastened to the support structure. The support shoe is a structural member that transmits the support loads to its supporting structure. Each support shoe is designed to restrain vertical, lateral, and rotational movement of the reactor vessel but allows for thermal growth by permitting radial sliding on bearing plates.

Each reactor coolant pump is supported on a three-legged structural system consisting of three connected columns fabricated of carbon steel members, structural sections, and pipe. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts which restrain the structure from movement beyond the calculated limits. Sliding shoes at the top of the support structures permit radial thermal growth of the pumps during heatup.

The pressurizer is supported on a heavy concrete slab spread between the concrete shield walls of its compartment. The pressurizer is a bottom skirt support vessel resting on a type of ring girder.

The concrete and steel internal structures are supported on a concrete foundation mat feet thick resting on 923 piles that extend down to bedrock.

The structural acceptance criteria for the Containment Internal Concrete Structures and the internal and other Seismic Category structural steel structures consists of compliance with the following requirements:

a) Concrete Structures - To assure that the structural integrity of Category I concrete structures is maintained for the service and factored load conditions, the limits of the stress and strain intensity of concrete generally follow the strength design method requirements of the American Concrete Institute (ACI) 318-63.

Using the factored loads, the various components have the required load capacity if the stresses in them do not exceed the yield strengths of the material used. To provide for the possibility that small, adverse variations in dimensions and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in a net under capacity of the component, the load capacities of the individual structural members are reduced by a reduction factor "ϕ" for the design cases.

The factors were established for the design on the basis of the function of the component and the effect on its net capacity of the variations enumerated above. These factors are generally in accordance with the ACI 318-63 Code and are tabulated in the FSAR.

b) Steel Internal Structures - Structural steel framing is designed for the loading combinations given in the FSAR to exhibit either elastic or plastic behavior in all load carrying elements. To assure that the structural integrity of the Seismic Category I steel structures is maintained, limits on the resulting stresses and the reduced strength capacities are observed.

2.6.3 Reactor Auxiliary Building

The Reactor Auxiliary Building (RAB), including the control room and the diesel generator room, is a Class I structure and has been designed in accordance with the procedures described for the containment structure.

Since the original construction, a waste evaporator enclosure has been installed on the roof of the building. The existing RAB, including the pile foundation, was analyzed to verify its structural adequacy to withstand all the original design loads plus the additional loads imposed by the waste evaporator enclosure and its associated equipment.

2.6.4 Spent Fuel Pit

The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor.. The spent fuel storage pit is constructed of reinforced concrete having 3 to 6 ft thick walls and is Class I seismic design. The entire interior basin face and transfer canal is lined with stainless steel plate.

The Fuel Handling Building is a Class III metal framed structure over the spent fuel pit. It was designed for seismic loads in accordance with the Uniform Building Code. The building was designed to carry a 150 ton crane and all its associated loads. The actual crane installed had a rated hook load of 100 tons which was later increased to 125 tons.

The Fuel Handling Building crane will not be stored in a position over the spent fuel pit. Hold down lugs have also been provided on both trucks and trolley of the refueling crane to prevent its wheels from lifting from its rails when subject to a vertical earthquake force of 0.133 times the unladen weight of the crane for the lugs on the truck or 0.133 times the deadweight of the trolley for the lugs on the trolley.

2.6.5 Seismic Class I Section of the Turbine Building

The Class I portion of the Turbine Building is north of the Class III portion and is a separate structure. All framework and supports for Class I equipment have been designed to Class I seismic design criteria. The sum of primary stresses resulting from operating conditions and the stresses resulting from the design earthquake was limited to 133 percent of allowable stresses, as permitted by the Uniform Building Code.

All safeguards equipment in this Class I area is located on the ground floor (Elevation 226 ft). There is a Class I concrete ceiling over the top of this area to protect it from above. In addition, the Class I trench in this area is below grade with a checker plate which covers and protects the contents of the trench from falling debris in the event of an earthquake.

The Class III structural steel portion of the Turbine Building and the structural steel parking facility for the Turbine Generator Crane have been dynamically analyzed using the same hypothetical earthquake accelerations as for the Class I portion of the turbine building. The dynamic analysis included considerations for the dead weight of the crane. Ebasco Computation Book 2 for drawings G190531-G190547 provides the detailed analysis for the columns, beams, and bracing members. The results of this analysis show that the stresses in the structure will be below yield. The parking facility steel includes holdown lugs, storage locks, and rail stops for the crane. The total maximum displacement for both portions of the Turbine Building do not exceed the spacing between both portions of the building.

The Turbine Building crane structure was designed to be capable of raising, lowering, holding in any position, and transporting an occasional load of 125% of the rated load without damage or distortion to any crane part. The crane was also proportioned and designed so that the stability moment of its dead load was equal to at least 150% of the overturning moment due to the wind load specified when the crane is not in operation. The crane will also be stored in the unloading bay which is west and south of the Class I bays. As was noted above, hold-down lugs were also provided as part of the parking facility to resist vertical uplift due to earthquake.

2.6.6 Intake Structure

The intake structure is designed as Seismic Class I and has been analyzed for normal operating and seismic conditions. It is therefore not subject to collapse under earthquake loading. In addition, hydrodynamic loading due to the contained and surrounding water has been computed under seismic conditions in accordance with the procedures given in "Nuclear Reactors and Earthquakes - TID 7024" Chapter 6. The intake structure was designed for the hydrodynamic, normal operating, and seismic loads in accordance with the stress limits defined in ACI 318-63 Part IVB.

The four service water pumps are located in three separate bays in the intake structure, the middle bay containing two pumps. The walls separating the bays and the deck above the piping are two and one half feet thick reinforced concrete.

2.7 EQUIPMENT SUPPLIED BY THE NSSS VENDOR

Robinson Steam Electric Plant Unit No. 2 incorporates a closed-cycle pressurized water Nuclear Steam Supply System (NSSS) and a Turbine-Generator System utilizing dry and saturated steam. The Nuclear Steam Supply System consists of a pressurized water reactor, Reactor Coolant System (RCS), and associated auxiliary fluid systems and is supplied by Westinghouse. The Reactor Coolant System is arranged as three closed reactor coolant loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to one of the loops.

A seismic analysis was performed on each reactor coolant loop which consists of the reactor vessel, steam generator, reactor coolant pump, the pipe connecting these components, and the large component supports. The components and piping were modeled as a system of lumped masses connected by springs whose values were computed from elastic properties input. A simplified support model was arrived at by representing the structural support system as equivalent springs rather than as member beams and columns.

The analysis was performed using a proprietary computer code called WESTDYN. As input, the code uses system geometry, inertia values, member sectional properties, elastic characteristics, support and restraint characteristics, and the appropriate CP&L seismic floor response spectrum for 0.5 percent critical damping. Both horizontal and vertical components of the seismic response spectrum were applied simultaneously along the Y and Z axis. Previous analysis indicated the Z direction to be the most critical horizontal direction for maximum pipe stress.

The modal participation factor was combined with the mode shapes and the appropriate seismic response spectrum values to give the structural response for each mode. Then the forces, moments, deflections, rotations, constraint reactions, and stresses are calculated for each significant mode. The modal stresses were then summed by the square root of the sum of the square method for each significant point in the system to determine the total stress.

The restraints, supports, and other constraints assumed for input into the seismic computer model are given below:

a) Reactor Vessel - The vessel is rigid

b) Steam Generator - The steam generator at the upper support point is permitted to translate along and rotate about the X,Y, and Z axis, but translations along X and Z are resisted by spring representing the upper support. The steam generator at the lower support point is permitted to translate along and rotate about the X,Y, and Z axis, but all movements are resisted by springs representing the lower supports' stiffness; and,

c) Reactor Coolant Pump - The reactor coolant pump is permitted to translate and rotate about the x, Y, and Z axis, but all movements are resisted by springs representing the supports' stiffness.

2.8 EQUIPMENT SUPPLIED BY OTHER THAN NSSS VENDOR

The purpose of this section is to identify the procedure used in the seismic design of Class I equipment supplied by other than the NSSS vendor.

2.8.1 Seismic Criteria

Seismic requirements and design adequacy were determined as follows:

a) The horizontal seismic accelerations used were equal to or greater than the accelerations that occur at the equipment location (i.e., at the proper building elevation) as determined from the building dynamic analysis for the DBE

b) The vertical seismic accelerations used were 2/3 of the value selected for horizontal acceleration

c) The relative stiffness of the equipment and its support was evaluated based on past experience with similar types of equipment. If the equipment was relatively rigid with a fundamental frequency above 15-20 cps then seismic design was performed by using a seismic g loading applied at the center of gravity of the equipment. This g loading corresponded to the combined mode g loading at the elevation of the building on which the equipment was supported. If the equipment was flexible, then a rationed g is applied at the center of gravity of the equipment. This g loading included potential response of the equipment to building motion and the effects of both building and equipment damping.

2.8.2 Seismic Evaluation

Class I equipment (flexible and rigid) have been evaluated to assure functional adequacy when considering potential equipment resonance with the building during earthquake conditions.

Electrical racks, panels, controls boards, etc. fall in the category of flexible equipment. This equipment is located in the Auxiliary Building at or below Elevation 258 ft. The peak acceleration that a one-degree-of-freedom system in resonance with the building would experience is at Elevation 258 ft. This peak acceleration is 2.0g for 0.2g ground acceleration with 0.5 percent of critical damping. For the minimum damping anticipated in this type of equipment, i.e., 1 percent of critical value, the peak acceleration is reduced to about 1.6g. This is the maximum equivalent static load that should be used to account for both floor acceleration and response spectra distortion.

When the equipment supports were designed, the equivalent static load was selected by accounting only for the floor acceleration. This means that a load of

(0.30g/0.20g) x 0.69g = 1.05g was selected for equipment at or below Elevation 258 ft. The design stresses were 2/3 of the material yield, e.g., 24,000 psi. Hence, the correct load of 1.6g would cause a maximum stress of 24,000 x (1.6g/1.05g) = 36,500 psi. The ultimate stress of this type of material is of the order of 70,000 psi. This gives a margin of 33,500 psi between the ultimate capacity and the maximum expected stresses.

For equipment considered as relatively rigid (i.e. having a fundamental frequency above 15-20 cps) seismic design was performed using a seismic g loading applied at the equipment center of gravity.

2.9 CLASS I PIPING OTHER THAN REACTOR COOLANT

Class I piping other than reactor coolant system piping was originally qualified by either static or dynamic analyses.

In response to NRC IE Bulletin No. 79-14 (Reference 33) all safety related piping 2.5 inches in diameter or greater was dynamically analyzed as well as all Seismic Category I piping regardless of size that had been previously dynamically analyzed by computer methodology.

During the resolution of IE Bulletin No. 79-14 (Reference 33), an OBE static and a DBE dynamic analysis was performed on the safety-related piping systems. The OBE static analysis followed the procedure outlined in Section 3.7.3.2.7 of the FSAR. The DBE dynamic analysis incorporated the following techniques:

a) The DBE response spectrum curves (0.5% damping) were broadened plus and minus ten percent.

b) The inclusion of closely spaced modes followed the guidelines of NRC Regulatory Guide 1.92 (Reference 34).

c) The cutoff frequency was 33 Hertz.

d) The participation of mass in the rigid range was included.

e) A 3D earthquake was formed using the SRSS method.

f) The vertical response spectrum was taken as 2/3 of the building ground response curve.

g) In some cases multi-level excitation (different response spectra for different restraints) was used.

The use of dynamic seismic analysis instead of the original static seismic analysis more closely models the actual seismic response of the pipe so the loads are more representative. The dynamic analysis methods used for the IE Bulletin Program represent techniques accepted by the NRC and generally used in the nuclear industry.

2.10 BURIED PIPING

Buried piping for Class I systems at Robinson is limited to the following three systems:

1. Service Water System - Two supply lines routed from the Intake Structure to the Reactor Auxiliary Building;

2. Primary Water System - The supply line from the Primary Water Storage Tank to the Primary Water Pumps; and,

3. Diesel Fuel Oil System - Two supply lines from the Fuel Oil Storage Tank to the Day Tank.

The Service Water System piping is composed of thirty (30) inch diameter cement lined AWWA carbon steel pipe with bell-and-spigot joints.

The Primary Water System piping is composed of four (4) inch diameter stainless steel welded pipe.

The Diesel Fuel Oil System is composed of two (2) inch diameter carbon steel welded pipe.

On April 14, 1992, CP&L made a presentation to the NRC concerning various structural parameters for the Robinson plant. This audit included a brief section on buried piping. The audit identified that a portion of the service water supply piping has been rerouted around the Radwaste Building as part of a major modification process. The report also summarized the current piping inspection program at Robinson. A portion of the Service Water buried pipe failed during a 1982 hydrotest. As a result of this failure, both sections of the Service Water buried pipe were inspected during RFO 13 (Fall, 1990). The north supply line was also reinspected during RFO 15 (Fall, 1993) and inspections are planned for future outages. Portions of the diesel fuel oil piping were uncovered and inspected for wall thickness and signs of exterior corrosion during RFO 14 (Spring, 1992). This inspection determined that the wall thickness remained acceptable and no evidence of exterior corrosion was found.

A corrosion concern was identified for the Service Water pipe in 1990. A task force was established to investigate and resolve any corrosion issues. The task force report provided additional information on the seismic acceptability of the service water piping which is applicable to other buried pipe as well.

The primary purpose of the report was to outline the issue of corrosion in the joint area of the buried service water piping and to outline code basis and requirements, and to provide recommendations for resolution.

The buried service water piping is 31.375 inch OD. It has a 0.188 nominal wall thickness with a 3/8 inch thick cement mortar liner. The piping was purchased and installed in 1968 per the following specifications:

1. AWWA(American Water Works Association) -C-202-064 entitled Mill Type Steel Water Pipe;

2. AWWA-C-205-062T entitled Cement-Mortar Protective Linings and Coatings for Steel Water Pipe - 4 Inch and Larger-Shop Applied; and,

3. AWWA-C-203-66 entitled Coal Tar Protective Coatings and Linings for Steel Water Pipelines.

The Service Water pumps and piping were given the classification of Class I. All systems and components designated Class I were designed so that there would be no loss of function in the event of the maximum hypothetical ground acceleration acting in the horizontal and vertical directions simultaneously. The pipe was designed per ASA B31.1-1955 but ASA B31.1-1955 did not address seismic design. Therefore, piping stress analysis for Robinson is governed by the use of USAS B31.1-1967.

The seismic issue was addressed in a response to question IIIE B10C dated September 17, 1969 incorporated in the Volume 4 of the Final Facility Description and Safety Analysis Report (Reference 11) to the Atomic Energy Commission (AEC). The question read as follows:

> In regard to the adequacy of the piles for lateral loading, it is stated on page 5.1.2-19 that the design of the piles per seismic loadings considered the piles as moving with the surrounding earth and that the full base shear on the base slab was transmitted to the piles. On page 5.1.2-20, it is stated that the ground movement during earthquakes is assumed to be 4 inches and that the piles are designed for a 4-inch differential deflection between base slab and pile tip. What is the effect of the ground displacement on buried Class I pipes?

The answer to this question is as follows:

> The buried Class I Pipe is designed to move with the ground and assume a sinusoidal shape corresponding to the ground wave without exceeding allowable stresses or opening of joints. The design criteria for the plant does not include provisions for ground fissures or other gross permanent ground displacements.

This response included the mechanical couplings which are installed to reduce the loads on the intake structure and strainers due to seismic and thermal conditions. The couplings basically act as a tied expansion joint. Evaluations exist which calculate the relative displacement of the pipe under maximum expected ground movement. The displacement was concluded to be negligible. The original licensing basis assumption is that the pipe would flex during an earthquake with no loss of function.

2.11 DISTRIBUTION SYSTEMS

The distribution system at the Robinson Steam Electric Plant Unit No. 2 consists of seismic Category I cable trays, conduits, HVAC ducts, instrument air lines, and other equipment supports. Robinson was an early vintage construction and was not subject to the same rigorous seismic requirements as later plants for operating equipment and components. For this reason, original plant distribution systems were installed based on accepted construction practice for similar systems in other heavy industrial facilities such as fossil power plants, petrochemical plants, manufacturing facilities, and pulp and paper mills. As the experience database proves, installations using standard construction practices are acceptable provided they are properly anchored. Support of distribution system components since approximately 1980 has used the modal response spectra analysis methodology. This methodology took into consideration the effects of multiple spans and multiple modes on seismic response.

Original cable tray and HVAC distribution systems at Robinson were supported by braced cantilever type hangers or by trapeze type rod hung hangers. The conduits and instrument air lines were typically supported by braced cantilever type hangers, rod hung trapeze type hangers, finger clamps directly to the wall, or beam clamps to building steel, pipe hangers, or other adjacent component hangers.

Later distribution systems were designed using the response spectra methodology. This methodology used a three dimensional mathematical model to construct a sufficient number of dynamic degrees of freedom to closely simulate the dynamic behavior of the subsystems. All of the significant modes of the subsystems were selected for the determination of the seismic response. When the supports for a subsystem were all mounted at the same floor, the relative displacements among supports were not considered. The relative displacements were considered where the supports of the same subsystems were located on different floors.

For the case where supports of the same subsystem were located in different buildings, the maximum relative displacements among the different supports were considered in the seismic dynamic analysis of the subsystem.

Cable trays and conduits have received the most scrutiny and have been reviewed on a room-by-room basis by Seismic Review Teams on several occasions. Initially, cable trays were walked down during May 1989. SQUG methodology was utilized as a guide although the criteria was still being prepared and negotiated for final acceptance by the NRC. During the Fall 1993 and Spring 1994, interaction evaluations involving the distribution systems were considered while the Seismic Review Teams reviewed the safe shutdown components. The primary distribution system walkdowns involving cable trays, conduits, HVAC ducts, and instrument air lines occurred during the Fall 1994. However, information obtained from all of the walkdowns were utilized when formulating the conclusions for the seismic adequacy.

The seismic evaluation program demonstrated that the as-installed cable trays, conduits, miscellaneous raceways, HVAC ducts, instrument air lines, etc. were adequate. There were thirty-seven (37) items involving cable tray and conduits that were determined to be outliers. These thirty-seven items have been identified to plant management at Robinson to be fixed or modified based on the extent of the outlier condition. The resolution of these thirty-seven (37) cable tray and conduit items by modifications, work tickets, and/or further analysis will allow the entire distribution system including cable trays, conduits, miscellaneous raceway, HVAC ducts, and instrument air lines to be screened out as acceptable.

2.12 SEISMIC SPATIAL SYSTEM INTERACTIONS

The Robinson Steam Electric Plant Unit No. 2 was licensed in 1971. Regulatory Guide 1.29 (Reference 36) concerning interdiscipline clearances and seismic II/I requirements had not been developed for incorporation into the design basis at that time. The licensing basis for Robinson did not require that these issues be addressed for the original design or installation and the Final Safety Analysis Report has not been revised to include this recent requirement. The following three examples demonstrate how some design issues have been addressed.

Continued operation of the Control Room air conditioning system during both normal and emergency conditions to maintain the Control Room habitable will be assured by the following:

a) Safety-related system components are designed to Seismic Class I requirements. Nonsafety-related system components are seismically supported to Seismic Class I requirements where failure during a seismic event could compromise the operability of safety-related components of the system.

All electrical systems and components vital to plant safety, including the emergency diesel generators (DG) are designed as Class I and designed or arranged so that their integrity is not impaired by the maximum hypothetical earthquake, wind storms, floods, tornado winds, or disturbances on the external electrical system. Power wires, control cabling, instrument cabling, motors, and other electrical equipment required for operation of the engineered safety features (ESF) are suitably protected against the effects of either a nuclear system accident or of severe external environmental phenomena in order to assure a high degree of confidence in the operability of such components in the event that their use is required.

The high-density spent fuel storage racks designated as Seismic Category I will comply with Regulatory Guide 1.29, Revision 3.

The Seismic Review Teams evaluated the safe shutdown equipment, the cable trays and conduits, the piping, the HVAC ducts and registers, the masonry walls, and other items of concern for seismic interaction concerns and flexibility issues during the A-46 and Seismic IPEEE walkdowns. All such interaction and flexibility issues have been identified and addressed on the SEWS forms or other suitable methods of reporting such issues.

Although not required by the licensing requirements, all new engineering design at Robinson takes into consideration interaction and flexibility concerns. The design engineer is responsible for evaluating any interaction or flexibility issue prior to issuance of the engineering document.

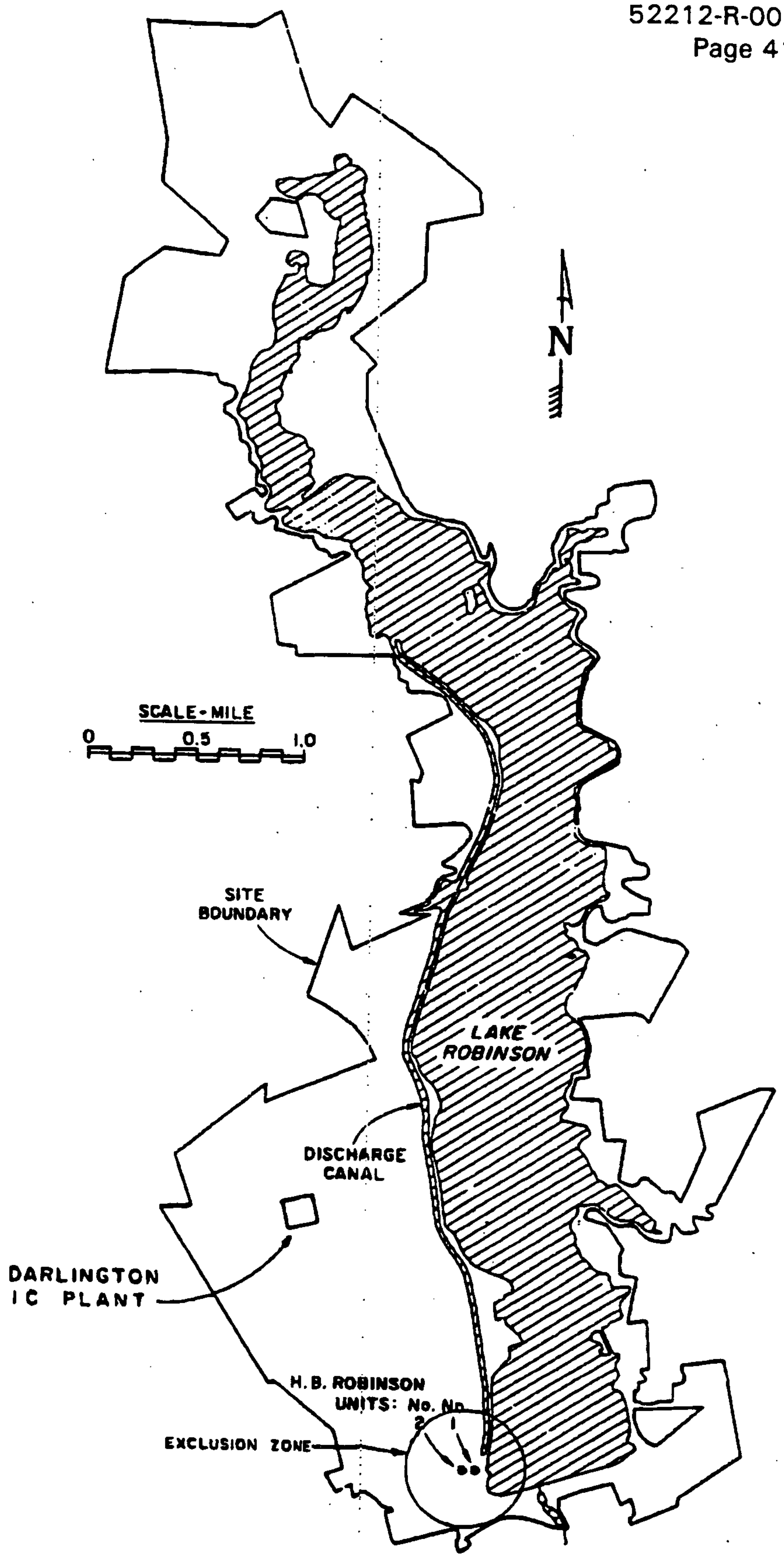


Figure 2-1: Robinson Site Boundary and Exclusion Zone

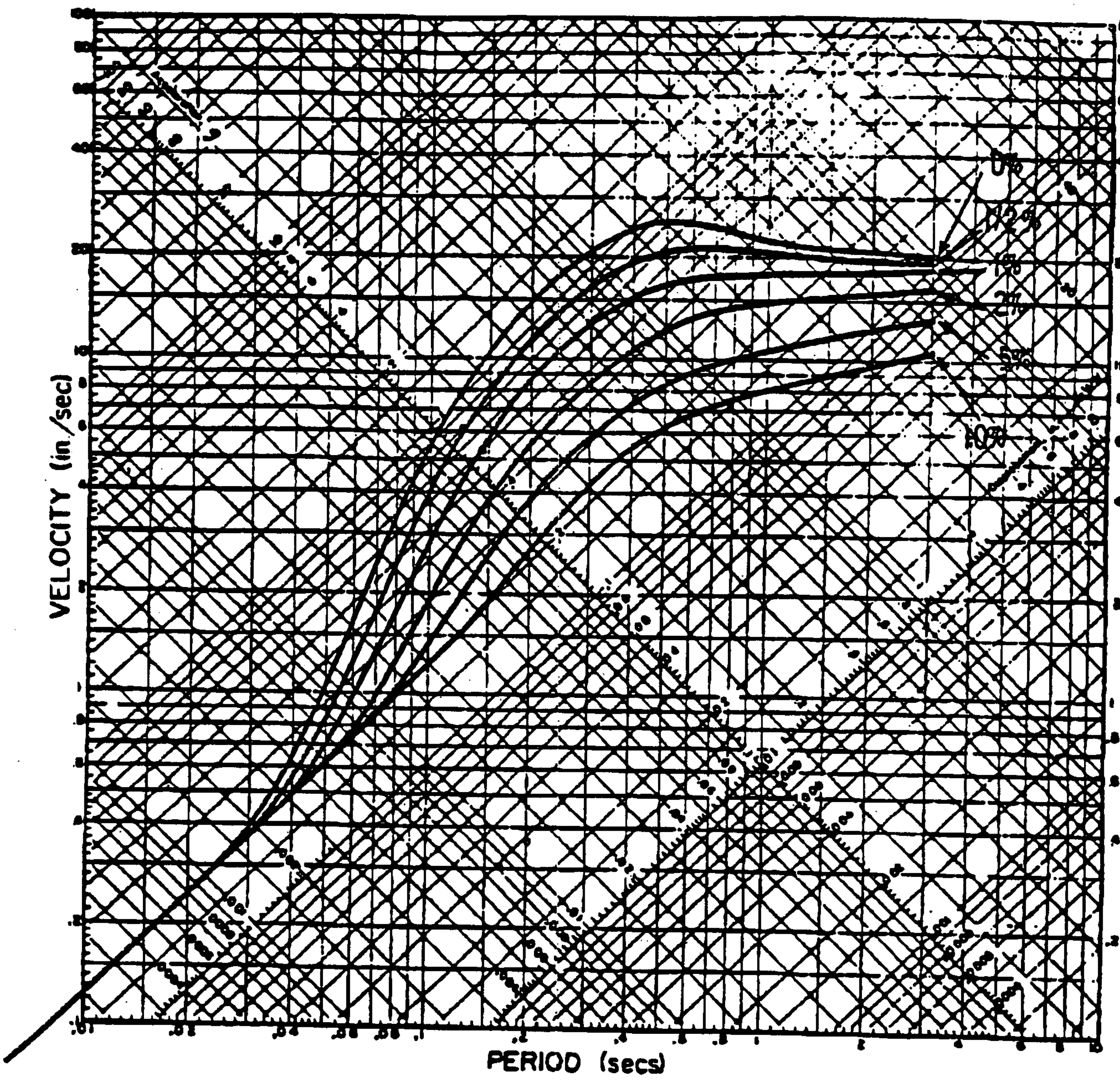


Figure 2-2: Horizontal Design Response Spectrum Safe Shutdown Earthquake

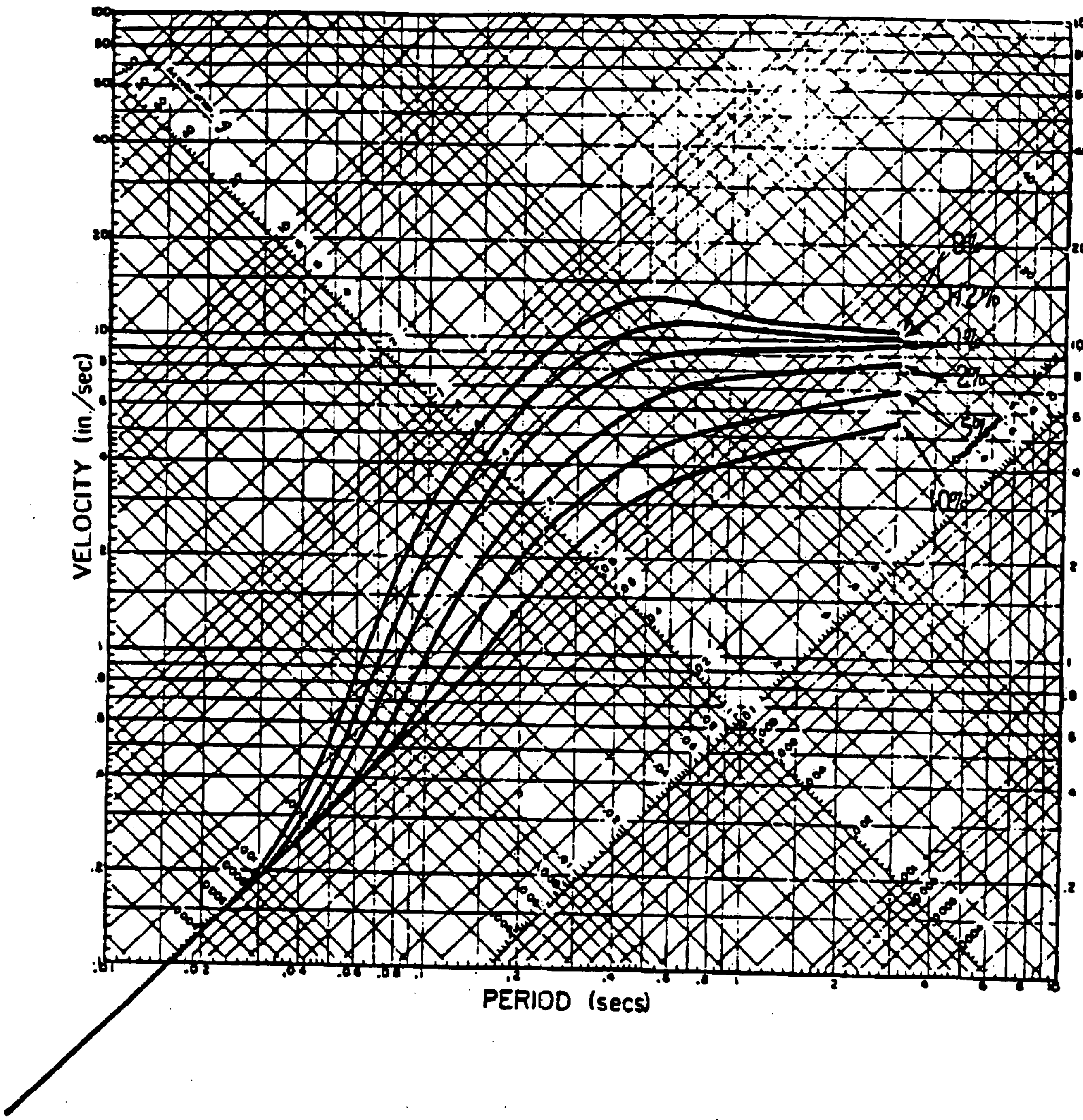


Figure 2-3: Horizontal Design Response Spectrum Operating Basis Earthquake

3. SYSTEM DESCRIPTION AND SUCCESS PATH LOGIC

This task involves the identification of components and structures for in-plant review. EPRI NP-6041 (Reference 3) was utilized in choosing the items and identifying boundary conditions and assumptions.

The functions involved in the plant response to the Review Level Earthquake are as follows:

1. Reactivity Control;
2. Reactor Coolant system Inventory Control;
3. Reactor Coolant System Pressure Control; and,
4. Decay Heat Removal

The following systems and support systems were identified for ensuring critical plant functions:

- Reactor Coolant System
- Safety Injection System
- Residual Heat Removal System
- Feedwater, Condensate, and Evacuation System
- Main and Extraction Steam System
- Instrument and Station Air System
- Nitrogen Supply System
- Service and Cooling Water System
- Component Cooling Water System
- Fuel Oil System
- Emergency Diesel Generator System
- Safety Related DC Power

- HVAC for Emergency Diesel Generator A Room, Emergency Diesel Generator B Room, and the Control Room

- Chemical and Volume Control System

The structures and areas where the systems and components are located include the following:

- Reactor Containment Building
- Reactor Auxiliary Building
- Service Water Intake Structure
- Class 1 Bay of the Turbine Generator Building
- Fuel Building
- Yard
- Residual Heat Removal Room

The resulting success path logic diagrams evolved from studying available plant equipment function as well as the plant's normal and emergency operating procedures. Two or more success paths are required for each of the four major system functions. The Success Path Logic Diagram considers two conditions. One condition is a loss of coolant accident involving a one-(1)inch diameter pipe. The other condition involves a transient without Reactor Coolant System (RCS) leakage.

The Success Path Logic Diagram was reviewed and agreed upon by the Robinson operations personnel. The Seismic IPEEE walkdowns were performed in conjunction with the USI A-46 walkdowns. The equipment selected for safe shutdown was a combination of equipment required to satisfy the USI A-46 criteria and also equipment required to satisfy the Seismic IPEEE criteria. Equipment selected for inclusion on the Safe Shutdown Equipment List (SSEL) was evaluated in a manner similar to that described in the SQUG Generic Implementation Procedure (GIP) (Reference 8). Guidance from EPRI NP-6041 (Reference 3) was also used in preparing the format for the list of components.

Appendix B, entitled "Success Path Logic Diagram for Seismic Margins Analysis," provides more detailed information for this section.

4. SEISMIC MARGINS IN-STRUCTURE RESPONSE SPECTRA

Floor response spectra were generated for major structures of H.B. Robinson Unit 2 for use in Seismic Margins Assessment (SMA). Five Class I structures, most of which were pile supported, were selected for study, *viz.* Reactor Auxiliary Building (RAB), Containment Structure (CS), Inner Structure (IS), Circulating Water Intake Structure (CWIS), and Turbine Building Class I bay (TGB).

The spectra generation employed sub-structure SSI technique. In the SMA method, a Review Level Earthquake (RLE), which constitutes a screening level for seismic capacity, is specified. For Robinson, the RLE was defined in Reference 2 as the NUREG/CR-0098 median rock or soil spectrum anchored to 0.3g peak ground acceleration (PGA). Variability in soil properties was considered by varying the soil shear modulus from 0.5 times the best estimate strain-compatible values up to 90% of the best estimate low strain values (Reference 3). Three analyses were then performed with different soil properties, followed by enveloping of in-structure response spectra. The key elements of the sub-structure approach to SSI are as follows:

- Specification of free-field ground motion
- Determination of soil profile
- Development of foundation impedances
- Calculation of dynamic characteristics of structures
- Analysis of the coupled soil-structure system

4.1 SPECIFICATION OF FREE-FIELD MOTION

Specification of free-field motion entails specifying the control point, frequency characteristics of the control motion, and spatial variation of ground motion. The RLE for Robinson was stipulated to be the NUREG/CR-0098 median spectral shape for soil anchored to PGA of 0.3g. Vertical ground spectrum was taken as 2/3 of the horizontal component. For surface motion defined by the CR-0098 spectral shape, control point at finished grade in the free-field was appropriate. Spatial variation of ground motion was defined by vertically incident plane waves.

The generation of spectrum compatible time histories used recorded motions as the starting "seed". This set of real time histories was from the Kern County, California earthquake of July 21, 1952 recorded at Station 95 (Taft Lincoln School Tunnel). The basic procedure was to adjust the frequency content of the initial time history in an iterative fashion until the response spectrum of the modified time history became close to the target spectrum across the frequency range of interest. This adjustment was performed in the frequency domain on the Fourier amplitudes computed through Fast Fourier Transform of the time history. The process preserved the phase angles of the recorded motion. The end result of this exercise is shown in the form of time history trace in Figure 4.1. Each component comprised 2000 time points digitized in time increments of 0.01 seconds. Figures 4.2 through 4.4 compare the response spectra from the adjusted time histories to the target spectra. Statistical independence of the three time histories was assured through a cross-correlation check.

For the purpose of the margins study, there was no explicit requirements for Power Spectral Density (PSD) checks. The possibility of power deficiency in any frequency band was precluded by ensuring that the response spectra remained close to the target.

4.2 DETERMINATION OF SOIL PROFILE

Based on soil information contained in Reference 17, the low strain soil profile is comprised of a 54-foot soft layer with shear wave velocity of 750 ft/sec overlying a 406 foot layer of harder soil with shear wave velocity of 3,600 ft/sec. For all practical purposes, the 406 foot layer may be considered as a half space. The low strain properties are given in Table 4-1.

Table 4-1
LOW STRAIN SOIL PROPERTIES

Layer	Thickness (ft)	Unit Weight (ksf)	Poisson's Ratio	Vs (ft/sec)	Vp (ft/sec)
1	54	0.125	1/3	750	1,500
2	406 (HS)	0.130	1/3	3,600	7,200

To account for the primary nonlinearities in soil behavior under seismic loading, soil properties are consistent with the level of shear strain associated with the assumed seismic motions. Since no specific soil degradation curves existed for the site, the curves corresponding to the mean values given in Figures 7 and 10 of Reference 18 were used to develop the strain compatible soil properties. This step, also known as a site response study, involved one-dimensional wave propagation analyses using computer program SHAKE (Reference 19).

Best estimate strain compatible shear moduli and damping values were calculated using CR-0098 input motion. The lower bound and upper bound profiles were calculated by directly scaling the best estimate strain compatible shear moduli by 2/3 and 3/2, respectively. Note that a factor of 2/3, instead of 1/2 as recommended in Reference 2, was used. The factor of 2/3 was judged to be appropriate for the pile founded structures due to the relative softness of the upper soil layer. As will be seen later, the horizontal foundation impedances for these structures were governed mostly by the properties of the upper soil layer. The piles did not contribute much in the horizontal direction. Also, using a factor of 2/3 resulted in a slightly higher response than a factor of 1/2. The upper bound factor of 1.5 on the best estimate shear modulus was deemed an adequate representation of the upper bound. Table 4.2 lists the strain compatible soil properties for the best estimate, lower, and upper bounds in the SMA approach.

4.3 DEVELOPMENT OF FOUNDATION IMPEDANCES

All of the structures with the exception of the Circulating Water Intake Structure (CWIS) are founded on flexible vertical piles. In the analysis, the upper ends of the piles were assumed to be connected to a rigid surface foundation mat. The piles traverse the soft 54-foot upper layer to penetrate at least 11 feet into the bearing stratum. Computation of foundation impedances used computer program SASSI (Reference 20). As noted earlier, all of the major structures under study were supported on piles. Program SASSI accounted for the existence of piles, including the pile-soil-pile dynamic interaction effects (also referred to as group effects) by the procedures in the following paragraph. A common and appropriate assumption made in these calculations was that of a rigid foundation mat supported by group of flexible piles. It was found in all cases that wave scattering effects were minimal for horizontal motion. This was because the impedance terms for horizontal displacements were mostly contributed by the soil medium, implying that the piles were flexible in the lateral direction and their deformation was driven by the surrounding soil. Therefore, foundation input motion may be assumed to be the same as the free-field for horizontal excitation.

The CWIS is embedded to a depth of 54' with the base founded directly on the lower firm stratum. Due to the softness of the upper soil layer, the characteristics of the foundation impedance would be controlled primarily by the properties of the stiff lower stratum. Hence, it was appropriate to first treat the CWIS foundation as founded on the surface of a uniform halfspace having the properties of the lower stratum, then correcting the surface foundation impedances for the effects of the soft upper layer. Another concern with respect to the CWIS is that deconvolution of the CR-0098 surface motion down to foundation level resulted in unrealistic amplification of high frequency contents on the basemat. This problem arises when a design type ground motion (with broad frequency content) is deconvolved through a soft soil layer that behaves as a low pass filter, and has previously been noted by Roesett in Reference 27. The upper soil layer has a fundamental frequency of about 2.3 Hz (54' thick @ Vs=500 fps). Therefore for the CWIS, the seismic input was directly applied at foundation level, i.e. no credit was taken for any reduction of motion due to embedment (soil support on the side of the basement walls).

The turbine building is a lightweight flexible structure for which SSI effects may be neglected. This fact was recognized in the original FSAR model. The main structural columns of the turbine building class I bay was founded on 12 independent footings, interconnected by ground beams. Therefore, this structure was analyzed in the fixed base condition.

4.4 DEVELOPMENT OF STRUCTURE MODELS

The development of the structure models used the original FSAR building models as a starting point. The original models for all structures, with the exception of the Turbine Building Class I Bay, were one dimensional stick representations for predicting horizontal responses only. The Turbine building was represented by a three dimensional beam model. These models were reviewed in detail to validate assumptions in developing the mass and stiffness characteristics. Also, where feasible, 3-D stick models were constructed from the 1-D representation including any building eccentricities. The process involved close interaction with plant personnel familiar with changes in equipment mass that have taken place since the development of the original FSAR models. In all cases, structural drawings were reviewed to identify major shear walls, or primary lateral and vertical load resisting elements. Equivalent stiffnesses and lumped masses were computed and compared against previously reported quantities. Foundation translation masses, not needed in the previous rocking spring model, were computed. Rotary masses from previous models were checked. In situations where differences existed, best estimate values were assumed. For all concrete structures, best estimate concrete modulus was used based on the latest recommendations in Reference 21. Median structural damping of 7% was assumed in the SSI analyses.

4.5 ANALYSIS OF THE COUPLED SOIL-STRUCTURE SYSTEM

The response analysis of the coupled soil-structure system was performed using the SSIN module from the CLASSI (Reference 22) family of computer codes. This step combined all the elements of the sub-structure approach described in Sections 3.0 through 6.0, -- *viz.* input motion, foundation impedances, and fixed base structure models -- to yield in-structure motions. The SMA approach required three separate analyses to bound the uncertainty in soil properties.

4.6 SSI ANALYSES RESULTS

In general, it was found that inertial interaction effects were dominant, whereas kinematic interaction was negligible. The overall effects of inertial interaction were twofold:

- Downshift of the fundamental soil-structure system frequency.
- Increase in soil-structure system damping due to radiation and hysteretic damping in the soil.

These inertial interaction effects were not completely accounted for in the original design analysis, resulting in a difference in the predicted dominant soil-structure system frequencies, and overly conservative representation of soil radiation and hysteretic damping. The results of the exercise showed that given the higher level of seismic input in the Review Level Earthquake, the amplified floor spectra were higher than the Housner-based FSAR spectra.

Median-centered floor response spectra suitable for Seismic Margin Assessment to a 0.3g review level earthquake were prepared. These spectra are calculated at equipment damping of 5%, and are the envelopes of three soil cases per SMA requirements. Table 4.3 is a tabulation of the 5% damped peak spectral acceleration and ZPA for the median-centered floor response spectra.

In the case of the Turbine Building where a fixed base analysis was performed, the enveloping was performed over the length and breadth of a given floor. Thus, the envelopes for this structure include the spatial variation of response due to torsion and in-plane flexibility of the floors. The vertical response spectra for the Turbine Building considers out-of-plane slab flexibility.

Table 4-2

STRAIN COMPATIBLE SOIL PROPERTIES FOR SMA APPROACH

Best Estimate					
Layer	Thickness (ft)	Damping Ratio	G (Ksf)	Vs (ft/sec)	Vp (ft/sec)
1	6.0	0.036	1892	698.1	1396.3
2	6.0	0.069	1479	617.2	1234.5
3	6.0	0.092	1220	560.6	1121.2
4	6.0	0.111	1023	513.3	1026.7
5	6.0	0.127	871	473.7	947.4
6	6.0	0.139	763	443.3	886.7
7	6.0	0.148	693	422.5	845.0
8	6.0	0.156	634	404.1	808.3
9	6.0	0.162	597	392.2	784.3
10	Half Space	0.020	52323	3600.0	7200.0

Lower Bound					
Layer	Thickness (ft)	Damping Ratio	G (Ksf)	Vs (ft/sec)	Vp (ft/sec)
1	6.0	0.036	1261.3	570.0	1140.0
2	6.0	0.069	986.0	504.0	1008.0
3	6.0	0.092	813.3	457.7	915.5
4	6.0	0.111	682.0	419.1	838.3
5	6.0	0.127	580.7	386.8	773.5
6	6.0	0.139	508.7	362.0	724.0
7	6.0	0.148	462.0	345.0	690.0
8	6.0	0.156	422.7	330.0	659.9
9	6.0	0.162	398.0	320.2	640.4
10	Half Space	0.020	34882.0	2939.4	5878.8

Upper Bound					
Layer	Thickness (ft)	Damping Ratio	G (Ksf)	Vs (ft/sec)	Vp (ft/sec)
1	6.0	0.036	2838.0	855.0	1710.1
2	6.0	0.069	2218.5	756.0	1511.9
3	6.0	0.092	1830.0	686.6	1373.2
4	6.0	0.111	1534.5	628.7	1257.4
5	6.0	0.127	1306.5	580.1	1160.3
6	6.0	0.139	1144.5	543.0	1086.0
7	6.0	0.148	1039.5	517.5	1034.9
8	6.0	0.156	951.0	495.0	989.9
9	6.0	0.162	895.5	480.3	960.6
10	Half Space	0.020	78484.5	4409.1	8818.2

Table 4-3

SUMMARY OF PEAK SPECTRAL ACCELERATION (D=0.05) AND ZPA FOR SEISMIC MARGINS SPECTRA

Structures	5% Damped Peak Spectral Acceleration			Zero Period Acceleration		
	N-S	E-W	Vertical	N-S	E-W	Vertical
Reactor Auxiliary Building						
Basemat	0.863	0.843	0.459	0.305	0.323	0.154
El. 246′	0.881	0.898	0.476	0.317	0.342	0.159
El. 262′	0.891	1.001	0.468	0.324	0.357	0.163
Containment Building						
Basemat	1.259	1.247	0.551	0.263	0.325	0.177
El. 289′	1.940	1.906	0.581	0.342	0.338	0.184
El. 352′	2.553	2.550	0.597	0.465	0.460	0.186
El. 401′	3.022	3.104	0.604	0.589	0.589	0.187
Inner Structure						
Basemat	1.259	1.247	0.551	0.263	0.325	0.177
El. 232.5′	1.369	1.349	0.553	0.262	0.324	0.178
El. 248′	1.480	1.458	0.556	0.278	0.332	0.180
El. 272.5′	1.649	1.630	0.559	0.311	0.348	0.183
El. 306.5′ (Crane)	4.591	3.620	0.563	0.729	0.685	0.192
Circulating Water Intake Structure						
Basemat	0.765	0.861	0.529	0.332	0.344	0.192
El. 206′	2.853	1.295	0.555	0.566	0.394	0.202
El. 226′	3.657	1.587	0.564	0.682	0.450	0.206
Turbine Building						
Foundation	0.640	0.640	0.427	0.300	0.300	0.200
El. 242.5′ (mid-span of beam)	2.634	1.941	1.530	0.460	0.492	0.415
El. 242.5′ (near column)	2.634	1.941	0.707	0.460	0.492	0.220
El. 262.5′ (mid-span of beam)	3.642	3.296	2.755	0.820	0.642	0.455
El. 262.5′ (near column)	3.642	3.296	0.719	0.820	0.642	0.220

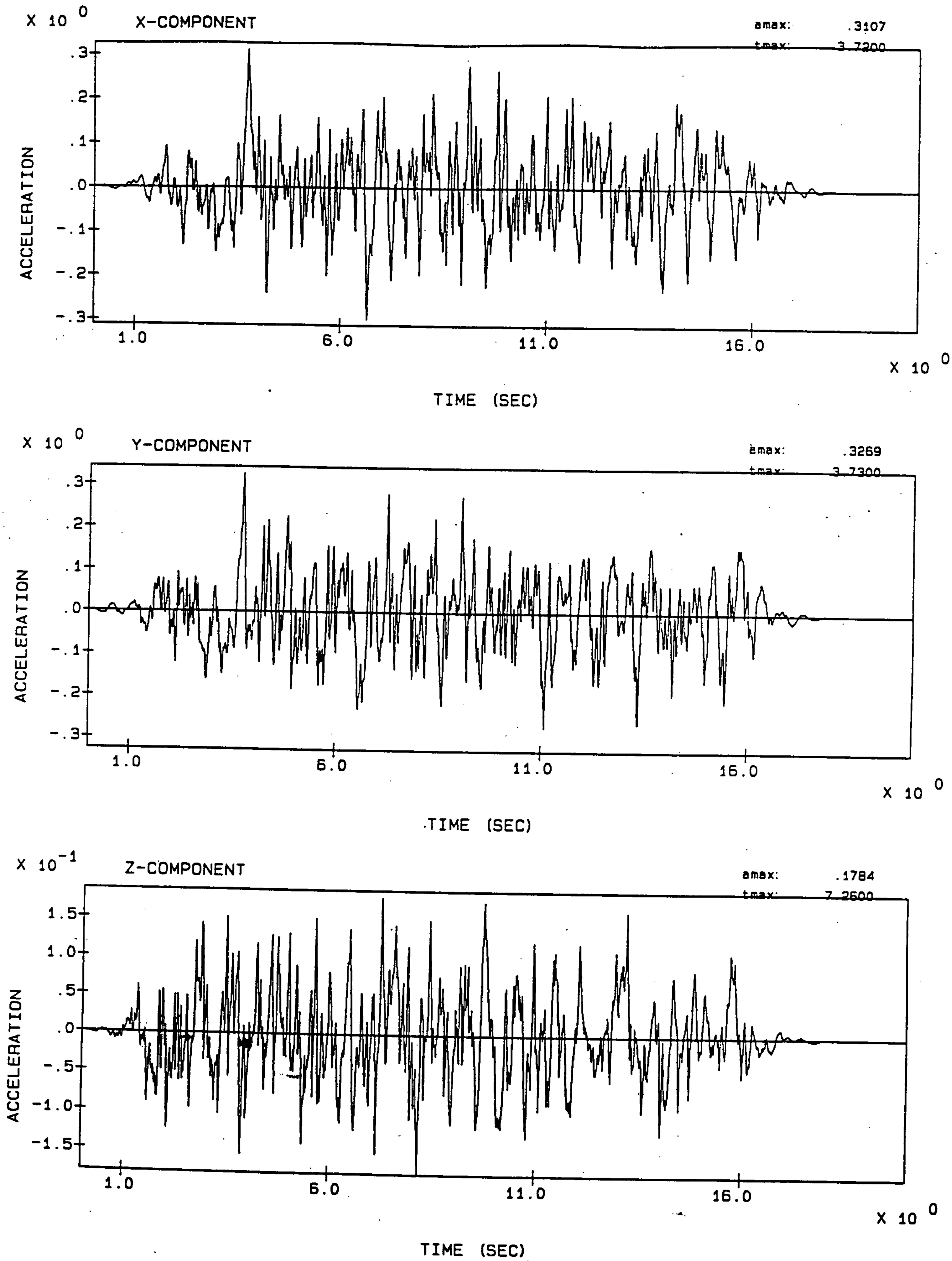
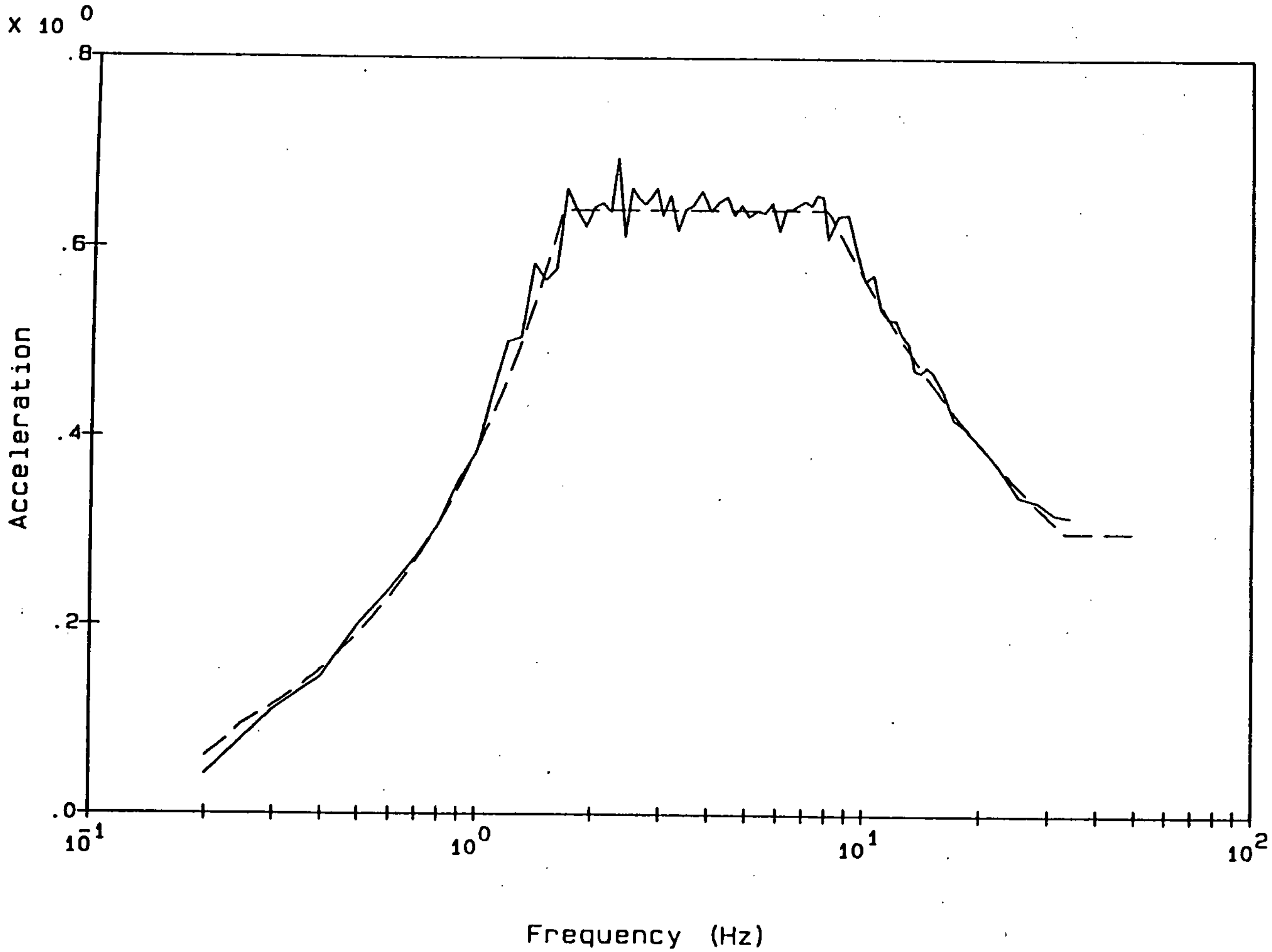


Figure 4-1: Taft Fitted to Median CR-0098 Spectra (5% Damped)



Legend:

X-DIR, MODIFIED TAFT ————

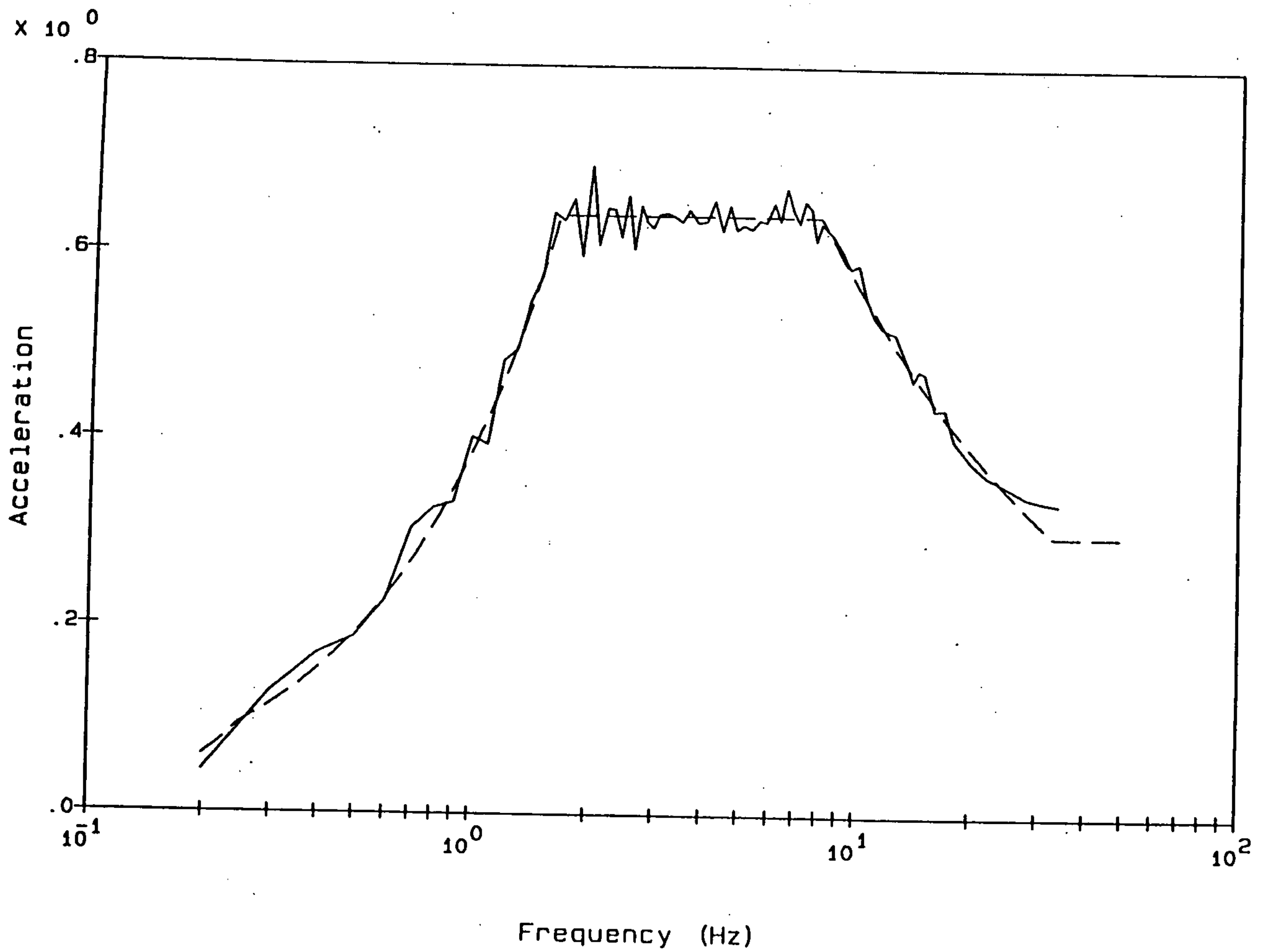
CR-0098 HORIZ — — — —

Notes:

Acceleration in g's

Spectral acceleration at D=0.05

Figure 4-2: Comparison of Artificial T/H to Median CR-0098 Target (X-Dir)



Legend:

Y-DIR, MODIFIED TAFT ————

CR-0098 HORIZ — — — —

Notes:

Acceleration in g's

Spectral acceleration at D=0.05

Figure 4-3: Comparison of Artificial T/H to Median CR-0098 Target (Y-Dir)

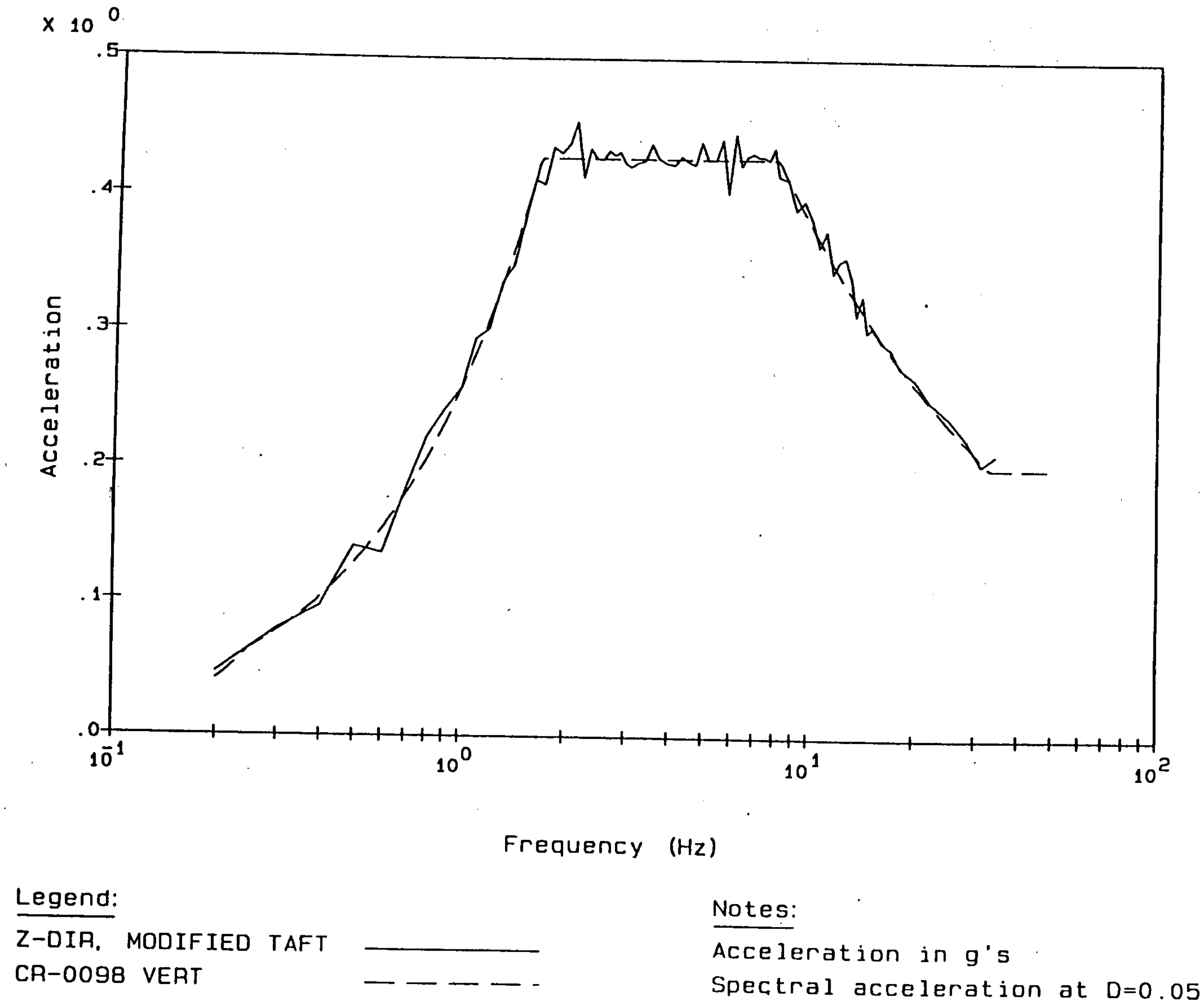


Figure 4-4: Comparison of Artificial T/H to Median CR-0098 Target (Z-Dir)

5. SEISMIC MARGIN ASSESSMENT SCREENING AND WALKDOWN

Section 5 summarizes the Seismic Review Team (SRT) walkdowns. The activities include selection of the SRT, walkdown preparation and pre-screening, establishment of screening criteria, and walkdown results.

5.1 SEISMIC REVIEW TEAM

The Seismic Review Team was assembled following guidance from EPRI NP-6041 (Reference 3). Members of the teams were typically engineers employed by EQE International, Inc. and Carolina Power and Light. These individuals were selected based on their experience and expertise.

Each walkdown team included a minimum of two SRT members who had completed the Seismic Qualification Utility Group (SQUG) Walkdown Screening and Seismic Evaluation training course. Most SRT members also completed EPRI's add-on training for IPEEE. Joint walkdown teams generally consisted of at least one EQE Engineer and at least one CP&L Engineer. The following persons participated in the SRT walkdowns:

- Jeffrey H. Bond
- Steven R. Bostian
- Leo J. Bragagnolo
- Ronald W. Cushing
- James R. Disser
- Gregory S. Hardy
- Daryl W. Hughes
- Ronald L. Knott
- Kelly L. Merz
- Robert N. Panella
- Kevin N. Poythress
- Thomas R. Roche

Among all the team members there is strong experience in each of the areas listed below:

- Knowledge of the failure modes and performance of structures, tanks, piping, process and control equipment, and active electrical and mechanical components during strong earthquakes.
- Knowledge of nuclear design standards, seismic design practices, and equipment qualification practices for nuclear power plants.
- Ability to perform fragility evaluations including structural/mechanical analysis of essential elements of nuclear power plants.
- Knowledge of the plant system functions and normal and emergency operating procedures.

The qualifications of each of the CP&L and EQE seismic walkdown team members are presented in Appendix A.

5.2 WALKDOWN PREPARATION AND PRE-SCREENING

Pre-screening of success path components was performed to ensure efficiency in the walkdowns and evaluations with a goal of completing the maximum amount of data entry in advance of the walkdown. This was accomplished by incorporating existing data onto the seismic IPEEE documentation forms prior to the walkdowns. Data that was reviewed consisted of the Final Safety Analysis Report (FSAR), design criteria, stress reports, equipment qualification reports (testing and analysis), structures and equipment support drawings, equipment location drawings, anchorage calculations, and records from other related programs previously performed at Robinson. An initial walkdown was performed by CP&L and EQE personnel as part of the pre-screening task to review the SSEL and to group items according to the "Rule of the Box."

Pre-screening was performed with three purposes in mind:

- To identify critical failure modes to be specifically reviewed on the walkdown..
- Assemble qualification and installation data for use as a basis for screening in the margins review.
- To provide data to be utilized in HCLPF calculations.

A considerable amount of information was extracted from the existing documentation and was subsequently recorded on the Screening and Evaluation Work Sheets (SEWS) prior to commencing the detailed walkdowns. Information entered into SEWS during prescreening was intended to provide available data to the SRT to assist in equipment screening. The information is not intended as the sole basis for screening, but assists the SRT in their review.

Pre-screening was enhanced by the use of the software program EHOST. EHOST is a data base program which has been adapted specifically for use in performing USI A-46 and IPEEE evaluations. The program is set up so that the data is incorporated onto SEWS forms which are consistent with those recommended in EPRI NP-6041 (Reference 3). In this manner the walkdown teams using portable computers with the companion program EWALK were then able to work more efficiently by having access to SEWS that had already been partially completed.

5.3 SCREENING CRITERIA

The Robinson seismic IPEEE was completed following the EPRI seismic margins methodology recommended by NUREG-1407 (Reference 2) for a full-scope plant.

Civil structures, equipment and subsystems were screened following the methodology provided in EPRI NP-6041 (Reference 3). Screening criteria are provided in Tables 2-3 and 2-4 of Reference 3 for civil structures and equipment and subsystems, respectively. The criteria corresponding to 5 percent-damped peak spectral acceleration less than 0.8g were used for Robinson based on the RLE. The guidelines are supplemented by Appendix A of the EPRI seismic margins methodology which provides the basis for the seismic capacity screening guidelines.

Combined A-46 and IPEEE walkdowns were performed using Screening Evaluation Work Sheets (SEWS) contained within the GIP (Reference 8), enhanced to capture issues specific to EPRI NP-6041. SEWS were loaded into EQE's computer program EWALK for field screening and data collection using portable pen-based computers. Prescreening information was downloaded from the database program EHOST. The effectiveness of in-plant reviews was improved by access to SEWS forms enhanced with plant specific data. This also allowed the walkdown teams to be alerted to specific concerns that may have been identified during pre-screening.

The SRT had liberal access to plant design drawings and analyses to use in conjunction with the screening criteria. Much of this information was reviewed and summarized in the SEWS prior to the field walkdowns. This provided the SRT with information such as:

- Seismic coefficients used in motor operated valve weak-link analyses to verify that valve mass and eccentricity guidelines were satisfied.
- Valve yoke material from vendor drawings to identify or rule out cast iron material.

5.4 SEISMIC MARGIN WALKDOWN RESULTS

Robinson combined A-46 and seismic margin walkdowns commenced during RFO-15 (September, 1993) and were completed in early 1995.

The walkdown concentrated on the strength and load path of the equipment to the structure as well as function and integrity. The review of equipment anchorage was a prime objective for the walkdown teams. The anchorage evaluation addressed both physical attributes of the anchorage installation and the capacity relative to the SSE for A-46 anchorage evaluations as well as the postulated demand at the RLE. Anchorage capacities were evaluated in accordance with the GIP for SSE seismic demand. Items judged to have relatively low margin or suspect configurations were also evaluated at RLE seismic demand levels.

Interaction reviews were performed to identify falling, impact, spray and flood issues that could affect success path items.

The Seismic Review Team noted housekeeping issues while performing A-46/Seismic IPEEE walkdowns for the SSEL components. Identified housekeeping issues were typically recorded on the SEWS forms for specific safe shutdown equipment when the Seismic Review Team judged the housekeeping issue to potentially impact safe shutdown equipment. Additionally, issues were noted for general cleanliness and safety reasons and were identified to the plant as items that could easily be remedied without the necessity of preparing work tickets or modifications.

The Seismic Review Team identified several housekeeping issues in the Control Room area. These items included unanchored book shelves and filing cabinets that could potentially interact with a non safety control board. The Seismic Review Team recommended that the book shelves and file cabinets be relocated to another place in the control room away from the control board. The Team also noted a wall hung fire extinguisher that was not positively secured to the support bracket. The Team recommended that the fire extinguisher be more positively attached to the support bracket so that the extinguisher could not lift off of the hook from an earthquake.

The Seismic Review Team also noted that a table and chair in the Hagan Room were not anchored to the floor. The Hagan Room is the location for approximately thirty (30) cabinets that house instrumentation and control circuitry for the control board indicators in the adjacent control room. The table and chair were located approximately five (5) feet from the nearest Hagan cabinet which is also not on the safe shutdown equipment list. However, this cabinet is bolted to adjacent cabinets that are on the list. The Team recommended that the table and chair be relocated within the room or moved outside of the room if possible.

The Cable Spread Room was also observed during the walkdown of the Auxiliary Relay Racks for any housekeeping issues. The Seismic Review Team noted that a computer printer and several filing cabinets were not anchored. Although there were no safety related, safe shutdown cabinets or panels within the general vicinity of these unanchored items, the Seismic Review Team recommended that these items should be anchored, relocated to another place in the room, or removed from the room altogether.

Lastly, the E1/E2 Emergency Buss Room was reviewed for housekeeping issues during electrical safe shutdown equipment walkdowns. A portable steps assembly used by plant personnel to operate overhead equipment was stored in the room. Also, several tools used for breaker installation and a fire extinguisher were stored on the barrier fence of the room adjacent to the MCC-9 cabinet. The Team recommended that the steps be relocated away from any safe shutdown equipment. The tools and fire extinguisher were identified on the SEWS for the MCC-9 as a potential interaction issue and a work ticket has been written to resolve this condition.

Other housekeeping issues that were identified by the Seismic Review Team that were considered significant interaction issues were included with specific safe shutdown components on their respective SEWS forms.

Cable tray and conduit were reviewed in accordance with the GIP screening guidelines as part of the A-46 program.

Other suspended systems, piping and ductwork were evaluated on a sampling basis in the plant. A general survey was performed to obtain an overview of the suspended system construction throughout the plant. This included a review of the variety of system layouts, support configurations, and construction details. The inspection also included known concerns for suspended systems, such as taut cables, sharp edges, or overloading of cable trays and supports, and potential anchor point displacements.

The ceiling above the control room was also reviewed to verify if the light fixtures and ceiling grid were adequately supported, and to evaluate the potential for ceiling panels to fall. Mod M-1010 contains the information for the seismic design of the unistrut grid layout in the control room. This design was prepared and installed during the Fall, 1990 refueling outage. The lighting layout has been modified since that installation, but the structural ceiling grid remains unchanged.

Containment penetrations were reviewed on an area basis to identify anomalies that might affect containment performance. Concerns such as falling and differential building displacement were considered. Also reviewed were displacement concerns between the containment shell and internal structure. Containment isolation valves were also reviewed on a walk-by basis based on the caveats listed on the valve SEWS.

Some equipment was found to be inaccessible during the walkdowns due to high radiation. The following equipment was inaccessible during the walkdown:

- CVC Regenerative Heat Exchangers
- TE-123
- Excess Letdown Heat Exchanger

In these cases, thorough drawing reviews were performed as well as reviews of equipment known to be similar in design and configuration. These inaccessible equipment, with the exception of the CVC Regenerative heat exchanger were screened out based on these reviews.

The CVC Regenerative heat exchanger could not be screened based on available documentation since the load path between heat exchanger shells and the rack could not be confirmed. The configuration is addressed in Section 6, Assessment of Elements Not Screened Out.

At the conclusion of plant walkdowns SRT members, including senior level participants from CP&L and EQE, convened to complete the ranking and screening task. SRT members reviewed SEWS and categorized components into the following resolution categories:

- Screened out by the SRT
- Housekeeping or maintenance issue

- Repairs or modification required
- Specific issues require clarification
- Selected for HCLPF evaluation

Seismic margin walkdown results are summarized for structures and equipment and subsystems in Tables 5-2 and 5-3, respectively.

Table 5-2 lists civil structures following the format of EPRI NP-6041, Table 2-3, along with screening results for the Robinson plant.

Table 5-3 lists equipment and subsystems following the format of EPRI NP-6041, Table 2-4, along with screening results for Robinson. Also included in Table 5-3 are A-46 outlier issues and associated resolutions.

Unscreened equipment and subsystems are categorized below:

- 32 specific SSEL components grouped into 27 categories had minor interaction, housekeeping or maintenance issues that will be resolved through routine maintenance activities via work requests (WR/JO).

- 34 specific SSEL components grouped into 21 categories were identified as requiring repairs or modifications.

- 20 calculations were performed for HCLPF evaluation for 47 specific SSEL components. The evaluations are summarized in Section 6.

5.5 STRUCTURES

Robinson structures include the reactor containment structures, concrete shear wall structures and the steel frame structures. The review of structures and pile foundations are summarized in the following sections.

5.5.1 Reactor Containment Structures

Reactor containment structures include the containment building and internal structures. The Robinson containment is a Seismic Category I steel lined reinforced concrete structure. The containment structure encloses the concrete structures and structural components which comprise the containment internal structures. Detailed descriptions of the containment building and internal structures are provided in Sections 2.6.1 and 2.6.2, respectively.

Reactor containment structures are screened based on the FSAR and Reference 3, Table 2.3 and Appendix A.

5.5.2 Concrete Shear Wall Structures

The Reactor Auxiliary Building, Fuel Building and Service Water Intake Structure are described in Sections 2.6.3, 2.6.4 and 2.6.6, respectively. The Seismic Category I structures were designed for the Robinson 0.2g design basis earthquake.

Concrete shear wall structures are screened based on the FSAR and Reference 3, Table 2.3 and Appendix A.

5.5.3 Steel Frame Structures

Steel frame structures include the Seismic Class I and Class III portions of the Turbine Building. The structures are described in Section 2.6.5.. The Class I and Class III portions of the building were dynamically analyzed using the 0.2g design basis earthquake

Steel frame structures are screened based on the FSAR and Reference 3, Table 2.3 and Appendix A.

5.5.4 Pile Foundations

Pile loads for the SMA earthquake loading were evaluated against the interaction diagram developed for the piles and illustrated in Figure 3.8.5-21 of the UFSAR. Loading conditions considered in the pile evaluation included seismic and dead weight. Piles for both the RAB and the Containment were determined to have adequate reserve capacity.

Resistance to lateral shear from structural embedment was neglected in the IPEEE assessment. This is similar to an assumption in the original design and is felt to be quite conservative for the containment structure. Treatment of differential movement between the firm soil and surface was accounted for based on the variation of soil shear strains from the SSI analysis. This more realistic treatment gave nearly a 75% reduction in differential movement.

5.6 SOILS EVALUATION

A review was performed of existing information regarding soils-related issues as part of the examination under the IPEEE program. The primary focus was on identification of liquefaction susceptibility at the Robinson plant site. Related to this effort was the identification of potential sources of seismic instability of Robinson Dam. Robinson Dam is required to function in order to assure adequate supply of cooling water for the Robinson plant.

The approach was based upon guidelines contained in Reference 44 for assessing liquefaction susceptibility utilizing standard penetration test data and previous geotechnical reports generated for the Robinson site. Evaluation of Robinson Dam was based on simplified approaches developed by Newmark (Reference 46).

Borehole investigations at the Robinson site indicate that site conditions are moderately consistent. However, no continuous deposits can be identified by the borehole records. The top 50 feet of soil contains various beds of moderate to dense sands interspersed with layers of relatively weak to moderate strength silty sands, sandy silts, and silty clays.

5.6.1 Identification of Potential Failure Modes Related to Site Soils

Critical structures at the Robinson site are pile supported with the piles founded in a hard clay layer approximately 50 feet below grade. The design of the pile foundations conservatively neglected any support that might be provided by shallower soil deposits. Furthermore, the piles were designed to take the entire lateral load from seismic forces on the structures. This is also conservative since the base of the reactor structure is generally 10 feet below grade and is at least 20 feet below grade beneath the reactor. Given this configuration, the surface soils will carry substantial lateral load through direct bearing with the structure.

The only critical slope or embankment identified at the site is associated with Robinson Dam. The design of the dam has a factor of safety of 1.02 against slope failure under the hypothetical earthquake having a peak ground acceleration (PGA) of 0.2g.

In examining potential failure modes associated with beyond design basis events, three potential failure modes have been investigated:

1. Liquefaction beneath critical structures resulting in large reductions in the lateral support provided to piles

2. Slope failure of Robinson Dam

3. Lateral spread deformations associated with liquefaction of deposits near the shore of Robinson Lake

5.6.2 Evaluation of Potential Failure Modes

A summary of the evaluations performed for the failure modes identified above is provided.

Liquefaction

Most of the evaluation effort focused on assessing the potential for a significant liquefaction hazard at the Robinson site. Assessment of liquefaction was based on Dames and Moore boring Logs 101-117 (Reference 17) and the Raymond Logs of Borings 1-5 obtained for Westinghouse Electric (Reference 50). These logs provided the basis for estimating total unit weights associated with each soil type, the equivalent blow counts corresponding to a standard penetration test (SPT), and the depth to the water table. Based on this review, the depth to the water table was found to range from ground surface to about 4 ft, with a predominant depth of 2 ft. Available grain size distributions (Reference 17) indicated that the amount of fines (i.e., percent passing the No. 200 sieve) ranged from 10% to 15% for poorly graded (SP) and 15% to 25% for silty sands. Silts (ML) are judged not susceptible to liquefaction at the Robinson site.

Initial review focused on an earthquake with a surface horizontal PGA of 0.3 g. Both the Robinson site and the Savannah River Site (References 53 & 54) are located within the same seismotectonic province. Based on this comparison, the magnitude of the earthquake associated with the above peak acceleration was taken as the mean magnitude associated with a 1 x 10^{-4} annual frequency of occurrence of peak ground acceleration as derived from seismic hazard analyses performed for the Savannah River Site. According to the Savannah River reports, this magnitude was estimated to be m_{bLg} 5.9, giving a moment magnitude of about M_w 5.5 based on relationships developed by Atkinson (Reference 47), EPRI (Reference 44), and Atkinson and Boore (Reference 48). The magnitude of 5.5 was adopted to be consistent with the M_L magnitude used by Seed and others (Reference 49). Magnitude scaling factors as developed from historical data (Reference 79) were used in the liquefaction evaluation.

Blowcount information from the available borehole data logs for the Robinson site were converted to equivalent SPT values for the conditions of 1 ton per square foot overburden pressure and corrected to account for the effects of fines content and earthquake magnitude. These modified blowcounts were taken to be equivalent to SPT data collected with a hammer having an energy efficiency of 60%.

Potentially liquefiable soil deposits for each boring log were identified by comparing the corrected blowcounts with threshold values corresponding to the onset of liquefaction.

This comparison indicates all of the data points fall in the non-liquefiable region. The few points that do fall below this threshold are considered statistically insignificant. Accordingly, the acceleration threshold for liquefaction at the Robinson site is considered to be above 0.3g.

Slope Stability

Robinson Dam is necessary to maintain cooling water supply to the power plant. Dynamic analysis of the dam indicate a factor of safety of 1.02 against slope failure for an earthquake with a PGA of 0.2g. To assess the impact of a 0.3g earthquake, reference was made to the relationship developed for unsymmetrical block sliding (Reference 46).

The resulting value for maximum earthquake related displacement at 0.3g is estimated to be less than 1 inch. This estimate is based on the assumption that liquefaction is not a concern at 0.3g for Robinson Dam. Considering the excavation plan for the dam and the results of the site evaluation of liquefaction susceptibility, liquefaction is not judged a likely concern at 0.3g for Robinson Dam. From descriptions of the physical layout of the dam and review of its operation, the small amount of displacement estimated using this approach is far less than that judged to be necessary for loss of dam function. Therefore, the dam is judged acceptable for 0.3g PGA.

Lateral Spread Movement

If the reduction in soil shear strength from partial or complete liquefaction results in an unstable slope configuration, lateral spread movement may occur. Based on the results of the liquefaction assessment, lateral spread movements are not likely for PGA less than .3g. Therefore lateral spreading is not a concern at or below 0.3g.

5.7 NSSS REVIEW

The NSSS review included screening of NSSS primary coolant systems and supports based on EPRI NP-6041 and information provided in the FSAR, a brief review of reactor internals based on the FSAR, and verification that the control rod drive mechanisms have an upper lateral support.

5.7.1 NSSS Primary Coolant Systems and Supports

The NSSS primary coolant system (piping and vessels) is screened from further review based on Table 2-4 of EPRI NP-6041.

The FSAR summarizes seismic analysis performed on the reactor coolant loop which consists of the reactor vessel, steam generator, reactor coolant pump, the pipe connecting these components, and the large component supports. The components and piping were modeled as a system of lumped masses connected by springs whose values were computed from elastic properties inputs. A simplified support model was arrived at by representing the structural support system as equivalent springs rather than as member beams and columns.

The analysis was performed using a proprietary computer code called WESTDYN (Reference 38). As input, the code uses system geometry, inertia values, member sectional properties, elastic characteristics, support and restraint characteristics, and the appropriate CP&L seismic floor response spectrum for 0.5 percent critical damping. Both horizontal and vertical components of the seismic response spectrum were applied simultaneously. The seismic shock spectra were applied simultaneously along the Y and Z axis. Previous analysis indicated the Z direction to be the most critical horizontal direction for maximum pipe stress. Results of the analyses are summarized in Table 5-1 (Reference FSAR, Table 3.7.3-2).

NSSS supports are screened from further review based on low stresses summarized in Table 5-1.

Table 5-1
PRIMARY COOLANT LOOP SUPPORT LOADS

Support	Analysis	Fx (kip)	Fy (kip)	Fz (kip)	Mx (kip)	My (kip)	Mz (kip)	Stress (ksi)
SG Lower	Original Revised	0 0	106 123	115 116	5069 5087	2064 2052	2253 2234	4 4
SG Upper	Original Revised	4 4	0 0	204 204	0 0	0 0	0 0	4 4
RCP	Original Revised	31 31	83 83	131 131	29 29	974 971	680 715	5 5
RPV	Original Revised	26 26	33 33	3 3	2447 2422	870 862	4672 4690	2 2

Table 5-1 Notes:
1. From FSAR Table 3.7.3-2
2. Deadweight and seismic loads combined.

5.7.2 Reactor Internals

The FSAR describes a proprietary seismic analysis of the Cycle 4 core (Exxon Fuel). The objective of the analysis was to determine fuel rod stresses, guide tube stresses, and interactive grid spacer loads during a lateral core motion seismic event. Fuel rod integrity, core coolable geometry, and the ability to insert control rods, must be maintained during this event.

The results of the analysis show that fuel rod, grid spacer, and guide tube integrity is maintained during a 0.4g lateral seismic event. Maximum guide tube, fuel rod, and grid spacer stresses occur in the fuel assemblies adjacent to the core boundary.

A dynamic test of the fuel bundle, a static load-deflection grid spacer test and dynamic spacer tests were performed to determine input fuel assembly dynamic properties. These also provide stress-strain information on the guide tubes, spacers, and fuel rods which verify the structural adequacy of the fuel assembly during a lateral seismic event.

The reactor internals HCLPF capacities are probably larger than 0.3g based on EPRI NP-6041, Appendix A and limited available information. Further research is not expected to identify any significant vulnerabilities.

5.7.3 Control Rod Drive Mechanisms

Robinson control rod drive mechanisms (CRDM) are restrained by 4 seismic ties (Reference 24). The CRDM are therefore screened based on EPRI NP-6041, Table 2-4.

5.8 DISTRIBUTION SYSTEMS

The following sections address the distribution systems; cable tray and conduit, HVAC duct and piping.

5.8.1 Cable Tray and Conduit

Cable tray and conduit were reviewed following USI A-46 criteria. Cable tray were reviewed in May, 1989 while the GIP criteria was under development. Conduit was reviewed in September 1994 following GIP criteria. A brief review of cable tray was also included in the 1994 walkdowns.

As part of the cable tray and conduit walkdowns, sixteen (16) items were identified as requiring repairs and/or rework to increase the seismic margin and bring the current conditions up to the original design intent. These items are in addition to other cable tray and conduit raceway items that were identified during walkdowns performed during May 1989 by EQE as a preliminary study during GIP methodology development. Items identified as requiring rework during that walkdown are incorporated into modification M-1114 scheduled to be implemented during Refueling Outage 15 in Spring 1995. A description of the remaining identified conditions is presented in Table 7-1 of the A-46 Seismic Evaluation Report.

Cable tray and conduit are screened based on the A-46 review, repairs, rework and Reference 3, Table 2.3 and Appendix A.

5.8.2 HVAC Duct

Heating ventilating and air conditioning (HVAC) ducting and in-line components such as dampers were reviewed on an area basis during SRT equipment and subsystem walkdowns to identify any anomalies that could lead to failure. Both the function of essential HVAC systems and the potential for failure and falling of ducting on success path equipment were considered. HVAC ducts in the diesel generator rooms were identified as potential interaction sources.

The SRT noted that some of the installed HVAC ductwork for the diesel generators was installed above buss ducts to the diesel generator control switchgear. The trapeze type rod hangers were evaluated and found acceptable.

HVAC duct are screened based on Reference 3 and the evaluation outlined above.

5.8.3 Piping

Piping systems were reviewed on an area basis during SRT equipment and subsystem walkdowns. The SRT looked for any anomalies related to potential displacement induced failure modes. The only piping issue selected for further evaluation was the in-line RHR pumps and associated piping. The pumps are guided in-line, rather than anchored to foundations. Design basis piping analyses were reviewed and scaled to the RLE resulting in a HCLPF capacity greater than 0.3g.

Additionally, the SRT looked for potential failure modes of piping system appurtenances such as instrument tubing and associated instruments, vent valves and drain valves. Seismic interaction and seismic anchor motion were considered potential failure modes for small bore lines attached to larger piping systems. No anomalies that could lead to the loss of a pressure boundary of a success path list system were observed.

Containment penetrations were also reviewed on an area basis to identify any anomalies that may effect containment performance. Anomalies such as seismic interaction (falling) and differential building displacement were considered. A walk-by of containment isolation valves for the intent of the caveats identified on the valve SEWS was also performed. No anomalies that could effect containment performance were observed.

Robinson piping was screened from further review based on qualification programs outlined in Section 2.11, Appendix A of Reference 3, and SRT walkdowns.

5.9 OTHER COMPONENTS

The following sections discuss the in-core flux mapping system and masonry walls.

5.9.1 In-core Flux Mapping System

In 1984, CP&L discovered that the in-core flux mapping system at their Harris plant was not seismically designed. Seismic induced failure of the flux mapping cart above the seal table could cause failures of the flux mapping tubing of fittings which would produce a small break LOCA. This discovery resulted in Information Notice 85-45 (Reference 25).

The Robinson in-core flux mapping system was modified in the late 1980's to provide restraints to resist seismic loads. Hold down restraints are fabricated of angle welded to the cart and bolted to the structure. Four restraints are installed, one at each corner of the cart.

Analyses of the holdown restraints for design basis loads resulted in adequate margin to screen the details at the RLE.

5.9.2 Masonry Walls

EPRI NP-6041 presents the guidelines for assessing various nuclear power plant structures, components, and subsystems for seismic margin. One assessment is masonry walls. Specifically, this document recommends that masonry walls, particularly those walls that are unreinforced or lightly reinforced, be reviewed. Carolina Power and Light Company, in cooperation with Ebasco Services Incorporated, prepared a report in October, 1980 that responded to the requirements of I.E. Bulletin 80-11 (Reference 39). All masonry walls that were in the proximity of safety related systems or equipment in the Reactor Building, the Reactor Auxiliary Building, and the Fuel Handling Building were analyzed and reinforced as necessary with structural steel supports to ensure that these walls will not collapse due the postulated hypothetical earthquake.

Table 5-4 lists walls identified for 80-11 review and analysis.

There were twelve (12) walls that were identified for review as part of the 80-11 project. The location of these walls is shown on drawing CAR-2762-SK-401. However, only six of the walls were actually further evaluated based on their function and the proximity to safety related equipment. These walls identified from Table 5-4 are 1, 2, 3a, 3b, 4, and 6 and are detailed on sketches CAR-2762-SK-402, CAR-2762-SK-403, CAR-2762-SK-404, CAR-2762-SK-404, CAR-2762-SK-405, and CAR-2762-SK-406 respectively. These walls are radiation shield walls that are constructed of multi-wythe solid block and were not intended to function as load bearing elements.

The remaining six (6) walls, identified as 5a, 5b, 5c, 7a, 7b, and 7c, did not receive further evaluation based on their function and their location in the plant away from safety-related equipment. Walls 5a, 5b, and 5c are located in the CV Access Area outside of the Reactor Auxiliary Building entrance and the Reactor Containment Building entrance and are not adjacent to any safety-related equipment in the area. Walls 7a, 7b, and 7c are located in the Fuel Handling Building in the hallway between the Cask and Large Equipment Decontamination Area and the Gas Decay Tank Room. These walls are arranged in a labyrinth configuration and serve as a radiation barrier for the gas decay tanks. The gas decay tanks and the equipment in the cask and large equipment decontamination area are not safety related.

The following codes were used for the design and analysis of the masonry block walls and any associated steel framing:

(1) American Concrete Institute (ACI) 67-23, Concrete Masonry Structures - Design and Construction and 531-79, Building Code Requirements for Concrete Masonry Structures.

(2) American Institute of Steel Construction, (AISC) - Specification for the Design, Fabrication and Erection of Structural Steel for Buildings - Sixth Edition, Revised 1963.

(3) American Welding Society (AWS) D1.1.76 - Structural Welding Code

(4) Phillips Catalog F-1000 dated May 1, 1973.

These walls were evaluated by Ebasco Services Incorporated. Ebasco assumed that the existing masonry walls do not have any horizontal or vertical reinforcing and no credit was taken for the mortar in the vertical joints between the wythes. The walls were first analyzed as single wythe cantilevers and in all cases it was found that structural steel reinforcing supports were required. The walls were then analyzed as single wythes spanning between new structural supports which are connected to existing reinforced concrete walls or floors. The sketches identified above provide details for the installation of structural steel supports for the masonry walls. With these supports added, the calculations show that the masonry walls will retain their function during and after the hypothetical safe shutdown earthquake.

Ebasco evaluated the masonry walls using the Safe Shutdown Earthquake acceleration for Robinson of .2g. The following results were obtained by multiplying a scaling factor times the results determined by Ebasco analysis using the safe shutdown accelerations. The scaling factor was determined by dividing the accelerations for a .3g RLE by the corresponding frequency acceleration values for the .2g earthquake. Further details for the results are shown in Calculation RNP-C/RAB-1049.

WALL 1

Wall tensile stress $f_t = 27.76$ psi $< F_{t\ allow} = 39$ psi

Channel bending stress fb $= 10{,}192$ psi $< F_{b\ allow} = 15{,}700$ psi

Anchor bolt tensile t = 3869# < 12,075# (ultimate and F.S. = 3.12

Anchor bolt shear s = 307# < 4845#

Baseplate thickness treqd = .98" < thickness used T_{used} = 1.0"

Weld size required W_{reqd} = .25" < Weld size used W_{used} = .3215"

WALL 2

Wall tensile stress f_b = 1.94 psi < $F_{b\ allow}$ = 39 psi

Ceiling Anchor bolt shear s = 226# < 3420#

Mid-support bending stress f_b = 26648 psi < $F_{b\ allow}$ = 29260 psi (increased allowable by 33%)

Mid-support anchor bolt tensile t = 4199# < 12,075# (ultimate and F.S. = 2.9).

Mid-support anchor bolt shear s = 1550# < 4845#

Baseplate thickness t_{reqd} = .97" , Thickness used T_{used} = 1.0"

Weld size required W_{reqd} = .1729" < Weld size used W_{used} = .3125"

WALLS 3a AND 3b

Wall tensile stress (between B-1 & floor) f_t = 1.62 psi < $F_{t\ allow}$ = 39 psi

(between B-3 & B-4, B-4 & wall, and B-5 and wall) F_t = 27.13 psi < 78 psi

Support B-2 bending stress f_b = 3521 psi < $F_{b\ allow}$ = 16,600 psi

Support B-3 bending stress f_b = 11,709 psi < $F_{b\ allow}$ = 22,000 psi

Weld size required W_{reqd} = .258" < Weld size used W_{used} = .25"

Anchor bolt tension t = 3715# < 4279#

Anchor bolt shear s = 307# < 10,089#

Baseplate thickness t_{reqd} = .99" < Thickness used t_{used} = 1.0"

Support B-4 bending stress f_b = 13,986 psi < $F_{b\ allow}$ = 16,600 psi

Weld size required W_{reqd} = .13" < Weld size used W_{used} = .25"

B-4 & B-5 baseplate thickness t_{reqd} = .99" < Thickness used T_{used} = 1.0"

B-4 & B-5 anchor bolt tension t = 5330# < 17,115# (ultimate and F.S. = 3.21)

B-4 & B-5 anchor bolt shear s = 755# < 10,089#

WALL 4

Wall tensile stress f_t = 18.6 psi < $F_{t\ allow}$ = 39 psi

Support X-1(X-2)bending stress f_b = 27,298 psi < $F_{b\ allow}$ = 29,260 psi (increased allowable by 33%)

Support X-1 anchor bolt tension t = 1449# < 2054#

Support X-2 anchor bolt shear s = 961# < 3420#

Support X-3 bending stress f_b = 6710 psi < $F_{b\ allow}$ = 22,000 psi

Support X-3 anchor bolt shear s = 961# < 3420#

Support X-3 weld size required W_{reqd} = .046" < Weld size used W_{used} = .25"

WALL 6

Wall tensile stress f_t = 20.0 psi < $F_{t\ allow}$ = 39 psi

Middle support bending stress f_b = 26,439 psi < $F_{b\ allow}$ = 29,439 psi (increased allowable by 33%)

Anchor bolt tension t = 3757# < 12,075# (ultimate and F.S. = 3.21)

Weld size required W_{reqd} = .17" < weld size used W_{used} = .3125"

Anchor bolt tension t = 4228# < 12,075# (ultimate and F.S. = 2.9)

Baseplate thickness t_{reqd} = .65" < thickness used t_{used} = .75"

The values reported above for bending stresses, anchor bolt tension and shear, baseplate thickness, and weld size are all less than the allowable values. These calculated values were based on the use of a scaling ratio of the acceleration values for the .3g Review Level Earthquake to corresponding values for the .2g Design Basis Earthquake. Therefore, the masonry block walls at the Robinson Steam Electric Plant Unit No. 2 are satisfactory for the IPEEE review.

Other masonry in fill panels were installed in penetrations through concrete walls. These panels were evaluated for seismic and penetrator (conduit, pipe, etc.) effects during the Fall 1990 outage to determine if they conformed to the original 80-11 bulletin criteria. CP&L calculation RNP-C/RAB-1038 performed the evaluation of the masonry in-fills based on DBE acceleration values. The evaluation included six (6) cases. These cases are as follows:

1. Case I - 3'x3' In Fill Panel

2. Case II - 3'x1' In Fill Panel
3. Case III - Circular In Fill Panel
4. Case IV - In Fill Panel with 2" Diameter Penetrator
5. Case V - Multiple Penetrators
6. Case VI Brick Over Duct

For Cases I through III, the analysis concludes that the wall will resist all seismic forces.

For Case IV, the two-inch diameter penetrator forces can be resisted by 2 wythes of brick and/or 7 inches minimum of grout. The only way that the brick wall will remain below design allowables for a one-wythe/foam penetration is if the penetrator has a foam seal around its circumference or the span lengths for the piping are short.

For Case V, a sampling of several penetrations with multiple penetrators revealed several items:

a. The actual pipe spans between supports were far less than the spans calculated from the Ebasco chart method;

b. The location of the penetrators tended to be more toward the edges of the penetration rather than at the center; and,

c. The loads applied to the walls created moments in the wall that were far less than the allowable.

Evidence was not available to determine if the brick walls were initially counted on for support during implementation of the chart method. Based on the spans measured, it was judged that the walls were not counted on for support of the penetrators because of the frequency of the supports and the closure of the penetration being one of the last construction items to be accomplished. The as-built spans measured for the penetrators during the sampling process were far less than the allowable chart spans. Based on field observations and sampling and the qualification of three (3) penetrations, the conclusion can be determined that the remaining penetrations with multiple penetrators could be individually qualified to resist all imposed loads.

Case VI concluded that a minimum of two (2) courses of brick must be installed over the ductwork to maintain a stable configuration. The masonry in-fills that were grouped into this case and investigated are acceptable based on repair, no brick installed, or two (2) or more courses of brick installed.

Calculation RNP-C/SPPT-1049 reviewed the results from the DBE analysis by multiplying a scaling factor times the results determined in the calculation RNP-C/SPPT-1038. The scaling factor is the ratio of the .3g RLE acceleration values for the corresponding frequency for the DBE. All six cases remained satisfactory for the use of the scaling factor.

Lastly, a masonry block wall is installed in the control room adjacent to some of the safety-related control and instrumentation panels. This masonry wall is seismically supported with structural steel per design shown on drawing SK-1105-C-1003, Sheets 1 and 2 of 2. Calculation RNP-C/SPPT-1783 provides the analysis and acceptance for the seismic design.

Table 5-2

SUMMARY OF CIVIL STRUCTURES SEISMIC MARGIN EVALUATION

(Format Follows EPRI NP-6041, Table 2-3)

TYPE OF STRUCTURE	DISPOSITION
Concrete containment	Screened based on EPRI NP-6041, Table 2-3. See Section 5.5.1.
Containment internal structure	Screened based on EPRI NP-6041, Table 2-3. The structures was designed for greater than 0.1g. See Section 5.5.1.
Shear walls, footing and containment shield walls	Screened based on EPRI NP-6041, Table 2-3. The structures were designed for greater than 0.1g. See Section 5.5.2. Piles are discussed in Section 5.5.4.
Diaphragms	Screened based on EPRI NP-6041, Table 2-3. Diaphragms were designed for greater than 0.1g. See Section 5.5.2.
Steel frame structures	Screened based on EPRI NP-6041, Table 2-3. Steel frame structures were designed for greater than 0.1g. See Section 5.5.3.
Masonry walls	Screened based on existing analyses. See Section 5.9.2.
Control room ceilings	Screened. The control room ceiling was upgraded in Fall 1990. See Section 5.4.
Impact between structures	Screened based on EPRI NP-6041, Table 2-3
Category III structures with safety-related equipment or with potential to fail Category I structures	Screened based on EPRI NP-6041, Table 2-3. See Section 2.6.5.
Dams, levees, dikes	Capacity exceeds the RLE. See Section 5.6.
Soil failure modes	HCLPF > 0.3g. See Section 5.6

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Table 5-3

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
NSSS System	Westinghouse primary coolant system piping and vessels	N/A	Screened based on EPRI NP-6041, Table 2-4. See Section 5.7.1.
NSSS Supports	Supports for NSSS coolant system piping and vessels	N/A	Screened based on EPRI NP-6041, Table 2-4. See Section 5.7.1
Reactor Internals	Westinghouse PWR reactor internals	N/A	The HCLPF capacity is probably larger than the RLE. See Section 5.7.2.
CRDM	Control rod drive housings and mechanisms	N/A	Screened based on EPRI NP-6041, Table 2-4 and Westinghouse DWG No. 618F121. See Section 5.7.3.
Category I Piping	Miscellaneous Category I piping.	N/A	No piping anomalies were identified. See Section 5.8.3.
N2 Header	The nitrogen header passes through non-seismic buildings.	The header has been removed from the A-46 SSEL.	The header has been removed from IPEEE success paths.
<u>Fluid Operated Valves:</u> TCV-1902A	Valve conduits and air line have marginal supports.	Additional supports will be installed for the conduits and air line. Reference NED-C-0163.	Screened based on resolution of the A-46 outlier issue.
PCV-4	The valve is mounted on tubing less than 1" in diameter. The tubing is well supported adjacent to the valve.	The issue is resolved based on test report No. EGS-TR-093400	Screened based on resolution of the A-46 outlier issue.
PCV-456	The mounting bracket for solenoid valves 1 and 3 is not secured.	The bracket will be mounted to the valve yoke. Reference NED-C-0163.	Screened based on resolution of the A-46 outlier.

EQE

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
Motor Operated Valves: FCV-6416	The Flow Control Valve actuator yoke is braced to the floor in the vertical and east-west directions. The associated piping system is supported from the floor above. The valve is located in the relatively flexible steel frame turbine building.	Loads on the valve yoke are acceptable based on finite element analyses (EQE Calculation 52212-C-053).	The A-46 analyses was run at the RLE resulting in a HCLPF capacity greater than 0.3g. See Section 6.1.
V6-16A V6-16B V6-16C	Valves 16-16A, B and C were identified as outliers because the valve bodies are Cast Iron.	Seismic stresses in the valve body due to piping loads are within 20% of specified minimum ultimate tensile strength (EQE Calculation 52212-C-051).	Screened based on results of the A-46 outlier evaluation (EQE Calculation 52212-C-051).
FCV-626	Valves operator height and weight exceed GIP screening guidelines.	Resolved by applying a 3g static load check (EQE Calculation 52212-C-052)	HCLPF capacity is greater than 0.3g (EQE Calculation 52212-C-052). See Section 6.1.
RC-535 RC-536	Valves operator height and weight exceed screening guidelines.	Resolved by applying a 3g static load check (EQE Calculation 52212-C-052)	HCLPF capacity is greater than 0.3g (EQE Calculation 52212-C-052). See Section 6.1.
RHR-744A RHR-744B	Valves operator weight exceeds GIP screening guidelines.	Resolved by applying a 3g static load check (EQE Calculation 52212-C-052)	HCLPF capacity is greater than 0.3g (EQE Calculation 52212-C-052). See Section 6.1.
RHR-759A RHR-759B	Valves operator height and weight exceed screening guidelines.	Resolved by applying a 3g static load check (EQE Calculation 52212-C-052).	HCLPF capacity is greater than 0.3g (EQE Calculation 52212-C-052). See Section 6.1.
CC-735	The valve operator is adjacent to a tube steel column. Piping analyses indicates that impact is credible. Valves operator height and weight exceed screening guidelines.	The potential interaction will be resolved by modification. Reference NED-C-0163. Resolved by applying a 3g static load check (EQE Calculation 52212-C-052).	The modification should consider RLE level displacements. HCLPF capacity is greater than 0.3g (EQE Calculation 52212-C-052). See Section 6.1.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
RHR-750 RHR-751	The valves have ductile iron yokes. The ASTM 536 Gr. 65-45-12 material is comparable in ultimate and yield strength to typical cast steel yokes, however, with 12% elongation, the material does not meet the caveat "no cast iron yoke".	Resolved by applying a 3g static load check (EQE Calculation 52212-C-052).	HCLPF Capacity is 0.28g (EQE Calculation 52212-C-052). See Section 6.1.
<u>Solenoid Operated Valves</u> EV-1711	The valve is mounted to a rack. A mounting bolt is missing.	The missing bolt will be installed. Reference NED-C-0162	Screened based on resolution of SRT walkdown issues.
Passive Valves	Check valves, manual valves, etc.	N/A	Screened based on EPRI NP-6041, Table 2-4.
<u>Heat Exchangers</u>: CVC Reg. HX	Attachment of the upper and lower shells to the rack is flexible to accommodate thermal expansion.	The configuration is adequate for design basis loads (EQE Calculation 52212-C-047.)	The HCLPF capacity is greater than 0.3g. (EQE Calculation 52212-C-047.) See Section 6.2.
CCW Heat Exchangers A & B	GIP analyses shows insufficient shear capacity in heat exchanger anchor bolts.	Anchorage is adequate considering friction to resist anchor bolt shear loads (EQE Calculation 52212-C-O-048)	The HCLPF capacity is greater than 0.3g. (EQE Calculation 52212-C-O-048) See Section 6.2.
RHR Heat Exchangers A & B	RHR heat exchanger anchor bolts have less that 4 bolt diameters edge distance and spacing reductions.	RHR heat exchanger anchorage is adequate (EQE Calculation 52212-C-049).	The HCLPF capacity is greater than 0.3g (EQE Calculation 52212-C-049). See Section 6.2.
	Existing analysis demonstrates adequate capacity for SSE levels. The analysis will be scaled to the RLE.	N/A	The HCLPF capacity is greater than 0.3g (CP&L Calculation RNP-C/EQ-1349). See Section 6.2.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
Atmospheric Storage Tanks: Condensate Storage Tank	The CST was overpressurized several years ago and it is founded on a ring-type foundation.	The Condensate Storage Tank anchorage is adequate. (EQE Calculation 52212-C-066).	The HCLPF capacity is greater than 0.3g. (EQE Calculation 52212-C-066). See Section 6.3.3.
Diesel Oil Storage Tank	The DOST is founded on a ring foundation.	The Diesel Fuel Oil Tank is adequate (EQE Calculation 52212-C-064).	The HCLPF capacity is greater than 0.3g (EQE Calculation 52212-C-064) See Section 6.3.
Refueling Water Storage Tank	The RWST is founded on a ring foundation.	The Refueling Water Storage Tank anchorage is adequate. (EQE Calculation 52212-C-065).	The HCLPF capacity is greater than 0.3g. (EQE Calculation 52212-C-065). See Section 6.3.2.
IC Turbine FOST	The fuel oil storage tank associated with the Unit 1 internal combustion turbine has suspect capacity.	The tank has been removed from the A-46 SSEL based on the day tank and DOST fuel oil inventory.	The tank has been removed from the IPEEE success paths based on the day tank and DOST fuel oil inventory.
Other Vessels: Raised Vertical Tanks	Steam dump N2 ACC, VCT, EDG-A & B ART, BAST A & B.	N/A	The capacities are greater than 0.3g with the critical case being the boric acid storage tank with a HCLPF = 0.32g (EQE Calculation 52212-C-054). See Section 6.4.
EDG A & B Air Dryers	The dryer cylinders are not positively anchored. Limited support is provided by threaded cooling water lines. Failure of the lines could result in water spray, flooding and/or loss of adequate cooling water flow to the diesels.	The lines will be modified to provide positive support. Reference NED-C-0163.	Screened based on resolution of SRT walkdown issues.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
N2 ACC A N2 ACC B	Tank anchorage requires evaluation.	Anchorage is adequate for design basis loads (CP&L Calculation RNP-C/SPPT-2009).	Screened based on adequate margin in the A-46 evaluation.
CCW Surge Tank	Tank anchor bolts do not meet minimum embedment requirements for J-bolts.	Anchor capacities were refined based on concrete bond strength and a reduced shear cone, resolving the outlier (EQE Calculation 52212-C-058).	Extension of the A-46 analyses resulted in a HCLPF capacity greater than 0.3g (EQE Calculation 52212-C-058). See Section 6.4.
Buried Tanks	None identified	N/A	N/A
<u>Battery Racks A & B</u> Batteries	Batteries are greater than 10 years old.	The batteries are being replaced during RFO-16 (Reference Work Requests A-94AKHD1 and B-94AKHF1).	Resolved based on battery replacement.
	An overhead room cooler is mounted on vibration isolation pads. The cooler is restrained from falling, however, displacement may result in failure of attached lines.	Further investigation revealed that that failure of the lines would result in discharge of only a small inventory of refrigerant.	Screened based on resolution of the A-46 II/I issue.
Battery racks	Racks are constructed of angle base details, tubular vertical members and strap cross braces.	N/A	The HCLPF capacity is 0.51g (EQE Calculation 52212-C-045). See Section 6.5.
<u>Diesel Generators</u> EDG-A EDG-B	Emergency Diesels Generators A & B	N/A	Screened based on high capacity.
	An unanchored steel platform is located close to the control panel.	A work ticket will be issued to eliminate the interaction potential. Reference NED-C-0162.	Screened based on resolution of SRT walkdown issues.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
	HVAC ducts pose a potential interaction issue with the DG buss ducts	The HVAC ducts were evaluated, resolving the interaction (Reference CP&L Calculation RNP-C/SPPT-2011). See Section 5.8.2.	Screened based on satisfactory resolution of the A-46 II/I issue.
Horizontal Pumps CCW Pumps A, B & C	The component cooling water pumps are anchored with grouted-in-anchors.	N/A	Screened based on high margin demonstrated in the A-46 review. (EQE Calculation 52212-C-031).
AFW MDP-A AFW MDP-B	The motor driven auxiliary feedwater pumps are anchored with grouted-in-anchors.	N/A	Screened based on high margin demonstrated in the A-46 review (EQE Calculation 52212-C-030).
CP-B and CP-C	Charging pump attached conduit and tubing are not well supported.	The conduit and tubing will be modified to provide adequate support. Reference NED-C-0163.	Screened based on resolution of SRT walkdown issues.
SI Pump A SI Pump B	The safety injection pumps are anchored with grouted-in-anchors.	N/A	Screened based on high margin demonstrated in the A-46 review.
Fuel oil transfer pumps A and B	Pump suction piping is threaded, reduces in size, and is rigidly attached to the adjacent Diesel Fuel Oil Storage Tank.	Verify the tank will not displace under SSE loading. The tank displacement was verified to be minimal.	The tank HCLPF capacity exceeds 0.3g. The line is judged to have adequate flexibility to accommodate minor tank displacement. See Section 6.3.
	The pumps are anchored with expansion anchors.	The installation is considered adequate based on small pumps with robust anchorage (EQE Calculation 52212-C-055).	Screened based on satisfactory resolution of the A-46 outlier issue.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
BAT Pump A BAT Pump B	The pumps are anchored with expansion anchors.	The installation is considered adequate based on small pumps with robust anchorage (EQE Calculation 52212-C-055).	Screened based on satisfactory resolution of the A-46 outlier issue.
Vertical Pumps: RHR Pumps A & B	The pumps are free to displace in the pipe axial direction and are not positively anchored in the vertical direction.	Existing piping analyses were reviewed to verify that pump inertial loads were addressed (EQE Calculation 52212-C-057).	Existing design basis piping analyses were reviewed considering RLE spectra, resulting in a HCLPF capacity is greater than 0.3g (EQE Calculation 52212-C-057). See Section 6.6.
Service water pumps A, B, C and D	The pumps, anchored with four 3/4" diameter bolts, have relatively low anchorage capacity. The pumps are housed in a security structure that is not seismic Category I. Potential interaction issue.	N/A The security structure was judged adequate to maintain integrity following an earthquake based of field walkdowns and a review of security drawings.	The HCLPF Capacity exceeds 0.3g (EQE Calculation 52212-C-056). See Section 6.6. Screened based on satisfactory resolution of the A-46 issue.
Fans HVS-5 HVS-6	Anchorage includes grouted-in bolts with less than minimum embedment and the detail may be susceptible to prying.	The configuration was evaluated as acceptable accounting for anchor prying and reduced allowables (EQE Calculation 52212-C-033).	Screened based on adequate margin in the A-46 evaluation.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
Air Handlers HVH-6A and B	Access panels are not positively restrained.	Verify units are not required to function (required for CCW pressure boundary only). Perform a II/I review for falling access panels. VERIFIED.	Screened based on EPRI NP-6041, Table 2-4 and SRT walkdowns.
HVH-7A and B	Access panels are not positively restrained. The units are not laterally restrained, displacement may result in attached piping failure.	Verify units are not required to function (required for CCW pressure boundary only). Perform a II/I review for falling access panels. VERIFIED. A modification will be issued to restrain the units to prevent displacement. Reference NED-C-0163).	Screened based on EPRI NP-6041, Table 2-4 and SRT walkdowns. Screened based on modification to add lateral restraint.
HVH-8A and B	Access panels are not positively restrained. The units are not laterally restrained, displacement may result in attached piping failure.	Verify units are not required to function (required for CCW pressure boundary only). Perform a II/I review for falling access panels. VERIFIED. The units have been anchored by field revision to Mod M-1144.	Screened based on EPRI NP-6041, Table 2-4 and SRT walkdowns. Screened based on resolution of SRT walkdown issues.
Chillers	Chillers WCCU-1A and 1B were reviewed by the SRT.	N/A	Screened based on EPRI NP-6041, Table 2-4 and SRT walkdowns.
Air Compressor	None	N/A	N/A

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
HVAC Ducting and Dampers	Miscellaneous ducts and dampers.	N/A	Screened based on EPRI NP-6041, Table 2-4; SRT walkdowns; and CP&L Calculation RNP-C/SPPT-2011. See Section 5.8.2.
Cable Trays	Miscellaneous cable trays.	N/A	Screened based on EPRI NP-6041, Table 2-4 (See Section 5.8.1) and SRT walkdowns.
Electrical Conduit	Miscellaneous conduit.	N/A	Screened based on EPRI NP-6041, Table 2-4 (See Section 5.8.1) and SRT walkdowns.
<u>Motor Control Centers:</u> MCC-5 MCC-6	MCC anchorage base details and overturning capacity are suspect.	The MCC anchorage will be upgraded. Reference NED-C-0163.	The upgrade should consider RLE loads.
AC MCCs	Various AC MCCs	N/A	The HCLPF capacity for a bounding AC MCC anchorage configuration is 0.85g (EQE Calculation 52212-C-050). MCC function is screened based on the SQUG Reference Spectrum. (See Section 6.7).
125 DC MCC-A	Anchorage has low margin beyond the DBE due to a base detail susceptible to prying.	Anchorage is adequate for A-46 (CP&L Calculation ID RNP-C/EQ-1316).	The anchorage HCLPF capacity is 0.3g (EQE Calculation 52212-C-043). MCC function is screened based on the SQUG Reference Spectrum. (See Section 6.7).
125V DC MCC-B	Overhead cable tray routed from MCC-B to station batteries B is not well supported above the MCC which represents an iteraction issue.	Modify the current cable configuration above the MCC to resolve the interaction issue.	Screened based on resolution of SRT walkdown issues.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
Low Voltage Switchgear EMER-BUS-E1 & E2	The internal load path of the buss through the frame to the based channels requires further evaluation.	The configuration was evaluated and found to be adequate (EQE Calculation 52212-C-059).	The HCLPF capacity exceeds 0.3g (EQE Calculation 52212-C-059). See Section 6.8.
	The breaker trolley on rails on top of the buss may rattle during an earthquake, resulting in vibrations that may effect relays.	The issue will be resolved via modification. Reference NED-C-0163.	Screened based on resolution of SRT walkdown issues.
	The buss contains DB-50 and DB-75 breakers. These breakers may displace and lose secondary contacts if not adequately restrained.	The issue will be resolved via modification. Reference NED-C-0163.	Screened based on modification to add lateral restraint.
RTB Cabinet	The Reactor Trip Switchgear cabinet is anchored to the structure with welds to thin sheet metal and base anchors have large gaps.	The cabinet will be modified to provide adequate anchorage (Reference NED-C-0163) or the item will be accepted by system evaluation.	Screened based on modification to install conservative anchorage.
	DB-50, DB-75, and DB-100 breakers were not restrained.		Add lateral restraint.
Medium Voltage Switchgear	None, RNP Category I systems operate at 480 V	N/A	N/A
Transformers CVT7.5/INST1	Conduit to the small transformer is not well supported.	A modification will be issued to support the conduit. Reference NED-C-0163.	Screened based on additional of conduit supports. Only small rugged transformers were observed since success path systems operate at 480 V.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
Motor Generators	None	N/A	N/A
Distribution Panels	Miscellaneous distribution panels	N/A	Screened based on EPRI NP-6041, Table 2-4, SRT walkdowns, and A-46 anchorage evaluation (EQE Calculation 52212-C-028).
I & C Panels: NIS Cabinets, RMS Console and RTGB	The control room cabinets are anchored to embedded angle by a combination of limited welds and friction-type support details.	The cabinets were evaluated as a single unit to demonstrate adequate anchorage (EQE Calculation 55212-C-060).	The cabinets have a HCLPF capacity greater than 0.3g pending the addition of inter-cabinet connection (EQE Calculation 52212-C-060).
	Adjacent panels are not positively attached together.	Adjacent panels will adequately attached together via modification.	Screened based on adequate resolution of the A-46 issue.
EDG CON SWTCHBRDS A & B	The EDG Control Switchboards are within 1/4" of adjacent walls, interaction potential due to essential relays within the cabinets. The EDG A switchboard door is missing a screw.	Impact is not credible at DBE levels (EQE Calculation 52212-C-046). The screw was replaced via Work Ticket WR 94AQMV1.	Impact is not credible at the RLE (EQE Calculation 52212-C-046). (See Section 6.9). The screw was replaced via Work Ticket WR 94AQMV1.
Auxiliary Relay Racks A - F and G-M	The racks are a unique configuration and not represented in the SQUG experience database inventory. Cable trays share common supports with the racks.	The rack will be analyses and necessary modifications implemented to resolve the issues. Reference NED-C-0163. See above.	Rack evaluation and should use conservative criteria and factors of safety to ensure that the resulting capacity exceeds the RLE. See above.

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Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
	Cable trays are in close proximity to the racks. Impact may result in vibration and relay chatter issues.	See above.	See above.
	Adjacent panels and appurtenances may impact the racks.	See above.	See above.
ERFIS MUX-1 & 2	The metering cabinet, auxiliary relay racks and MUX 1 & 2 are not bolted together. Impact may result in relay chatter.	Positive attachment between adjacent cabinets will be added during RFO-16. (Reference Mod 1144)	Screened based on positive cabinet attachment.
	Slide-in PC boards are not restrained.	The boards were tug-tested by the SRT to confirm adequate mounting.	Screened based on resolution of SRT walkdown issues.
ERFIS MUX 3	Slide-in PC boards are not restrained.	The boards were tug-tested by the SRT to confirm adequate mounting.	Screened based on resolution of SRT walkdown issues.
FDAP-A2	The fire detection actuation panel is in contact with MCC-2. Impact between the panels may result in relay chatter.	Relay chatter is acceptable based on systems consequence (CP&L Memo NED-C-151).	Screened pending resolution of the chatter issue.
Safeguards Racks: Rack 50 Rack 51 - 52 Rack 53 - 57 Rack 58 - 62 Rack 63 -64	The Safeguards Racks have questionable anchorage. The SRT identified anchorage capacity and panel frequency as significant issues. Base grout pads are deteriorating.	Modification was proposed to resolve anchorage issues. Reference NED-C-0163. See above.	Screened based on conservative anchorage upgrade. See above.
Instrument Cabinet A	Essential relays are located in a front-mounted cantilevered rack. The Rack lacks rear support.	The cabinet was removed from the SSEL since the relays are for annunciation only.	The cabinet was removed from the success paths since the relays are for annunciation only.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
Diesel Control Panels A & B	The diesel control panels are mounted to the EDG skid with spring vibration isolation mounts. The panels contain essential relays.	Modification was proposed to resolve the issue. Reference NED-C-0163. The SRT recommends replacement of the existing supports with a new floor or ceiling mounted support such that relocation of the panel should not be required.	Screened based on modification to eliminate vibration isolators.
EDG-A-480V-PNL EDG-B-480V-PNL	Conduit exiting the panels is in contact with service water piping.	Based on further review, Interaction is not credible since the panel and piping are rigidly attached to a common skid.	Screened based on resolution of SRT walkdown issues.
CET Panel A CET Panel B	The panel is located within about 1/4" to an adjacent cabinet.	Verify no sensitive devices such as essential relays are located within the cabinets. VERIFIED.	Screened based on resolution of SRT walkdown issues.
<u>Instruments on Racks</u> FY-1425A, B & C	Enclosure mounting to Unistrut does not have a square plates to ensure proper bearing. Also, for FY-1425A, the lower left Unistrut bolt is not properly installed.	Work tickets were proposed to resolve the issues. Reference NED-C-0162. The configuration was judged adequate but recommended for rework to increase the seismic margin.	Screened.
FY-1426A, B & C	Enclosure mounting to Unistrut does not have a square plates to ensure proper bearing. Also, for FY-1426C, two Unistrut bolt is not properly installed.	Work tickets were proposed to resolve the issues. Reference NED-C-0162. The configuration is recommended for rework to increase the seismic margin.	Screened based on rework of square plates to Unistrut support.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
PIC-1393	Entry conduit lacks proper vertical support. Conduit is currently supported by wire from adjacent conduit.	The conduit will be upgraded via modification. Reference NED-C-0163.	Screened based on resolution of SRT walkdown issues.
PSL-1476-1	The enclosure has an unusual mounting detail, attached to a bolted flange connection.	The mounting configuration is satisfactory for A-46 (CP&L Calculation ID RNP-C/EQ-1323).	The HCLPF capacity is over 1g (EQE Calculation 52212-C-044). Relay capacity is addressed via system consequence review.
	The instrument is missing a mounting bolt.	The missing bolt will be installed. Reference NED-C-0162.	Screened based on resolution of SRT walkdown issues.
	Conduit that terminates at the enclosure is supported by baling wire.	New supports will be installed Reference NED-C-0163.	Screened based on resolution of SRT walkdown issues.
	Loose wires inside the enclosure should be secured.	The loose wires will be secured Reference NED-C-0162.	Screened based on resolution of SRT walkdown issues.
LT-1454A	Conduit is not attached to its intended support.	A Work Ticket will be issued to resolve the issue. Reference NED-C-0162.	Screened based on resolution of SRT walkdown issues.
PT-117	Conduit from the PT to CVC 256 has a loose clamp.	A Work Ticket will be issued to reinstall the conduit clamp. Reference NED-C-0162.	Screened based on resolution of SRT walkdown issues.
A1-E1/2	A sheet metal cover above the rack is not positively attached, resulting in an interaction concern.	A Work Ticket will be installed to remove the sheet metal cover. Reference NED-C-0162.	Screened based on resolution of SRT walkdown issues.
FT-122	A storage cabinet near the transmitter is not anchored. Overturning of the cabinet may impact the instrument.	The adjacent cabinet will be anchored. Reference NED-C-0163.	Screened pending resolution of SRT walkdown issues.
FT-154A	The instrument is in contact with a blind flange for CVC 305C.	Upon further SRT review, the issue is resolved since the instrument and flange will displace as a system.	Screened based on resolution of SRT walkdown issues.

Table 5-3 (Continued)

SUMMARY OF EQUIPMENT AND SUBSYSTEMS SEISMIC MARGIN EVALUATION

Equipment	Description	A-46 Outlier Issues	IPEEE Disposition
PC-611	A suspended chain operator for valve CC-712A is adjacent to the instrument. Potential interaction issue.	Upon further SRT review, the issue is resolved based on adequate clearance between the chain and instrument.	Upon further SRT review, the issue is resolved based on adequate clearance between the chain and instrument.
PIC-157	Tubing and piping associated with the instrument are adjacent to a block wall.	The block wall is adequate for SSE loads (CP&L Calculation RNP-C/RAB-1049).	The block wall is adequate for RLE loads (See Section 5.9.2).
TIC-107 TIC-109	The instrument support is a pipe that slips over a larger support pipe, secured with four set screws.	Upon further SRT review, the set screws are tight and a tug test was performed to ensure adequate attachment.	Screened based on resolution of SRT walkdown issues.

Table 5-4

MASONRY WALLS

NO.	LOCATION	ELEV.	WALL TK.	SYSTEM
1	FHB-SPENT FUEL PIT	226	1'-0"	SPENT FUEL PIT DEMINERALIZER SPENT FUEL PIT HEAT EXCHANGER SPENT FUEL PIT COOLING PIPING
2	RAB-PIPE ALLEY	226	3'-0"	SPENT FUEL PIT COOLING PIPING
3a	RAB-CP ROOM	226	1'-6"	SEAL WATER INJECTION FILTERS SEAL WATER INJECTION PIPING
3b	RAB-CP ROOM	226	1'-6"	SEAL WATER INJECTION FILTERS SEAL WATER INJECTION PIPING CHARGING PUMP "C" SUCTION PE
4	RAB-PIPE ALLEY	226	2'0"	MDAFW DISCHARGE PIPING WASTE DISPOSAL PIPING SG BLOWDOWN PIPING AUX. STEAM PIPING RAD. HVAC DUCT
5a	RAB	226	1'-0"	NON-SAFETY RELATED EQUIP.
5b	RAB	226	1'-0"	NON-SAFETY RELATED EQUIP.
5c	RAB	226	1'-0"	NON-SAFETY RELATED EQUIP.
6	RCB	228	1'-6"	REGENERATIVE HEAT EXCHANGER HEAT EXCHANGER INLET PIPING HEAT EXCHANGER OUTLET PIPING
7a	FHB	226	2'-0"	NON-SAFETY RELATED EQUIPMENT
7b	FHB	226	2'-0"	NON-SAFETY RELATED EQUIPMENT
7c	FHB	226	2'-0"	NON-SAFETY RELATED EQUIPMENT

6. ASSESSMENT OF ELEMENTS NOT SCREENED OUT

Thirty-one items were selected for HCLPF evaluation by the SRT. The items were grouped into HCLPF calculations based on similar characteristics. Results of the HCLPF evaluations are summarized below.

6.1 MOTOR OPERATED VALVES

Motor operated valves (MOV) were selected for HCLPF evaluation due to a valve yoke support configuration susceptible to differential displacement, ductile iron valve yokes, and operator height and weight exceedances. Motor operated valve HCLPF evaluations are discussed in the following sections.

6.1.1 MOV Yoke Support

Control valve FCV-6416 was selected for HCLPF evaluation since the actuator yoke is braced to the floor in the vertical and east-west directions and the associated piping system is supported from the floor above. The valve is located in the relatively flexible turbine building. The SRT identified the potential for excessive differential floor displacement loads on the valve yoke.

The configuration was modeled using a finite element code to determine loads on the valve yoke. The analysis resulted in a HCLPF capacity greater than the 0.3g RLE.

6.1.2 Ductile Iron MOV Yokes

Valves RHR-750 and 751 were selected for HCLPF evaluation since the valve yoke material is ASTM-A536, Gr 65-45-12 Nodular (ductile) cast iron. The ductile iron material is comparable in ultimate and yield strength to typical cast steel yokes, however, with 12% elongation, the material does not meet the caveat "no cast iron yoke". Additionally, the valves are located high in the Structure where spectral accelerations exceed the SQUG Reference Spectrum. The valves were evaluated based on a 3g static load check on the yoke's weakest direction, as recommended in the GIP (Reference 8). Yoke stresses were within allowables considering the 3g static and maximum normal operating thrust loads on the yoke.

The HCLPF capacity was calculated since spectral accelerations at the valve elevation exceed the SQUG Reference Spectrum. The analyses calculated a HCLPF capacity of 0.28g. This results in a capacity less than the RLE. 20% of ultimate may be overly conservative for ASTM-A536, Gr 65-45-12 Nodular (ductile) cast iron since the material is more ductile than lower grade cast iron. Therefore, the 0.28g HCLPF capacity is considered a lower bound estimate.

6.1.3 MOV Height and Weight Exceedances

Operator height and weight exceedances were identified for motor operated valves FCV-626, CC-735, RC-535, RC-536, RHR-744A, RHR-744B, RHR-759A, and RHR-759B. Additionally, valves RC-535 and RC-536 are located high in the Structure where spectral accelerations exceed the SQUG Reference Spectrum. The valves were evaluated based on a 3g static load check on the yoke's weakest direction, as recommended in the GIP (Reference 8). Yoke stresses were within allowables for all of the valves considered the 3g static and maximum normal operating thrust loads on the yoke.

The HCLPF capacity was calculated for the valves RC-535 and RC-536 since spectral accelerations at the valve elevation exceed the SQUG Reference Spectrum. The analyses calculated a HCLPF capacity of 0.42g. This results in a capacity greater than the RLE.

6.2 HEAT EXCHANGERS

The Robinson CVC Regenerative, CCW, RHR, and seal water heat exchanger were selected for HCLPF evaluation due to support and anchorage details that were not screened by the SRT. Heat exchanger HCLPF evaluations are discussed in the following sections.

6.2.1 CVC Regenerative Heat Exchanger

The CVC Regenerative Heat Exchanger was selected for HCLPF evaluation due to load path issues associated with the upper and lower shell support configurations. The supports are flexible to accommodate thermal expansion of the 3-shell heat exchanger.

The analysis resulted in a capacity greater than 0.3g.

6.2.2 CCW Heat Exchangers A & B

A-46 analyses of the CCW heat exchangers result in insufficient anchor bolt capacity. The outlier configuration was resolved by considering friction to resist shear loads.

HCLPF analysis considered friction to resist shear loads, resulting in a capacity greater than 0.3g.

6.2.3 RHR Heat Exchangers A & B

RHR Heat Exchangers A & B were selected for HCLPF evaluation since the design basis evaluation did not address anchor bolt capacity reduction due to small edge distance and bolt spacing. Several of the bolts have less than 4 bolt diameters of edge distance.

The analyses considered reduced anchor capacity for each direction of seismic input, resulting in a HCLPF capacity greater than 0.3g. This results in a capacity greater than the RLE.

6.2.4 Seal Water Heat Exchanger

The seal water heat exchanger was selected for HCLPF capacity evaluation due to the unique configuration. The small vertical heat exchanger was reviewed by scaling existing analysis to the RLE.

The HCLPF capacity is greater than 0.3g with the upper lateral restraints. The SRT verified the existence of lateral restraints.

6.3 ATMOSPHERIC STORAGE TANKS

The Condensate Storage Tank (CST), Refueling Water (RWST), and Diesel Fuel Oil Tanks (DOST) were selected for HCLPF evaluation. Atmospheric storage tank HCLPF evaluations are discussed in the following sections.

6.3.1 Diesel Fuel Oil Storage Tank

The Diesel Fuel Oil Storage Tank was selected for HCLPF evaluation since the anchorage could not be screened out and the tank in supported by a ring foundation. Additionally, the fuel outlet piping (2" diameter) runs horizontal for about 2', then drops vertical about 1', then runs horizontal for about 2' into a 2" x 1" reducer at the threaded pump suction nozzle.

The tank was evaluated in accordance with Appendix H of Reference 3. The minimum HCLPF capacity was calculated to be 0.32g. The fuel outlet line is judged to have adequate flexibility considering a tank HCLPF capacity of over 1g for sliding. This results in a capacity greater than the RLE.

6.3.2 Refueling Water Storage Tank

The Refueling Water Storage Tank was selected for HCLPF evaluation since the anchorage could not be screened out and the tank in supported by a ring foundation.

The tank was evaluated in accordance with Appendix H of Reference 3. The minimum HCLPF capacity was calculated to be greater 0.3g. This results in a capacity greater than the RLE.

6.3.3 Condensate Storage Tank

The Condensate Storage Tank was selected for HCLPF evaluation since the anchorage could not be screened out and the tank in supported by a ring foundation. Additionally, the tank was overpressurized several years ago resulting in stretching of the anchor bolts and shell deformation.

The tank was evaluated in accordance with Appendix H of Reference 3, considering previous strain on the anchor bolts. The minimum HCLPF capacity was calculated to be greater than 0.3g. This results in a capacity greater than the RLE.

6.4 OTHER VESSELS

Other vessels selected for HCLPF evaluation are the lowest capacity raised vertical tank and the CCW Surge Tank. HCLPF evaluations for other vessels are discussed in the following sections.

6.4.1 Raised Vertical Tanks

Miscellaneous success path raised vertical tanks were evaluated to determine the HCLPF capacity for the bounding configuration Diesel generator air receiver tanks, boric acid storage tanks (BAST) A & B, the nitrogen steam dump accumulator, and volume control tanks were reviewed.

The most critical configuration is the BAST with a HCLPF capacity of 0.32g. This results in a capacity greater than the RLE.

6.4.2 CCW Surge Tank

The CCW surge tank was selected for HCLPF Capacity evaluation because the cast-in-place J-bolts do not meet minimum embedment requirements specified in the GIP. GIP screening criteria assign no capacity for 1" diameter 90° anchor bolts with about 11" of embedment. The analysis accounted for the bond strength between the anchor bolt and the concrete to establish reasonable anchorage capacity.

The analyses calculated in a HCLPF capacity greater than 0.3g. This results in a capacity greater than the RLE.

6.5 STATION BATTERY RACKS

Station Battery Rack B was selected for HCLPF evaluation since the rack is not obviously robust and batteries are critical success path elements. Rack B was selected since the A-46 evaluation found B to be more critical than Battery Rack A.

The analyses calculated in a HCLPF capacity of 0.51g. This results in a capacity greater than the RLE.

6.6 VERTICAL PUMPS

Two sets of vertical pumps were selected for HCLPF capacity evaluation: the RHR Pumps and the Service Water Pumps.

6.6.1 RHR Pumps

The RHR pumps were selected for HCLPF evaluation since they are not positively anchored to their foundation. Instead, they are guided in-line pumps.

Existing piping analyses were reviewed to confirm that pump inertia loads were addressed in the design basis analyses. The analyses were then scaled to the RLE, resulting in a HCLPF capacity greater than the RLE.

6.6.2 Service Water Pumps

Service Water Pumps A, B, C, and D were selected for HCLPF evaluation due to a lack of robust anchorage and substantially higher SMA seismic response compared to design basis response in the Service Water Structure.

The HCLPF capacity was calculated considering both the pump anchorage and well supported discharge piping. The HCLPF capacity is greater than the RLE.

6.7 MOTOR CONTROL CENTERS

Two groups of motor control centers were selected for HCLPF evaluation to capture both the AC and DC configurations.

6.7.1 AC Motor Control Centers

A bounding AC motor control center was selected for HCLPF evaluation. The selection considered anchorage capacity/demand based on A-46 anchorage evaluations, elevation within the structure, and other considerations based on SRT observations. The HCLPF was calculated based on anchorage capacity. MCC function was screened based on enveloping of SMA in-structure spectra by the SQUG Reference Spectrum at mounting locations.

The analyses calculated in an anchorage HCLPF capacity of 0.85g. This results in a capacity much greater than the RLE.

6.7.2 125V DC-MCC-A & B

125V DC-MCC-A & B were selected for HCLPF evaluation since the anchorage configuration is susceptible to significant prying. One end of the 2-bay MCC is anchored via a bracket fabricated from plate that can introduce prying at a factor of about 3 due to overturning loads in the long direction. Any prying loads due to front-to-back loads would be alleviated by slight slippage of the expansion anchors. Such slippage is judged acceptable since the cabinets do not contain essential relays.

The analyses calculated in a HCLPF capacity of 0.3g. This results in a capacity equal to the RLE.

6.8 LOW VOLTAGE SWITCHGEAR

Low voltage switchgear Busses E-1 and E-2 were selected for evaluation to assess the anchorage configuration. The SRT identified the internal load path of the buss through the base channels to embedded steel as the limiting attribute. The evaluation considered existing Westinghouse qualification documentation.

The analysis resulted in a HCLPF capacity greater than the 0.3g RLE.

6.9 INSTRUMENTATION & CONTROL PANELS

Various control room cabinets and the Emergency Diesel Generator Control Switchboards were selected for HCLPF evaluation.

6.9.1 Control Room Cabinets

Control room cabinets including the RTGB, RMS console and nuclear instrumentation system panels were evaluated for HCLPF capacity since the anchorage configurations are a combination of welds and friction clamps that are not regularly spaced. The cabinets were evaluated as a single unit, such that panels that are conservatively anchored will capture inertial loads from adjacent sections. The non-safety incore instrumentation and APDMS panels were also included since they are part of the main control room cabinets which were evaluated as a single unit.

The evaluation resulted in a HCLPF capacity greater than the 0.3g RLE.

6.9.2 Emergency Diesel Generator Control Switchboards

Emergency Diesel Generator Control Switchboards A & B were selected for HCLPF evaluation due to a potential impact issue with adjacent walls. A protruding bolt from the switchboard is within about 1/4" from the south wall and 1/2" from the east wall. The evaluation calculated panel spectral displacement at the frequency of interest and concluded that impact will not occur.

The analyses concluded that impact will occur at 0.6g. This results in a HCLPF capacity greater than the RLE.

6.10 INSTRUMENTS ON RACKS

Pressure Switches PSL-1476-1 & 2 were selected for HCLPF evaluation due to an unusual mounting configuration. The switches are mounted to a pipe flange by an angle bracket. The instruments do not contain essential relays.

The analyses calculated in a HCLPF capacity of greater than 1g. This results in a capacity much greater than the RLE.

7. RELAY EVALUATION

Robinson is identified as a full-scope plant for the .3g earthquake by NUREG-1407. NUREG-1407 (Reference 2) requests that full-scope plants such as Robinson which are also included as a USI A-46 Plant should follow the USI A-46 procedures for the relay review. NUREG-1407 also states that the plant systems should be reviewed within the scope of the IPEEE, including those that were within the scope of USI A-46, using appropriate margins from EPRI NP-6041 or USI A-46 procedures for the RLE. The USI A-46 criteria for relay functionality review are contained in Generic Letter GL 87-02 (Reference 40), which endorses the review procedure established in the GIP.

The GIP states that the purpose of the relay functionality review is to determine if the plant safe shutdown systems could be adversely affected by relay malfunction in the event of an SSE and to evaluate the seismic adequacy of those relays for which malfunction is unacceptable.

The GIP methodology for evaluation of the seismic functionality of relays is based on a two part screening process. The first part identifies a minimum set of relays whose function is essential for safe shutdown. The second part of the relay evaluation process uses relay Generic Equipment Ruggedness Spectra (GERS) (Reference 41) and other test data to assess the seismic adequacy of the essential relay types.

The identification of a minimum set of relays whose function is essential to the safe shutdown of the plant was prepared by engineers in the CP&L Probabilistic Risk Assessment (PRA) Group and the Robinson Engineering Support Section (RESS) Instrumentation and Control (I&C) Group. There were seven hundred and eighty-nine (789) relays and switches that were identified whose function was required for safe shutdown (Reference 78).

These seven hundred and eighty-nine (789) relays and switches were organized into an H.B. Robinson Unit 2 Essential Relay List. The relays on this list were investigated at the same time that the safe shutdown equipment was evaluated. This investigation and evaluation were performed by CP&L and EQE engineers who had successfully completed the SQUG Walkdown Screening and Seismic Evaluation Training Course.

The Seismic Capability Engineers (SCEs) verified that the manufacturer, the make, and the model were accurate according to the information provided on the essential relay list for a representative majority of the relays. The SCEs also observed and evaluated the mounting of the relays on or within electrical panels. It should be noted that the seismic adequacy of the panel structures and the anchorage was addressed by the separate evaluation of the panel as an SSEL equipment component.

After the completion of the walkdown and physical determination phase of the relay evaluation, the seismic adequacy of the essential relays was then assessed by EQE using GERS and other test data. The A-46/IPEEE Relay Report (Reference 42) provides a summary of the results of the relay seismic capacity vs. seismic demand study.

There were seven hundred and eighty-nine relays and switches that were identified for functionality review.

Seven hundred and forty-five relays (745) passed the USI A-46 capacity screening levels 0, 1, and 2. Level 0 screening is associated with switchgear only. Level 1 is associated with high capacity relays, the use of response spectra comparison, the location of relays within the plant, and the identification of no known low-ruggedness relays. Level 2 capacity screening is based on the use of in cabinet amplification factors, appropriate factors of safety, and the use of GERS or relay-specific seismic test data. Forty-four (44) relays did not meet the screening criteria and were submitted for further evaluation in the form of relay system consequence reviews.

Seven hundred and seven (707) relays passed the IPEEE capacity screening levels 0, 2, and 3. Levels 0 and 2 are identical to the A-46 screens described above. Level 1 is not applicable to IPEEE. Level 3 capacity screening is based on EPRI computer program GENRS (Reference 28). Eighty-two (82) relays and switches did not meet the IPEEE screening criteria and were submitted at the same time as the USI A-46 relays for further evaluation in the form of relay system consequence review.

It should be noted that four (4) relays were originally submitted for relay consequence review but have not been included in the above USI A-46 or the IPEEE totals where further review was required. However, these four relays have been included in the summary information below that identifies relays that were evaluated for consequence review. These four relays are associated with the Emergency Diesel Generator A and B Control Switchboards. After the consequence reviews were performed, CP&L identified these relays as controlling switchgear only. This change permitted the relays to be screened using the GIP and NP-6041 guidance negating the need for the system consequence reviews.

The relay system consequence reviews were performed by two separate evaluations. One consequence review was performed by Ricky Summitt Consulting. The other consequence review was performed in-house by a CP&L I&C engineer.

Ricky Summitt Consulting evaluated seventy-five (75) relays for relay consequence review. The report concluded that all seventy-five relays can be successfully screened. Eleven (11) of the relays could be screened out because they were not associated with equipment on the SSEL. Twenty-six (26) of the relays can be screened out because they provide annunciation input only. Twenty-four (24) of the relays were screened as being acceptable for relay chatter. This relay chatter would not result in the malfunction of the associated SSEL equipment or its ability to perform its required function. Fourteen (14) relays were screened based on operator action. Plant procedures are already in place which stipulate operator action for this relays for certain scenarios. A memorandum (NED-C-0189) was addressed to the plant operations department documenting these relays and the results of the consequence review.

Eleven (11) relays were evaluated for consequence review internally by CP&L engineering. All eleven (11) relays were determined to be acceptable. The review concluded that any chatter from seismic activity by these specific relays would only be momentary and that the overall function of the equipment associated with these relays would not be jeopardized enough to prevent continued operation.

In summary, all seven hundred and eighty-nine (789) relays on the Robinson Essential Relay List have been accepted by either capacity screening or system consequence screening. There were no low ruggedness relays encountered during the relay evaluation. It should be noted that the relay evaluation was based on adequate and direct mounting of the relays to the electrical panel. Any missing mounting hardware or loose relay connections were addressed on the panel SEWS forms and work tickets and/or maintenance requests have been identified or already issued to correct these concerns. It should also be noted that the adequacy of the panel structure and anchorage of the panels or cabinets is addressed by the separate evaluation of the panels as an SSEL equipment component. Relay panels and cabinets that were not anchored properly or had other unacceptable criteria were addressed on the SEWS forms and are being corrected via the applicable plant requirements.

8. SEISMIC INDUCED FIRE AND FLOOD EVALUATION

Seismic/fire interactions, effects of suppressants on safety equipment, and control system interaction should be addressed in the IPEEE per NUREG-1407. The majority of seismic/fire issues are identified in the NUREG/CR-5088 (Reference 30), and Information Notice 94-12, (Reference 31). A description of the fire suppression systems is given below. Specific fire issues are addressed later in this section.

8.1 DESCRIPTION OF FIRE SUPPRESSION SYSTEMS

The following five types of automatic fire suppression systems are used at Robinson:

1. Fire Water System;
2. Carbon Dioxide Fire Suppression System for Diesel Generator Rooms;
3. Halon Fire Suppression System;
4. Low Voltage Fire Detection System; and,
5. North and South Cable Vault High Pressure CO2 Fire Suppression System.

A description for each of the fire suppression systems is given below.

8.2 FIRE WATER SYSTEM

The fire water system is designed to accomplish the following objectives:

a. Provide an adequate water supply to appropriate fire protection systems and components so as to minimize losses to life and property.

b. Protect systems and equipment important to safety including equipment required for safe shutdown.

c. Provide reasonable assurance that a fire will not significantly increase the risk of radioactive release to the environment.

The fire main loop is part of and supplies the outside distribution system as well as the inside distribution system. The outside distribution system supplies fire hydrants and deluge sprinkler systems. The inside distribution supplies pre-action sprinkler systems, a dry standpipe system and hose stations.

The fire water system is part of the outside distribution system and provides water to five external fire hydrants and hose stations between the Intake Structure and the Power Block and sixty-seven fire hose stations located around and in the Power Block.

A deluge sprinkler system is also part of the outside distribution system and is used in areas where it is desirable to wet down an entire fire area by discharging water through open spray nozzles. These areas include the main transformer area, the auxiliary and startup transformer area, the hydrogen seal oil unit and hydrogen manifold area, and the turbine lube oil storage area. This deluge system was not evaluated as part of the Seismic/Fire/Flood walkdowns.

The pre-action system is part of the inside distribution system. It consists of closed sprinkler heads, water supply valves, controlling gate valves, interconnecting piping, manual and automatic actuation devices, and sensors to indicate system actuation. This inside system was evaluated by the Seismic Review Team for Seismic/Fire/Flood interactions.

Pre-action systems are primarily used to protect areas or equipment where there is serious danger from water damage as a result of damaged sprinkler heads or broken piping. Therefore, pre-action systems are used to protect the following areas:

A. Reactor Coolant Pumps - Zones 25A, 25B, and 25C;
B. Containment Electrical Penetration Area - Zone 24;
C. Auxiliary Building Hallway -Zone 12;
D. Solid Waste Handling Room - Zone 14.

Zones 25A, 25B, 25C, 24, and 12 were evaluated for the criteria in the Seismic/Fire/Flood caveats.

A pre-action system will only discharge water if heat from a fire has fused a sprinkler head causing loss of supervisory air and one of the following occurs:

1. The Low Voltage Fire Detection and Actuation System causes the water supply valve to open by the activation of an A and B train detector.

2. Manual activation on the Fire Detection and Actuation Panels for Zones 12 and 14, and on the Containment Fire Protection panel for Zones 24, 25A, 25B, and 25C.

3. The water supply valve is activated at the Manual Actuation Station.

The dry standpipe system consists of a dry standpipe (riser), hose station, dry pipe valve, controlling gate valve, interconnecting piping. The dry standpipe system provides fire water to Hose Station 52 in the southwest corner of the Hagan Room.

The wet pipe sprinkler system is the most widely used system. These systems consist of piping with an alarm check valve and sprinkler heads. If a fire occurs, the sprinkler head directly over the fire opens, discharging water. The water flows through the alarm check valve, sounds an alarm, and the sprinklers discharge until manually shut down. The only building and area evaluated by the Seismic/Fire/Flood walkdowns and protected by wet pipe sprinklers is the CCW Room (Partial Protection) in the Reactor Auxiliary Building.

The Dry Pipe Sprinkler System provides protection in the Chemical/Barrel Storage Warehouse which was not evaluated during the Seismic/Fire/Flood walkdowns. This system has a dry pipe valve with piping and sprinkler heads. Air pressure in the piping keeps the dry pipe valve clapper shut. Upon loss of air pressure i.e., sprinkler head fused), the valve will open, filling the system with water, and water will spray from any fused sprinkler heads.

8.3 CO_2 FIRE SUPPRESSION SYSTEM FOR DIESEL GENERATOR ROOMS

The purpose of the high pressure carbon dioxide (CO_2) Fire Suppression System for the diesel generator rooms is to extinguish a fire in either the "A" or "B" Diesel Generator Room.

The high pressure CO_2 system is located in the Auxiliary Building Hallway along the outer wall of the diesel generator rooms.

There are nineteen (19) seventy-five (75) pound capacity high pressure CO_2 cylinders and two (2) fifty (50) pound whistle alarm cylinders arranged into an "A" and a "B" bank. This "A" and "B" bank configuration refers only to the diesel generator room nearest each bank of cylinders. Upon system actuation, all nineteen (19) cylinders will discharge into the room containing the fire. All system controls are located adjacent to the "B" bank.

Pneumatic pressure controls operate the CO_2 system which is completely independent of all other Fire Suppression Systems. Pneumatic pressure is used to open valves, sound a whistle alarm and to operate switches all of which in turn activate the system and related equipment.

The CO_2 system must be either automatically activated by Heat Actuated-Devices (H.A.D.) located in one of the diesel generator rooms or by manual actuation in the Auxiliary Building Hallway.

There are several pieces of equipment that are associated with the CO_2 Fire Suppression System for the Diesel Generator Rooms. Panels FDAP-A1 and FDAP-B1 provide indication and annunciation of a local audible alarm, an evacuation alarm, and a zone actuation indication. Also, FPCD-68A, FPCD-68B, FPCD-69A, FPCD-69B are pressure switches that are included with this system. FPCD-68A and FPCD-68B are aligned with Zone 1 for the Emergency Diesel Generator B and stop the intake fan HVS-5 and the exhaust fan HVE-17 and close the fire dampers. FPCD-69A and FPCD-69B are aligned with Zone 2 for the Emergency Diesel Generator A and stop the intake fan HVS-6 and the exhaust fan HVE-18. Isolation of the diesel room ventilation makes the emergency diesel generators inoperable.

8.4 HALON FIRE SUPPRESSION SYSTEM

The purpose of the Halon 1301 fire extinguishing system is to provide a permanently installed automatic means of fire extinguishment for the Cable Spreading Room (Zone 19) and the Emergency Switchgear Room (Zone 20). Both of these rooms contain equipment identified as required for safe shutdown on the SSEL.

The Halon 1301 fire extinguishing suppression system is an integral part of the RNP fire protection system. Normal or automatic actuation is provided by a signal from the Low Voltage Fire Detection System (LVFDS). To ensure operation and to provide the required redundant actuation capabilities, the system may also be manually operated at the Fire Detection and Actuation Panels (FDAP's) or it can be manually actuated at the cylinder bank with the manual pull lever.

The Halon 1301 fire extinguishing agent is stored in specially designed cylinder assemblies. Most of these Halon 1301 Fire Suppression cylinders and associated equipment are located on the Turbine Building Mezzanine level at Elev. 242.5 feet.

The cylinders are divided into two distinct banks. The main bank is numbered A-1 through A-10 and the secondary bank or reserve bank is numbered B-1 through B-10. Either bank is sufficient to provide the required Halon concentration to extinguish a fire in either zone. The Halon system is still operational with the remaining bank of cylinders. There are four pressure switches that are associated with these Halon cylinders that were reviewed as part of the Seismic/Fire Walkdown evaluations. They are identified as A1-E1/2, B1-E1/2, A2-CSR, and B2-CSR. Switches A1-E1/2 and B1-E1/2 are the A and B train switches for the E1-E2 room and A2-CSR and B2-CSR are the switches for the Cable Spreading Room.

8.5 LOW VOLTAGE FIRE DETECTION SYSTEM

The purpose of the Low Voltage (24Vdc) Fire Detection System is to provide the control room with early indication of a fire by using automatic flame, heat, and smoke detectors and to protect various vital plant equipment and personnel by actuating selected fire suppression systems along with locally isolating ventilation fire dampers where applicable.

The Low Voltage (24 Vdc) Fire Detection System is divided up into thirty-two (32) fire detection zones. Each zone consists of one or more of the following: Ionization smoke, photo-electric smoke, infrared flame, thermal fire detectors, and manual pull stations. The detection signal is sent to a Fire Detection and Actuation Panel (FDAP). There are four (4) FDAP panels located in different areas of the plant. These were identified earlier and were evaluated during the walkdowns.

8.6 NORTH AND SOUTH CABLE VAULT HIGH PRESSURE CO_2 FIRE SUPPRESSION SYSTEM

The purpose of the North (Zone 9) and South (Zone 10) Cable Vault High Pressure Carbon Dioxide Fire Suppression System is to extinguish a fire in the North and South Cable Vaults. The use of CO_2 in one vault does not affect fire suppression capabilities for the remaining vault.

The CO_2 contains thirty-six (36) seventy-five (75) pound high pressure CO_2 cylinders arranged into a Main and Reserve Bank which are piped to a common manifold. There are eighteen cylinders in each bank. The Main and Reserve Banks are redundant. Main and Reserve bank cylinders 1 through seven are for the North Cable Vault. Main and Reserve Bank cylinders one through eighteen are for the south vault. Discharge of CO_2 from either the main or Reserve Bank for either the North or South Cable Vault will leave the opposite bank in a back-up status once the Main-Reserve selector switches are properly placed.

Electro-pneumatic controls operate the CO_2 system. The Low Voltage Fire Detection System will automatically actuate the CO_2 system for the affected vault upon receipt of two separate fire alarm signals from the same zone. Remote-manual operation is accomplished at the Fire Detection and Actuation Panel (FDAP) A-1 or B-1 and the manual pull stations located outside the South Cable Vault. The system can be manually actuated at the cylinder storage area in the pipe alley by operating the appropriate valves and cylinders for each vault.

The pressure switches that control the CO_2 high pressure cylinders are similar to those used to actuate the cylinders for the Cable Spreading Room and the E1/E2 Room. These pressure switches are identified as A1-NCV, A2-SCV, B1-NCV, and B2-SCV. These identifications represent equipment for the A and B Train for the North Cable Vault (A1-NCV and B1-BCV) and the South Cable Vault (B1-SCV and B2-SCV).

8.7 SCOPE OF WALKDOWN EVALUATIONS

The following fire zones and fire suppression systems were evaluated during the seismic/fire walkdowns in the Reactor Containment Building and the Reactor Auxiliary Building:

1. Fire Zone 1 - Diesel Generator "B" Room

    SAFE SHUTDOWN EQUIPMENT - EDG-B, EDG-B Day Tank, EDG-B Engine Control Panel

    FIRE SUPPRESSION - Automatic High Pressure Carbon Dioxide System

    ISSUE - Seismic support of components and interaction.

2. Fire Zone 2 - Diesel Generator "A' Room

    SAFE SHUTDOWN EQUIPMENT - EDG-A, EDG-A Day Tank, EDG-B Engine Control Panel

    FIRE SUPPRESSION - Automatic High Pressure Carbon Dioxide System

    ISSUE - Seismic support of components and interaction.

3. Fire Zone 4 - Volume Control Tank Room

    SAFE SHUTDOWN EQUIPMENT - Volume Control Tank, CP-A, CP-B, and CP-C

    Concern - Hydrogen piping to the Volume Control Tank is located in this area.

4. Fire Zone 5 - Component Cooling Water Room

    SAFE SHUTDOWN EQUIPMENT - CCW Pump A, CCW Pump B, CCW Pump C, CCW-A-HTX, CCW-B-HTX, BAST-A, BAST-B, Boric Acid Transfer Pump A, and Boric Acid Transfer Pump B

    FIRE SUPPRESSION - Wet Pipe Automatic Sprinkler System

    ISSUE - Seismic support of components and interaction.

5. Fire Zone 7 - Auxiliary Building 226 Hallway (Ground Floor)

   SAFE SHUTDOWN EQUIPMENT - Cable runs for both normal shutdown trains, MCC-5, MCC-10

   FIRE SUPPRESSION - Automatic Preaction Sprinkler System

   ISSUE - Seismic support of components and interaction.

6. Fire Zone 9 - North Cable Vault

   SAFE SHUTDOWN EQUIPMENT - Cables for Train A and B equipment and instrumentation inside containment

   FIRE SUPPRESSION - Automatic High Pressure Carbon Dioxide System

   ISSUE - Seismic support of components and interaction.

7. Fire Zone 10 - South Cable Vault

   SAFE SHUTDOWN EQUIPMENT - Cables for Train A and B equipment and instrumentation located inside containment

   FIRE SUPPRESSION - Automatic High Pressure Carbon Dioxide System

   ISSUE - Seismic support of components and interaction

8. Fire Zone 15 - Auxiliary Building 246 Hallway (Second level)

   SAFE SHUTDOWN EQUIPMENT - Cable runs for both normal shutdown trains, and the building supply and exhaust ventilation fans

   ISSUE - Hydrogen piping to the Volume Control Tank is routed for a short distance in this area.

9. Fire Zone 19 - Unit 2 Cable Spreading Room

SAFE SHUTDOWN EQUIPMENT - Cable runs including those for control and instrumentation for all of the normal shutdown systems.

FIRE SUPPRESSION - Automatic Halon 1301 System

ISSUE - Seismic support of components and interaction

10. Fire Zone 20 - E1/E2 Room (Emergency Switchgear Room and Electrical Equipment Area) SAFE SHUTDOWN EQUIPMENT - Switchgear E1, E2, MCC-6, and MCC-9.

FIRE SUPPRESSION - Automatic Halon 1301 System

ISSUE - Seismic support of components and interaction

11. Fire Zone 25A - Reactor Coolant Pump A Bay

SAFE SHUTDOWN EQUIPMENT - Reactor Coolant Pump

ISSUE - Reactor Coolant Pump A contains a lubricating oil reservoir which could leak oil if damaged and potentially cause a fire.

12. Fire Zone 25B - Reactor Coolant Pump B Bay

SAFE SHUTDOWN EQUIPMENT - Reactor Coolant Pump

ISSUE - Reactor Coolant Pump B contains a lubricating oil reservoir which could leak oil if damaged and potentially cause a fire.

13. Fire Zone 25C - Reactor Coolant Pump C Bay

SAFE SHUTDOWN EQUIPMENT - Reactor Coolant Pump

ISSUE - Reactor Coolant Pump C contains a lubricating oil reservoir which could leak oil if damaged and potentially cause a fire

8.8 SPECIFIC COMPONENTS EVALUATED

The following specific pieces of fire suppression equipment were identified on the Safe Shutdown Equipment List and evaluated during the A-46/Seismic IPEEE fire/interaction walkdowns:

1. **FDAP-A1** - Fire Detection and Actuation Panel A1
2. **FDAP-A2** - Fire Detection and Actuation Panel A2 (This FDAP-A2 also contains FP-FDRP-A2)
3. **FDAP-B1** - Fire Detection and Actuation Panel B1
4. **FDAP-B2** - Fire Detection and Actuation Panel B2 (This FDAP-B2 also contains FP-FDRP-B2)
5. **FP-FDRP-A1** - Fire Damper Relay Panel A1
6. **FP-FDRP-B1** - Fire Damper Relay Panel B1
7. **A1-E1/2** (Listed as FPHS-A1 in EDBS) - Pressure Switch for Train A Zone 20 Halon
8. **A1-NCV** (Listed as FPCD-A1 in EDBS)- Pressure Switch for Train A Zone 9 NCV
9. **A2-CSR** (Listed as FPHS-A2 in EDBS) - Pressure Switch for Train A Zone 19 Halon
10. **A2-SCV** (Listed as FPCD-A2 in EDBS)- Pressure Switch for Train A Zone 10 SCV
11. **B1-E1/2** (Listed as FPHS-B1 in EDBS)- Pressure Switch for Train B Zone 20 Halon

12. **B1-NCV** (Listed as FPCD-B1 in EDBS) - Pressure Switch For Train B Zone 9 NCV

13. **B2-CSR** (Listed as FPHS-B2 in EDBS) - Pressure Switch for Train B Zone 19 Halon

14. **B2-SCV** (Listed as FPCD-B2 in EDBS)- Pressure Switch for Train B Zone 10 SCV

15. **FPCD-68A** - Pressure Switch for Train A to Diesel Generator B Cardox

16. **FPCD-68B** - Pressure Switch for Train B to Diesel Generator B Cardox

17. **FPCD-69A** - Pressure Switch for Train A to Diesel Generator A Cardox

18. **FPCD-69B** - Pressure Switch for Train B to Diesel Generator B Cardox

The purpose of the walkdowns for these components in these zones is twofold The first purpose was to verify that tanks and piping associated with flammable liquids located in the Reactor Containment Building and the Reactor Auxiliary Building will not be damaged by interaction during a seismic event. If these components and piping are susceptible to failure and a potential exists for a resulting fire that could damage the seismic safe shutdown components, simple fixes should be considered to improve their seismic capability.

The second purpose was intended to verify the design of the fire suppression systems located in the Reactor Containment Building and Reactor Auxiliary Building to determine that the suppression systems will not be subjected to spurious operations or failures leading to loss of integrity during a seismic event. If potential seismically induced failure is deemed to be a possibility, the vulnerability of specific safe shutdown equipment to spraying or flooding should be considered.

The seismic capability of fire suppression systems and components containing flammable materials was assessed using the same techniques used to address the seismic safe shutdown equipment. Emphasis was placed on ensuring tanks had proper anchorage, piping was not subject to large deflections and there are no potential interactions such as pipes or sprinkler heads impacting other objects. The aim was to demonstrate that such components were sufficiently robust with respect to maintaining their integrity that they would meet the screening criteria adopted at Robinson for seismic safe shutdown components.

The following seismic/fire issues identified in the NUREG/CR-5088 "Fire Risk Scoping Study," (Reference 30) were considered and evaluated during the walkdowns:

- **Identify unanchored CO_2, Halon, oxygen, or hydrogen tanks.**

Gas bottles are stored in the following locations:

1. RAB, El. 226, North 226 Corridor outside of the Emergency Diesel Generator Room A and Emergency Diesel Generator Room B - These carbon dioxide bottles are associated with the CARDOX system for the Diesel Generator Rooms and are stored in seismically designed storage racks.

2. RAB, El. 226, south section of Pipe Alley - These carbon dioxide bottles are associated with the CARDOX system for the North and South Cable Vault Rooms and are stored in seismically designed storage racks.

3. Class III Bay Turbine Generator Building, El. 242.5, below stairs leading to the Control Room - These Halon bottles are associated with the system for the Cable Spread Room and the E1/E2 Room and are stored in storage racks designed to prevent overturning.

4. RAB, El. 226, south 226 Corridor outside of the CCW Pump Room - These bottles are stored in seismically designed storage racks.

All gas bottles are adequately stored and restrained within the power block to preclude any damage to SSEL equipment and permanent plant equipment.

- **Identify actuation systems that are sensitive to vibration, relay chatter, and/or locking circuits:**

The Fire Detection and Actuation Panels FDAP-A1, FDAP-A2, FDAP-B1, and FDAP-B2 and the Fire Damper Relay Panels FP-FDRP-A1 and FP-FDRP-B1 (FP-FDRP-A2 and FP-FDRP-B2 are installed within FDAP-A2 and FDAP-B2 respectively) all have relays mounted within them. Panels FDAP-A1, FDAP-B1, FDAP-B2, FP-FDRP-A1, and FP-FDRP-B1 were judged by the Seismic Review Team to be seismically anchored to the floor and to the adjacent wall and there would be no inadvertent spurious chatter. Individual SEWS have been prepared for these components. Panel FDAP-A2 was anchored to the floor but was not anchored to an adjacent wall because of its location in the E1/E2 Room. The panel is installed adjacent to an MCC and was reviewed for interaction with this cabinet. It was determined that the design system logic would not allow fire suppression activation even if the relays in this panel were to chatter. The system logic requires signals from both the FDAP-A1 and the FDAP-A2 for activation and since FDAP-A1 is seismically anchored, the required scenario is not possible.

The remaining components identified above are all pressure switches. These components are identified as A1-E1/2, A1-NCV, A2-CSR, A2-SCV, B1-E1/2,B1-NCV, B2-CSR, B2-SCV, FPCD-68A, FPCD-68B, FPCD-69A, AND FPCD-69B. All of these components have been added to the Robinson Plant Safe Shutdown Equipment List. Seismic Evaluation Work Sheets (SEWS) have been completed for each of these components and are available for review. The sheets document the conclusions concerning the installation and interaction caveats for the equipment. All components except A1-E1/2 and B1-E1/2 were determined to have adequate seismic anchorage and would not be impacted by any adjacent equipment or components. Components A1-E1/2 and B1-E1/2 were adequately anchored but an interaction issue existed because a loose sheet metal cover was located above the switches which could displace during a seismic event and potentially impact the switches. It has been recommended to plant management to remove the sheet metal cover.

- **Identify fire detection systems with only ionization detectors where dust may cause a spurious alarm:**

Table ? lists the fire zones within the Reactor Containment Building and the Reactor Auxiliary Building where ionization detectors are installed. Typically, an ionization detector from one safety train works in conjunction with another detection device from the other safety train to determine if there is a legitimate fire source before activation of the fire suppression system. However, Zone 19 (Cable Spreading Room) and Zone 24 (Containment Electrical Penetration room) have ionization detectors associated with both trains which could detect smoke at the same time and activate the fire suppression system for those zones. The suppression for the cable spread room is Halon. Halon would fill the room. However, if the detection were not required, then operators could expel the Halon by simply opening the room doors and venting the Halon. The suppression system for the Containment Electrical Penetration Room is the Preaction Sprinkler System. Detection by the A and B Train ionization components would cause the header pipes to become charged with water but the system is designed to not immediately discharge the water but to wait for another signal from the heat detectors. Therefore, despite the activation by the ionization detectors, the equipment located within these areas would not necessarily be damaged. All of the other rooms having ionization detectors would not result in any inadvertent actuation of the fire suppression system but would only causes an alarm signal.

- **Identify fire pumps that have weak mounts or vibration mounts:**

There are three pumps associated with the firewater system. There is one motor driven fire pump, one fire water booster pump, and one emergency fire pump and engine located at the Circulating Water System Intake Structure (i.e. Service Water Intake Structure). The pumps are anchored to the concrete with cast-in-place anchor bolts without vibration isolator mounts. Therefore, the mounting of the fire pumps is not a concern.

- **Verify that all electrical cabinets are properly anchored and have sufficient slack in the cables entering the cabinet:**

The Seismic Review Team walkdown included an interaction review between the SSEL electrical cabinets and adjacent components. The SRT walkdown also verified whether there was adequate slack existing in the cables entering the electrical cabinets. Several of the electrical cabinets have rigid conduit routed from the overhead directly into the top. However, the SRT determined that there was adequate flexibility for this type of configuration. Several of the cabinets were noted as having marginal anchorage but were judged not to represent any operability issues. The anchorage issues have been addressed to the plant for modification.

- **Identify credible interactions between sprinkler systems and adjacent piping:**

The Seismic Review Team evaluated the water piping routed to the sprinkler heads and other dispersion devices in the Auxiliary Building 226 Hallway and the Component Cooling Water Room. The piping was observed to be well supported with engineered type supports installed at regular intervals. This installation is consistent with the current design guidelines for the plant where all piping, whether Q or Non-Q, will be seismically supported within a Q, Safety Related area of the plant. The Seismic Review Team concluded that the water supply piping is well supported to assure system pressure integrity after a SSE. No interaction issues between the sprinkler system and adjacent piping were noted during the SRT walkdown.

The following seismic/fire issues are identified in NRC Information Notice 94-12 (Reference 31) and were considered and evaluated during the walkdowns:

- **Mercury Relays**

  No mercury relays have been located in the fire protection circuits.

- **Seismic Dust/Smoke Detectors:**

Several fire zones within the power block have ionization detectors only which sends an alarm to the control room. This information noted above provides the required discussion. Actuation of the fire suppression system (pre-actuation sprinklers) is only by thermal detection and is independent of the detection. Therefore, inadvertent actuation will not occur.

- **Water Deluge Systems:**

The Seismic Review Team walkdown included an interaction review for potential sources that could flood or spill onto the electrical cabinets. This issue is addressed on the SEWS forms for the electrical cabinets. No interaction issues, with respect to flooding or fire protection systems flooding electrical cabinets were noted during the SRT walkdowns. No further action is needed.

- **Fire Suppressant Availability During a Seismic Event:**

There are three pumps associated with the firewater system. There is one motor driven fire pump, one fire water booster pump, and one emergency fire pump and engine located at the Circulating Water System Intake Structure (i.e. Service Water Intake Structure). If a seismic event affects the operation/functionality of these pumps, water can be provided by a variety of sources as detailed in the Robinson Plant Operating Procedures OP-801. This procedure provides direction for the connection of available water supplies through the Unit 2 system. Operators would cross connect to the Unit 2 Fire Water System. Section 8.36 through 8.39 provides directions for using the Unit 1 Fire Water System and the Unit 1 electric pump, using the Unit 1 Fire Water System and the ash sluice pumps, using a fire pumper truck or auxiliary fire pump taking suction from the lake or discharge canal, or using the Darlington County water system. Therefore, the operation/functionality of the non-safety related fire pumps is not an issue because of the other sources.

- **Switchgear Fires:**

The Seismic Review Team walkdown included screening/evaluation of switchgear. The SRT walkdown verified adequate switchgear anchorage, sufficient slack in cables entering the switchgear or sufficient flexibility of the rigid conduit attached to the top of the switchgear cabinets and sufficient separation to other electrical cabinets. The results from the SRT walkdown did not identify any seismic/fire interaction issues with regards to the switchgear.

In addition to the seismic/fire interactions addressed above, fire sources are to be identified and evaluated. The following fire sources were identified and evaluated:

1. **Hydrogen Hazard** -As noted above, the sole source for the gaseous fire hazard was a hydrogen line that supplies hydrogen gas to the Volume Control Tank to inhibit corrosion within the tank and system. The line originated at parked tankers located outside of the Unit 2 security fence on the grounds of Unit 1. The line is routed underground from the tanker until it surfaces at the northeast corner of the exterior of the Reactor Auxiliary Building. The line was then routed up the north exterior face of the RAB to an elevation approximately 20 feet above grade. The line

proceeded to be routed to the south along the exterior west face of the RAB until it entered the Auxiliary Building through a grouted penetration. The hydrogen line entered the Auxiliary Building from the outside into the MCC-18-SB room. The line was routed through this MCC-18-SB room, into the Reactor Coolant Filter room through a grouted penetration, and then through another penetration into the Room containing the Volume Control Tank. The Seismic Review Team evaluated the hydrogen line as it was routed on the exterior and on the interior of the building. The hydrogen line has adequate attachment and supports at both locations. The Team noted that the interior support is seismic in maintaining the design approach where all interior components are seismically supported. The Seismic Review Team concluded that the hydrogen line is adequately supported and a seismic event will not cause any credible interaction concerns to damage the line. The SRT judges that there will be no seismic/fire interaction issues.

2. **Liquid Fire Hazards**- There were three liquid fire hazard sources that were identified in areas where safety related equipment are present. The Seismic Review Team evaluated all three of these potential hazards.

   One source was the lubricating oil reservoir that is an integral part of each of the Reactor Coolant Pumps A, B, and C. The Team determined that the lubricating oil reservoirs and associated piping for the Reactor Coolant Pumps A, B, and C were sufficiently attached to the main structural steel housings and would not be dislodged or damaged by adjacent equipment and components by a seismic event. The reactor coolant pumps are also substantially anchored and braced within the pump bays and they are judged to also not displace enough to come into contact with any adjacent equipment or components

   The second potential liquid fuel source is the Auxiliary Boiler Fuel Oil Pumps. Both of these small pumps are located adjacent to the

Auxiliary Feedwater Steam Driven Pump. The oil piping routed to the Auxiliary Boiler pumps and then to the Auxiliary Boilers was supported with typical commercial/industrial type of flexible rod-hung type hangers. The hangers were not regularly spaced but there was no indication of any sagging or other extraordinary conditions. The piping was installed in an East-West direction from the structural steel of the Class I Bay of the Turbine Generator Building. There were no anchor type supports or lateral supports to prevent excessive displacements. The piping was connected to the Auxiliary Boiler pumps with threaded screw type fittings. There was some concern about the potential for these threaded fittings breaking loose from the connection and then oil spraying over to the Auxiliary Feedwater Steam Driven Pump. However, further research showed that the auxiliary boilers are only in operation during startup and shutdown. The Auxiliary Boilers and therefore the auxiliary boiler pumps are not in operation when the nuclear reactor is at full power. It was also learned that the Auxiliary pumps are a low head pump and do not have the pressure to spray oil from the pump location over to the Steam Driven Auxiliary Feedwater Pump location. It was also stated that diesel oil is not easily flammable and would not be highly susceptible to catching on fire based on the scenario caused by a seismic event. Therefore, the Seismic Review Team concludes that the fuel oil to the auxiliary pumps and boilers is not subject to causing any fire concerns due to seismic interaction with other components.

The third potential fire source was the Diesel Generator Fuel Oil Day Tanks A and B located in Diesel Generator Buildings A and B respectively. Lastly, the Seismic Review Team walkdown included the Diesel Generator Day tanks which are on the Safe Shutdown Equipment List. The tanks were screened out from further review during the SRT walkdowns based on the seismic hangers that are currently installed around the tanks.

Other sources of liquid fire hazards are the lube oil systems of various pumps (excluding the RCP's). These sources are insignificant in terms of risk and can be ignored in the seismic/fire walkdowns. Even though these may be insignificant, the lube oil systems of the pumps included on the SSEL were considered rule-of-the-box and evaluated/screened with the pump during the SRT walkdowns. The results from the SRT walkdown did not identify any vulnerabilities in this area.

No further action is required as a result of the seismic/fire evaluations reviews and walkdown.

8.9 SUMMARY

The A-46 walkdowns incorporated the requirements of the IPEEE to evaluate the Seismic/Fire interaction issue. The final conclusion is that the components of the Robinson fire suppression system including firewater piping, fire detection panels, pressure switches, equipment associated with gaseous hazards, and equipment associated with liquid hazards are adequately supported and isolated with respect to other adjacent equipment. The Seismic Review Team has determined that the continued functionality and operation of equipment and components required for the safe shutdown of the plant will not jeopardized by Seismic/Fire Interaction.

Table 8-1

FIRE ZONES WITH SMOKE IONIZATION DETECTORS

ZONE	BUILDING/LOCATION	COMMENTS (Other Detectors/Stations)
1	RAB/Diesel Generator Rm B	Train A - None Train B - None (Infrared Flame, Heat, and Manual Pull Station)
2	RAB/Diesel Generator Rm A	Train A - None Train B - None (Infrared Flame, Heat, and Manual Pull Station)
3	RAB/Safety Injection Pump Room	Train A - 1 Smoke Ionization Train B - 1 Smoke Ionization (Heat and Manual Pull Stations)
4	RAB/Charging Pump Room	Train A - 2 Smoke Ionization Train B - None (Heat and Manual Pull Stations)
5	RAB/Component Cooling Water Pump Room	Train A - 3 Smoke Ionization Train B - None (Heat and Manual Pull Stations)
6	Hot Lab	No SSEL Components in this Fire Zone
7	RAB/Motor Driven Auxiliary Feedwater Pump Room	Train A - 1 Smoke Ionization Train B - None (Heat and Manual Pull Station)
8	RAB/Boron Injection Tank Room	Train A - 1 Smoke Ionization Train B - None (Heat and Manual Pull Stations)
9	RAB/North Cable Vault Room	Train A - 1 Smoke Ionization Train B - None (Smoke Photo-Electric, Heat, and Manual Pull Station)
10	RAB/South Cable Vault Room	Train A - 2 Smoke Ionization Train B - None (Heat, Smoke Photo-Electric, and Manual Pull Station)

Table 8-1 (Continued)

FIRE ZONES WITH SMOKE IONIZATION DETECTORS

ZONE	BUILDING/LOCATION	COMMENTS (Other Detectors/Stations)
11	RAB/North 226 Corridor	Train A - 2 Smoke Ionization Train B - 2 Smoke Ionization (Heat and Manual Pull Station)
12	RAB/Center 226 Corridor	Train A - 2 Smoke Ionization Train B - 2 Smoke Ionization (Smoke Photo-Electric, Heat, and Manual Pull Stations)
13	RAB/South 226 Corridor	Train A - 2 Smoke Ionization Train B - 2 Smoke Ionization (Heat and Manual Pull Stations)
14	RAB/Drumming Room	No SSEL Components in this Fire Zone
15	RAB/246 Corridor	Train A - None Train B - 5 Smoke Ionization (Heat and Manual Pull Station)
16	RAB/Battery Room	Train A - 1 Smoke ionization Train B - 1 Smoke Ionization (Heat and Explosion Proof Station)
17	RAB/HVAC Equipment Room	Train A - 3 Smoke Ionization Train B - 1 Smoke Ionization (Heat and Manual Pull Station)
18	RAB/Unit 1 Cable Spread Room	Train A - 1 Smoke Ionization Train B - 1 Smoke Ionization No SSEL Components in this Fire Zone (Manual Pull Station)
19	RAB/Cable Spread Room	Train A - 3 Smoke Ionization Train B - 5 Smoke Ionization (Smoke Photo-Electric, Heat, and Manual Pull Station)
20	RAB/E1-E2 Emergency Switchgear Room	Train A - 4 Smoke Ionization Train B - None (Heat, Smoke Photo-electric, and Manual Pull Station)
21	RAB/Rod Drive Control	Train A - 2 Smoke Ionization

Table 8-1 (Continued)

FIRE ZONES WITH SMOKE IONIZATION DETECTORS

ZONE	BUILDING/LOCATION	COMMENTS (Other Detectors/Stations)
	Room	Train B - 1 Smoke Ionization (Manual Pull Station)
22	RAB/Control Room	Train A- 11 Smoke Ionization Train B - 5 Smoke Ionization (Heat)
23	RAB/Hagan Room	Train A - 1 Smoke Ionization Train B - 1 Smoke Ionization (Heat, Manual Pull Station)
24	RCB/Containment Electrical Penetration Area	Train A - 2 Smoke Ionization Train B - 2 Smoke Ionization (Heat, Manual Pull Station)
25a	RCB/RCP A Bay	Train A - None Train B - None (Heat, Infrared Flame)
25b	RCB/RCP B Bay	Train A - None Train B - None (Heat, Infrared Flame)
25c	RCB/RCP C Bay	Train A - None Train B - None (Heat, Infrared Flame)
26 - HVH-1 HVH-2 HVH-3 HVH-4	RCB/Operating Level - Elevation 289?	Train A- None;Train B- None Train A- None;Train B- None Train A- None;Train B- None Train A- None;Train B- None (Heat, Smoke Photo-Electric)
27	RCB/RHR Pump Room	Train A - 1 Smoke Ionization Train B - 1 Smoke Ionization (Heat, Smoke Photo-Electric, Manual Pull Station)
28	RAB/Pipe Alley	Train A - 2 Smoke Ionization Train B - 3 Smoke Ionization (Heat, Manual Pull Station)
29	Class III TGB/4160 Switchgear Room	Train A - 9 Smoke Ionization Train B - None No SSEL Components in this Fire Zone
30	RAB Roof/Battery Room C	Train A - 2 Smoke Ionization Train B - None No SSEL Components in this Fire Zone

9. CONTAINMENT INTEGRITY

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 provided 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 such as spatial interactions, unique penetration configurations, etc. 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 which is addressed in another section of this report.

A review of seismic capacities for containments of similar design to Robinson indicates that the containment structure is expected to have a seismic capacity far above the review level earthquake (see Section 2.6). The combination of concrete, reinforcing steel, and an internal steel liner plate is satisfactory to prevent any escape of radiation or radioactive material. In addition to the containment structure, NUREG-1407 suggests that certain considerations could require some additional study. The items requiring additional study are the electrical penetrations, the piping penetrations, and the personnel and equipment hatches.

In general, a penetration consists of a sleeve embedded in the concrete wall and welded to the containment liner. The weld to the liner is shrouded by a continuously pressurized channel which is used to demonstrate the integrity of the penetration-to-liner weld joint. The pipe, electrical conductor cartridge, or duct passes through the embedded sleeve and the ends of the resulting annulus are closed off, either by welded end plates, bolted flanges, or a combination of these. Pressurizing connections are provided to continuously demonstrate the integrity of the penetration assemblies.

There are 50 electrical penetrations. Electrical penetrations can be the cartridge type (48 of the 50) or the capsule type (2 of the 50) penetrations. The cartridge type penetration is used for all electrical conductors passing through the containment except for one penetration in the north cable vault and one in the south cable vault. The penetration cartridge is a hollow cylinder closed on both ends through which the conductors pass. The cartridge is provided with a pressure connection to allow continuous pressurization of the penetration. There are several methods used to seal the joint between the cartridge end plate and the conductor. In the capsule penetration, a single stainless steel plate is machined with the required quantity of feed-through ports which are interconnected by peripherally machined gun drills which creates a manifold system for pressure monitoring. These methods are explained in greater detail in the Robinson Updated FSAR in Section 3.8.1.1.6.1 concerning electrical penetrations.

Double barrier piping penetrations are provided for all piping passing through the containment. The pipe is centered in the embedment sleeve which is welded to the liner, except for small pipes where several pipes may pass through the same penetration sleeve. The penetrations for the main steam, feedwater, blowdown, and sample lines are designed so that the penetration is stronger than the piping system and that the vapor barrier will not be breached due to a hypothesized pipe rupture. End plates are welded to the pipe at both ends of the sleeve. A connection to the penetration sleeve is provided to allow continuous pressurization of the compartment formed between the piping and the embedded sleeve. If a pipe is carrying hot fluid, the pipe is insulated and cooling is provided to maintain the concrete temperature adjoining the embedded sleeve below 150 degrees Fahrenheit. There are 46 containment penetration sleeves for pipes. Pipes have anchor supports to the structural steel girders as close as possible to the inside of the wall or to the crane wall. Loads due to pipe ruptures within the containment or due to thermal stresses are not transferred to the liner.

The Containment Isolation (CI) System is designed to isolate the interfaces between the containment and the environment such that any radioactivity released into the containment atmosphere following a postulated initiating event will be confined to Containment. CI is not a system but rather the combined operation of components in several systems which work in concert to isolate containment following an initiating event. Piping penetrating the containment was designed for pressure at least equal to the containment design pressure. Isolation valves were provided as necessary for all fluid system lines penetrating the containment to assure at least two barriers for redundancy against leakage of radioactive fluids to the environment. The valves were designed to facilitate normal operation and maintenance of the system and to ensure reliable operation of other engineered safeguards systems. In general isolation of a line outside the containment protects against rupture of the line inside concurrent with a LOCA, or closes off a line which communicates with the containment atmosphere in the event of a LOCA.

An equipment hatch is provided which is fabricated from welded steel and furnished with a double-gasketed flange and bolted dish door. The hatch barrel is embedded in the containment wall and welded to the liner and is a portion of the structural frame embedded in the wall. Provision is made to continuously pressurize the space between the double gaskets of the door flanges and the weld seam channels at the liner joint, hatch flanges, and dished door. Pressure is relieved from the double gasket spaces prior to opening the door.

The personnel hatch is a double door, manually operated, hydraulically latched, welded steel assembly fabricated by the Chicago Bridge and Iron Company. It is attached to the structural frame embedded in the wall of which the frame barrel forms the central portion of the lock. The personnel hatch doors have a mechanical interlock mechanism that prevents both doors from being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. The mechanical interlock can be defeated for specific reasons such as during outages to expedite entry into the containment building. Otherwise, the mechanical interlock mechanism is always fully functional.

The fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the containment and the spent fuel pit. There is an inner and outer pipe. The inner pipe is the transfer tube and is fitted with a pressurized double-gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pit. This arrangement prevents leakage through the transfer tube. The outer pipe is welded to the containment liner and provision is made by use of a special seal ring for pressurizing all welds essential to the integrity of the penetration during plant operation. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures.

The ventilation system purge are each equipped with two quick-acting tight-sealing butterfly valves for isolation. The space between the valves is pressurized above incident pressure while the valves are normally closed during plant operation.

Containment heat removal is an important aspect in evaluating containment performance. If heat is not adequately removed from the containment, the containment pressure may increase to the containment failure pressure. Two mechanisms can lead to energy being transmitted to the containment. The first is due to a small LOCA. As Reactor Coolant System (RCS) inventory is lost through the break, it carries stored energy which is then released to the containment and pressurization occurs. Feed and bleed cooling also results in energy being transferred to the containment. Containment fan coolers can reduce the pressurization due to these mechanisms.

The containment pressure is not expected to increase to the design limit as long as the RHR heat exchangers are available to remove heat. Thus, the fan coolers represent an additional heat removal mechanism but are not required for successful containment cooling as long as the RHR heat exchangers are present. Failure of this heat removal function will result in containment heatup and pressurization.

The pressurization, however, is predicted to occur over many hours and would not result in an early, rapid containment overpressurization. It is concluded that containment fan coolers are not needed to ensure early containment integrity. As a result, the only containment issues to be addressed are the seismic relay review and walkdown. The relay review is addressed in Section 7.0.

The Containment walkdown consisted of observing the penetrations for any unusual conditions or configurations such as spatial interactions, unique penetrations, piping hard spots, etc. The containment walkdown was performed by the Seismic Review Teams.

No unusual conditions or configurations were identified during the containment walkdowns. The main objective of the containment analysis was to identify vulnerabilities that involve early failure of containment functions. The SRT review and walkdown performed for the containment did not reveal any significant vulnerabilities.

10. PEER REVIEW SUMMARY

The Robinson IPEEE peer review was performed by Mr. Charbel M. Abou-Jaoude and Mr. Steve Reichle of Vectra Technologies, Inc. during May, 1995. Peer reviewer resumes are included in Appendix A.

The IPEEE efforts for the Robinson Plant were found to have been conducted in a very thorough and competent manner. The Peer reviewers found that the programs are being performed in accordance to the guidance of the SQUG GIP and EPRI NP-6041, in addition the seismic reviews met the stated objectives of NUREG-1407. The results and findings from the program appear to be reasonable and are consistent with expectations for a plant of this vintage. A number of voluntary upgrades to equipment were noted during the plant walk-through which have resulted in improved seismic ruggedness; in addition a number of the outliers that were noted by the SRT were in the process of being upgraded during the outage indicating good initiative and responsiveness to seismic issues.

11. SUMMARY AND CONCLUSIONS

The Robinson seismic IPEEE was completed in accordance with NUREG 1407 guidelines using the EPRI seismic margins methodology provided in EPRI NP-6041.

The most important aspect of the program was plant walkdowns. Detailed SRT walkdowns were performed in conjunction with A-46 walkdowns using the methodology, criteria and SEWS provided in EPRI NP-6041 and the GIP.

The SRT identified 33 issues related to maintenance, housekeeping and seismic interaction that required work orders to satisfy SRT field issues. 21 items were noted as requiring repairs or modifications. Several components were identified for subsequent HCLPF evaluation.

The relay evaluation successfully evaluated all 789 success path relays as follows:

- 707 were screened based on seismic capacity. Relay seismic capacity is greater than seismic demand at the RLE.
- 72 were screened by system consequence review. Relay chatter would not adversely impact success path equipment.
- 14 were screened based on operator actions. Any effects of relay chatter would be mitigated by operator action in accordance with operating procedures.
- No low ruggedness relays were encountered.

The evaluation concluded that the Robinson plant HCLPF is at least 0.28g. Only one issue contributed to a capacity less than the 0.3g RLE. A HCLPF capacity of 0.28g was calculated for two motor operated valves, RHR-750 and RHR-751. Calculated stresses in the nodular (ductile) iron valve yokes exceed the allowable at the RLE. The evaluation considered operating stem thrust loads and a cast iron allowable equal to 20% of the specified minimum ultimate capacity.

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APPENDIX A

SEISMIC REVIEW TEAM QUALIFICATIONS

PEER REVIEWER RESUMES

JEFFREY H. BOND

Mr. Bond has over seventeen years of experience in the design, analysis, testing and qualification of industrial and nuclear systems, structures, and components. His responsibilities have included finite element modeling and analysis; vibration testing and analysis; and load, shock, vibration, and environmental testing for hardware qualification. His experience includes fourteen years with an engineering consulting organization with primary responsibilities in the area of equipment qualification for both manufacturers and utilities. His three years of experience with CP&L have included design responsibilities, NRC audit preparations, forced-outage plant material condition resolution programs, and responsibility for SQUG/IPEEE implementation at CP&L's Brunswick Plant. He completed the SQUG and IPE Seismic Add-On courses in preparation for participation in USI A-46/IPEEE resolution at all CP&L's nuclear power plants. He holds both BS and MS degrees in mechanical engineering, and is a registered professional engineer in the state of North Carolina.

STEVEN R. BOSTIAN

Mr. Bostian has over thirteen years of experience in nuclear plant construction and design. This experience includes two years of on-site field engineering during the construction phase of the Comanche Peak Nuclear Power Plant, three years of on-site field engineering during the construction phase of the Shearon Harris Nuclear Plant, six years of civil/structural design engineering for the three nuclear plants operated by Carolina Power and Light Company, and two years in the USI A-46/Seismic IPEEE project. Primary engineering responsibilities have been in seismic support design and justification of mechanical and electrical components including electrical raceway, small and large bore piping systems, instrumentation, HVAC equipment, cabinets, and panels. He was selected for CP&L's USI A-46/Seismic IPEEE project in late 1992. He completed the SQUG and IPE Seismic Add-On courses in preparation for participation in the USI A-46/Seismic IPEEE resolution at all CP&L's nuclear facilities in early 1993. He is currently the responsible engineer for the A-46/Seismic IPEEE project for the Robinson Steam Electric Plant Unit No. 2 in Hartsville, S.C. He has also participated in the efforts for the Harris and Brunswick plants. He is a graduate of North Carolina State University with a

Bachelor of Science Degree in Civil Engineering. He is currently registered as a professional engineer in both North and South Carolina.

LEO J. BRAGAGNOLO

Mr. Bragagnolo has over ten years of experience in the seismic evaluation of structures and equipment, seismic criteria development, and structural analysis and design. He has participated in and managed projects for industrial, petrochemical, power, Department of Energy (DOE), and nuclear facilities. Most of the projects Mr. Bragagnolo has been involved with concern the seismic evaluation and upgrade of equipment and structures. He has also performed site investigations following the 1987 New Zealand, 1987 Whittier, and 1989 Loma Prieta earthquakes. Mr. Bragagnolo is a Principal Engineer with EQE International and has participated in A-46 and/or IPEEE evaluations for the following plants: CP&L Robinson, CP&L Brunswick, Duke Keowee/Oconee, TVA Sequoyah, TVA Browns Ferry, and Nebraska Public Power Cooper Station. He has completed the SQUG A-46 training course as well as the EPRI seismic individual plant evaluation of external events add-on course. He is a registered Civil Engineer in the state of California.

RONALD W. CUSHING

At EQE, Mr. Cushing is a principal engineer for EQE Engineering Consultants involved in the application of earthquake experience data for component seismic verification at nuclear power plants. Mr. Cushing has investigated sites which have experienced major seismic activity for the SQUG earthquake experience database. He has extensive experience in performing nuclear power plant walkdowns for seismic adequacy in association with A-46 and IPEEE programs. Mr. Cushing is responsible for maintaining a database of replacement parts and components for equipment in nuclear power facilities. He is an author of the industry guidelines for the seismic technical evaluation for replacement items (STERI).

Mr. Cushing has over 17 years experience in nuclear plant walkdowns and startup testing, including valve testing, pump performance and vibration testing, system functional and preoperational testing on such systems as plant cooling water, condensate, main and auxiliary steam, turbine control and lube oil, main and auxiliary feedwater, chemical injection, service gas, and demineralizer systems.

Mr. Cushing is an instructor for the SQUG walkdown screening and seismic evaluation training course. He has also completed the SQUG equipment selection and relay evaluation training course and the EPRI add-on and seismic IPE training course. He is a registered Mechanical Engineer in the State of California.

JAMES R. DISSER

Mr. Disser has over fourteen years of experience in seismic design, analysis and qualification of piping, HVAC, and electrical distribution systems, structures, and mechanical and electrical equipment for nuclear power generation facilities. This includes over eleven years of on-site experience at the Beaver Valley Nuclear Station, the Comanche Peak Steam Electric Station and the Brunswick Nuclear Plant. His experience includes design and analysis, design supervision, project management and walkdown and analytical resolution experience in various Seismic Category II/I, hazards and material condition programs. Mr. Disser is a Project Engineer for EQE Engineering Consultants and completed the SQUG Walkdown Screening and Seismic Evaluation Training Course in preparation for A-46/IPEEE programs. He holds a Bachelor of Science Degree in Civil Engineering from the University of Michigan.

GREGORY S. HARDY

Mr. Hardy has over 18 years experience in the design, analysis and testing of chemical, nuclear and aerospace structures and components. His responsibilities have included probabilistic risk assessments, earthquake experience data-based studies, stress analysis, finite element analysis, seismic margin studies, mass property studies, and shock and vibration environmental testing for hardware qualification.

Mr. Hardy has served as project manager on many projects including multi-million dollar efforts for Comanche Peak Nuclear Power Plant and for the Department of Energy K, L and P reactors at the Savannah River site. He also managed a seismic probabilistic risk assessment for the Chinshan Nuclear Power Plant in Taiwan, a seismic margin study for the Idaho National Engineering Laboratory ICPP facility, seismic research efforts for Mitsubishi Atomic Power in Japan and numerous seismic related projects for the commercial nuclear industry, the Department of

Energy and the oil/gas industry. Mr. Hardy participated in the seismic safety margins research project SSMRP (the original margin research effort for LLNL) and was instrumental in the original development of the equipment fragility methods. He has also been a consultant to the Seismic Qualification Utilities Group(SQUG) for 9 years. He has directed and/or participated in the capacity evaluations of mechanical and electrical components on over 25 Probabilistic Risk Assessments (PRAs) for nuclear power plants. He has played a major role in both the development of the methodology and in the completion of the equipment fragility studies.

DARYL W. HUGHES

Mr. Hughes has over fourteen years of experience associated with structural design, analysis, testing, and construction of nuclear power plant systems, equipment and components. His responsibilities have included seismic qualification of mechanical and electrical equipment and their supporting structures, review and approval of vendor supplied seismic qualification reports, providing seismic requirements for equipment specifications, and evaluating equipment modifications. He has coordinated and directed re-verification efforts of HVAC air handling units, plenums and equipment supports including supervision of personnel, design of hardware modifications, evaluation and resolution of design changes. He has four years of on-site design and construction experience at two nuclear plants. he completed the SQUG walkdown Screening and Seismic Evaluation Training Course and the Add-On Seismic IPE Training Course in preparation for participation in USI A-46 and Seismic IPEEE resolution at CP&L's Harris, Brunswick, and Robinson nuclear power plants. He holds a BS in Mechanical Engineering from the University of Illinois. He is a registered profession engineer in the state of North Carolina.

RONALD L. KNOTT

Mr. Knott has over eleven years of experience associated with the design and construction of nuclear power plants. For the majority of that time, he has been involved with the seismic qualification of equipment. He has reviewed vendor reports and prepared calculations and reports documenting the dynamic analysis and qualification of distribution systems, structures, tanks, valves, and mechanical and electrical equipment for seismic loads. He served as equipment seismic qualification supervisor for the nuclear engineering department at CP&L. He was assigned to the probabilistic risk assessment section and later assigned to the Brunswick Plant for restart following a dual unit shutdown associated with structural deficiencies. In this capacity, he was responsible for the reanalysis of, 250 masonry walls under the IEB 80-11 criteria, plant walkdowns and evaluations for material condition deficiencies, electrical equipment anchorage assessments, HVAC upgrades and instrument rack replacements. He has completed the SQUG and the IPEEE Seismic Add-On courses. He has participated in the development of a resolution approach for CP&L, performed walkdowns and documentation reviews. he holds a BS in Civil Engineering. He is a registered Professional Engineer in North Carolina.

KELLY L. MERZ

Mr. Merz has over 25 years of professional experience in the design, analysis, and testing of structures and mechanical and electrical equipment systems, subjected to dynamic environments for the utility, energy, nuclear power, and defense industries. He has been responsible for the design of system components to resist extreme loadings, including seismic, wind, shock, and transient pressure thermal conditions.

At EQE, Mr. Merz has conducted several A-46 walkdowns of nuclear plant equipment and systems. Relay evaluation studies for three plants have been performed. He has been a major contributor to the first-of-a-kind engineering effort to apply experience-based qualification methods to Advanced Light Water Reactor plant equipment.

Specific recent experience includes: conduct of relay and contact chatter seismic testing in support of the SQUG program; evaluation of nuclear power plant equipment using existing seismic data in support of the SQUG program; conduct of piping component fragility test series; seismic qualification of equipment by analysis and test; tests and studies of nuclear power plant piping, conduit, and raceway systems and design of piping systems in accordance with the ASME code; in-situ testing of reinforced concrete and steel structures; design and shake table testing of HVDC-AC thyrister valve scale model; ambient vibration surveys and assessment of equipment isolation adequacy; development of transmission line load limiter device (patented) for mitigation of conductor break loads on towers; and conduct of full scale conductor break tests.

Past experience has included: U.S. NRC research studies on engineering characterization of earthquake ground motion; application of random vibration theory to determine in-structure response spectra directly from ground response spectra; evaluation of nuclear containment vessels for postulated accident conditions; evaluation of ultimate seismic capacity of masonry wall partitions; conducted major studies of U.S. DOE applied technology program to develop and update seismic design criteria and analysis methods for future liquid metal breeder reactors. Studies included feasibility of nuclear plant isolation, comparison of spectral analysis techniques for piping systems with multi-support input, and guidelines for verification

ROBERT N. PANELLA

Mr. Panella has over twelve years of experience associated with design and construction of nuclear power plants. He has experience in seismic response spectra development, masonry wall analysis, pipe stress analysis, pipe and conduit support design, and seismic qualification of equipment. He has served as department expert in computer assisted design of steel frames and reanalysis of existing structures. He has also provided training to engineering personnel on structural design issues to insure consistency of approach among designers. He completed the SQUG Walkdown Screening and Seismic Evaluation Training Course and the Add-On Seismic IPE Training Course in preparation for participation in USI A-46 and Seismic IPEEE resolution at CP&L's Harris, Brunswick, and Robinson nuclear power plants. He was the project manager for the USI A-46 and Seismic IPEEE programs of all three plants from 1992 through 1993. He holds a BS in Civil Engineering.

KEVIN N. POYTHRESS

Mr. Poythress has over four years of experience in structural design and analysis. He has been working for CP&L in the HESS civil stress subunit for 1-1/2 years performing pipe stress analyses, and has completed Harris Basic Systems Training. He was the lead pipe stress analyst for the replacement of the Brunswick Emergency Diesel Generator Service Water Piping. This MOD replaced a significant amount of piping and therefore required careful attention to schedule, budget, and technical issues. He has performed pipe stress operability and long term evaluation for the RESS Piping Improvement Program. He completed the SQUG Walkdown Screening and Seismic Evaluation Training Course and the Add-On Seismic IPE Training Course in preparation for participation in USI A-46 and Seismic IPEEE resolution at CP&L's Harris, Brunswick, and Robinson nuclear power plants. He has a BS and MS in civil engineering and is a licensed engineer in the state of Tennessee.

THOMAS R. ROCHE

Mr. Roche has over eleven years of experience in the design, engineering, startup and analysis of systems and equipment at power, industrial and nuclear facilities. His responsibilities have included evaluation and analysis of systems and equipment for seismic events, preoperational testing of nuclear power plant systems, system engineer for nuclear and non-nuclear power plant systems, equipment qualification and post earthquake investigations. Mr. Roche is a Technical Manager with EQE International. He is responsible for various seismic evaluation efforts for nuclear facility systems and equipment. Mr. Roche is the Electric Power Research Institute (EPRI) Principal Investigator for investigating the 1989 Loma Prieta, 1994 Northridge and 1995 Great Hanchin earthquakes. He completed the SQUG walkdown and relay evaluation courses as well as the EPRI seismic individual plant evaluation of external events add-on course. He is a registered Mechanical Engineer in the State of California.

Mr. Roche has contributed to the development of the earthquake experience data base generated for the Seismic Qualification Utilities Group (SQUG). He concentrates on the response of systems to earthquakes at power and industrial facilities. Systems are investigated for the effects of power interruption, relay actuations due to vibration, relay actuations due to system transients, spurious electrical and pneumatic signals, and control room alarms. This seismic experience data is being utilized by the nuclear industry to resolve the seismic issues associated with the NRC's Unresolved Safety Issue A-46.

PEER REVIEWER RESUMES

CHARBEL M. ABOU-JAOUDE, P.E.

Mr. Abou-Jaoude is a Project/Service Area Manager in VECTRA's Boston Office, with a broad technical and managerial experience in the power industry. His areas of technical expertise are Structural Mechanics and Seismic Design; he has an in-depth knowledge of various industry codes/standards such as Sections III & XI of the ASME Code, ANSI B31.1, IEEE-344 and 382, various USNRC Reg. Guides and NUREG Reports, WRC Bulletins, AISC, and ACI-349. He is well versed in the Generic Implementation Procedure developed by the Seismic Qualification Utility Group for the resolution of USI-A-46, and the methodologies developed by the industry for the response to Generic Letter 88-20 as outlined in NUREG-1407; he has completed the SQUG/EPRI sponsored A-46 and Seismic IPEEE training courses and has participated in several A-46/IPEEE walkdowns as an SRT member. While at VECTRA, he has lead the engineering efforts of various work scopes; his responsibilities have included: Criteria development, training and personnel development, project execution, interface with regulators and outside organizations, and overall project management.

STEPHEN P. REICHLE

Mr. Reichle has over 20 years of power plant engineering, design, maintenance, and operations experience. As Technical Services Consultant for Mechanical Systems in VECTRA's Boston office he is currently assigned as the Project Manager for the Fire Hazards Analysis (FHA) project for the New York Power Authority. This project consists of updating the FHAs for both the James A. Fitzpatrick and Indian Point 3 nuclear plants. The project also includes the preparation of an analysis that assesses the effects of pipe rupture, inadvertent actuation and manual use of fire protection systems on safety-related equipment at JAF and IP3.

Mr. Reichle is also currently serving as the Systems Project Engineer on the NRC's Unresolved Safety Issue (USI) A-46 projects for: Northeast Utilities (Millstone 1, 2 and Connecticut Yankee), Philadelphia Electric (Peach Bottom and Limerick) and Public Service Electric & Gas (Salem). In this role, he is responsible for the identification of safe shutdown paths and the development of a Success Path Component List for each unit. These NRC programs deal with the seismic adequacy, or margin of equipment in operating plants.

APPENDIX B

H.B. Robinson Unit 2
Individual Plan Examination for External Events

Success Path Logic Diagram
for
Seismic Margins Analysis

1.0 SUCCESS PATH LOGIC DIAGRAM

As a part of the individual plant examination of external events (IPEEE) [1], an evaluation of plant response to a seismic event which exceeds the design basis earthquake is required. A successful response is defined by the ability to maintain plant frontline systems which provide critical plant functions. The functions involved are: reactivity control, reactor coolant system inventory control, reactor coolant system pressure control, and decay heat removal. In addition, the systems which support frontline system operation, e.g., ac power, service water, etc., must be available. The Electric Power Research Institute (EPRI) has developed a process for Seismic Margins Assessment (SMA) documented in EPRI report NP-6041 [2]. This document outlines the steps required to perform the assessment and identifies the boundary conditions and assumptions that are required for this assessment. The major assumptions are summarized below.

- Offsite power is assumed to be lost following the seismic event. The analyst should, however, consider the potential for adverse effects should ac power not be lost or if it should be restored.

- The success paths must be capable of maintaining the plant in either hot or cold shutdown for a period of 72 hours.

- The SMA should address two conditions. The first is a transient without RCS leakage and the second is a 1 inch LOCA condition. For the LOCA case, one reactor coolant system inventory control path must be capable of mitigating a 1 inch LOCA.

- Success is measured at the system level for success path logic diagram elements that represent multiple train systems. In other words, if one train is sufficiently rugged the other trains should provide similar seismic capacity.

- Non-seismic component failures are not explicitly addressed in the analysis. The analyst should provide a check to ensure that the reliability of components will be adequate to exclude random failures. This is especially important for single train systems.

- The potential for relay chatter should be addressed. Note: Relay identification is provided in a separate analysis [9].

- Only core damage prevention systems should be addressed. Containment mitigation systems are not in the scope of the seismic margins evaluation.

The development of a Safe Shutdown Equipment List (SSEL) for SMA was done in conjunction with USI A-46 [3]. All of the equipment required to meet the SMA are included in the USI A-46 analysis.

The NRC in their generic letter [1], indicated that the EPRI methodology is acceptable given two proposed supplements are adopted: 1)non-seismic failures and human actions should be considered in accordance to the guidance provided in NUREG-1407 [4]; and 2)containment isolation and required mitigation systems should be examined as appropriate.

NUREG-1407 [4] indicates that non-seismic failures and human errors should be considered during the selection of systems needed to respond to a seismic event. It further suggests that a method similar to the method provided in NUREG/CR-4826 [5] (Maine Yankee evaluation) is considered acceptable. The Maine Yankee evaluation provides quantitative guidance for determining if non-seismic or human error events should be included.

The EPRI approach adopts a more qualitative criterion which serves as a guide for choosing systems and equipment for the success path logic diagram. In choosing the systems required for the HBR2 success paths, the equipment train reliability is qualitatively considered and only the most reliable alternative is chosen.

With regard to the analysis of the containment, NUREG-1407 [4] states that "the primary purpose of the evaluation for a seismic event is to identify vulnerabilities that involve early failure of containment functions. These include containment integrity, containment isolation, prevention of bypass functions, and some specific systems depending on a containment design (e.g., ignitors, suppression pools, ice baskets)." The major concern presented is for early containment failure modes.

The guidance further 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 (for example, spatial interactions, unique penetration configurations) 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 which is addressed in the relay evaluation.[9] The walkdown confirms that isolation valves are seismically rugged and examines containment systems, for example, containment spray and fan coolers.

In considering this guidance, there appears to be only a marginal benefit to be gained by preparing a listing for containment functions (which is not required by either the EPRI or NRC guidance) and only a brief discussion is provided in Section 3 to assist in guiding the walkdown.

The information provided in the EPRI report outlines each step in the seismic margins evaluation process. This report documents the performance of step 3 of the EPRI process which involves the development of the Success Plant Logic Diagram (SPLD) for H. B. Robinson Unit 2 (HBR2), the development of equipment walkdown lists for systems of interest, which is provided in the USI-A46 analysis [3].

The SPLD is in the format of a reliability block diagram which identifies the systems required for success. The methodology requires that two or more success paths be provided for each of the major functions that must be accomplished to meet the success criteria. It is read in a similar fashion to an electrical diagram. Single entries indicate that the system must function to ensure success. Parallel entries indicate two or more options for success. The following discussion outlines the development of success paths for HBR2 and serve as the basis for the SPLD.

1.1 REQUIRED SUPPORT SYSTEMS

Along with the frontline systems that are assigned to the success paths, the status of support systems necessary to maintain required frontline functions must be identified. To define these systems, a review of the HBR2 IPE system notebooks was performed to identify interfaces between frontline and support systems. The information contained in the IPE system notebooks provides a concise source for identifying support system requirements. In addition to frontline systems, some support systems require cooling or power. Table 1-1 summaries the links between the frontline and support systems addressed in the SPLD.

Using the information presented in Table 1-1, the support systems which must be addressed in the assessment are chosen. As the table demonstrates, AC power is required by most equipment following a seismic event. Since it is assumed that there will be a loss of offsite power, AC power to all systems will be supplied by the emergency diesel generators (EDGs). The fuel oil storage tank contains sufficient fuel for the EDGs to run in support of vital loads for over 72 hours.

Some systems, however, are only needed for selected equipment. For example, HVAC is only required for operation of the diesel generators. Although not required for operation, equipment is identified in the SSEL to provide HVAC cooling to the following rooms: the AFW room, the HHSI/CS room, the RHR room, and the control room. Other components do not require HVAC cooling over the period of interest.

The compressed nitrogen and the instrument air systems provide motive power to both the pressurizer PORVs and steam generator PORVs, both of which are identified in the SSEL. The pressurizer PORVs are equipped with nitrogen accumulators. This quantity of nitrogen is sufficient for PORV operation. If instrument air or backup nitrogen from the steam dump nitrogen accumulators is lost to the steam generator PORVs, procedures direct local operator action to manually dump steam through the main steam line drain valves. The compressed nitrogen and instrument air systems, therefore, are not required or addressed by this evaluation.

Based on this information five support systems were identified as important and are examined by this report. These support systems are:

- Safety-related AC power (including diesel generators and fuel transfer system) - required for AC power following LOOP
- Safety-related DC power - required for equipment control
- HVAC - required for cooling the EDG rooms
- Service water - required for decay heat removal and equipment cooling
- Component Cooling Water (CCW) - required for decay heat removal and equipment cooling

Each of these systems provides an important support function which must be assured following a seismic event. As such, the important components in each support system success path must be included in the development of the Safe-Shutdown Equipment List (SSEL) and evaluated.

Table 1-1

Matrix of Front-Line System Direct Dependencies on Support Systems

DEPENDENCY SYSTEM[2]	Emerg. AC		DC		Inst. Power		SW[1]	CCW[1]	RSAS[3]		IA	HVAC
	E1	E2	A	B	A	B			A	B		
HHSI train A	X		X				X	X1	X			X
HHSI train B		X		X			X	X1		X		X
RHR/LHSI train A	X		X					X1	X			X
RHR/LHSI train B		X		X				X1		X		X
AFW train A	X		X		X		X2		X			X
AFW train B		X		X		X	X2			X		X
AFW steam-driven	X	X	X				X3		X	X		
CVCS	X4	X4						X				
CCW	X4	X4					X					
PORV (456 train)	X5		X						X		X6	
PORV (455C train)		X5		X						X	X6	
SG PORVs			X								X7	
Emergency AC Power							X					X

[1] These cooling water systems have redundant pumps but the headers are cross-tied.

[2] These systems include all functions, for example, injection and recirculation require other systems when isolating the system from the RCS after use.

[3] Reactor Safety Actuation System was evaluated for effects of relay chatter on front-line systems operation under another analysis.

X1 Required only during recirculation (and shutdown cooling for RHR).

X2 Required for cooling and as a backup suction source.

X3 Can be manually changed to a self cooling mode.

X4 Pump A can be connected to dedicated shutdown diesel, pumps B and C are on E1 and E2, respectively

X5 Block valves are powered by ac power and are used in some models for closure of the relief path.

X6 Compressed nitrogen is primary source, IA is backup.

X7 Can manually control steam dump operation through drain valves if instrument air or backup nitrogen is unavailable.

1.2 REACTIVITY CONTROL

Following the convention established by the EPRI document [2], the first block involves the ability of the plant to establish and to maintain adequate shutdown margin following the seismic event. Two paths are identified for this function. The primary path is the insertion of the control rods. This is the normal method for reactor shutdown and occurs automatically when a reactor trip signal is generated. As a backup action the operators can execute a reactor trip from the main control board. Another possibility is that the loss of offsite ac power will result in a loss of power to the control rod motor generator sets. This will result in rod insertion by gravity as the motor control rod drives unlatch and release the control rods. Since the EPRI guidance suggests that the analyst should consider averse effects due to power not being lost, this path is not considered a success and a trip signal must be received to ensure rod insertion. The control rods provide adequate shutdown margin to allow for the control rod of the highest worth to fail to insert.

As the RCS temperature decreases, additional boration may be required to maintain adequate shutdown. This function can be accomplished using the normal charging system. If normal charging is not present the RCS may begin to heat up which will increase the temperature component of negative reactivity and trend the reactor to shutdown. RCS inventory control addresses the need to monitor and maintain long-term shutdown.

Should an inadequate number of control rods be inserted, a backup action can be initiated by the operators to introduce makeup water with a high boron concentration using either of two paths. Inadequate control rod insertion could be the result of control rod binding caused by a shift in reactor internals due to the seismic event or rod misalignment at refueling. The operators can align the charging system to the boration system and provide highly borated water to the RCS to increase shutdown margin using the emergency boration steps of the HBR2 ATWS procedure (FRP-S.1). The operators can also use the RWST source as a means of increasing boron concentration and negative reactivity. The use of the RWST is somewhat slower due to the lower concentration of boron but is adequate to ensure shutdown. The use of emergency boration is chosen as a secondary option due to the need for operator action. The block diagram for reactivity control is shown in Figure B-1.

1.3 REACTOR COOLANT SYSTEM INVENTORY CONTROL

RCS inventory control requires that the operators be able to maintain core coverage for decay heat removal. Inventory may be lost from the RCS from many paths: normal letdown, through the reactor coolant pump seals, and through the pressurizer via the Power Operated Relief Valves (PORVs) or the Safety Relief Valves (SRVs). Makeup is also required due to shrinkage during cooldown.

There are several options available to control the RCS inventory. The normal letdown lines can be isolated to preclude that leakage path but this would complicate other activities such as chemistry control or the addition of boron for long-term shutdown.

At HBR2, the RCS high pressure makeup capability is provided by the charging pumps. The normal makeup function is provided by taking suction from the volume control tank (VCT) to the charging pumps and then discharging through the normal charging paths to the RCS. As an alternative, the RWST can be employed as a suction path when the VCT is unavailable. The use of the charging pumps to support normal makeup is governed by operating procedure OP-301 which addresses the chemical and volume control system and procedure EPP-004 which deals with the response to a reactor trip. For letdown leakage and RCP seal leakage this path is adequate. Normal charging is also available for responding to the 1" LOCA required to be evaluated by the EPRI guidance but by itself is not capable of mitigating the event.

The use of the high head SI pumps provides an adequate response to a 1" LOCA and a secondary means of RCS inventory control. Suction is taken from the RWST. The use of SI pumps to support RCS makeup is governed by procedure EPP-004. To provide a long term source of coolant, suction can be transferred to the containment sump via the RHR system. Procedural guidance for aligning the SI and RHR systems for cold leg recirculation is provided in EPP-009.

Based on procedural guidance (EPP-008, Post-LOCA Cooldown and Depressurization) the operators are directed to cooldown and depressurize the RCS if safety injection is not available and use the RCS accumulators and RHR in low pressure injection (LPI) mode to provide makeup. This requires the availability of secondary-side heat removal and some operator action. The use of this path does provide a means independent of the other paths with the exception of the RWST and the common injection lines. Analyses performed for the IPE using the MAAP code indicate that the accumulators are not required for a 1" LOCA and their failure to inject does not fail successful injection using the RHR system in LPI mode.

In considering the success paths for this function several factors influence the final choice. First, isolation of normal letdown would significantly reduce the need for RCS makeup such that normal charging may not be required. Over 72 hours, however, RCS shrinkage and small leaks within the range of technical specification allowances may result in unacceptable RCS inventory losses and a need for additional makeup. As discussed, additional boron may be required to maintain adequate shutdown margin after many hours of decay heat removal. To increase boron concentration the letdown path is, although not required, desirable. These considerations lead to the conclusion that the isolation of letdown is not a preferred path for RCS inventory control.

For cases without a LOCA the use of normal charging seems the most logical choice since it is the normal method of control and provides reactor coolant pump seal injection. The combination of makeup and letdown provides the operators with a flexible means to control RCS inventory and is a method familiar to the operators. The normal charging path, therefore, is chosen as the primary path for RCS inventory control. Although letdown is not required, its presence would improve the operators ability to respond to any challenges.

The high head SI pumps provide an alternative means of RCS inventory control. This path is capable of providing adequate makeup flow even during a 1" LOCA. The RHR system is also required to provide a long term source of coolant via containment sump recirculation.

A third option is depressurizing the RCS and using the RHR system in LPI mode. This is not a preferred path for RCS inventory control however it does provide additional inventory control independence. Further, the use of this option requires additional systems function (namely that secondary side heat removal will be available) and additional operator actions and monitoring.

Thus, the normal charging pumps are chosen as the primary path, high head SI as the alternative, and depressurization and low head SI as a third option. Figure B-2 summarizes the success paths for this function.

1.4 REACTOR COOLANT SYSTEM PRESSURE CONTROL

The potential exists for a seismically-induced loss of offsite power and for an RCS pressure challenge due to the sudden loss of condenser cooling and an isolation of the power conversion system due to the closure of the main steam isolation valves. This, in turn, temporarily increases RCS temperature and pressure and may require RCS pressure relief and control.

Several potential paths exist for pressure control. Under normal situations the pressurizer sprays can be used to reduce RCS pressure. The spray driving force is provided by the reactor coolant pumps. The assumption that ac power is lost results in a loss of the reactor coolant pumps and normal sprays as a means of pressure control. It is possible to provide auxiliary spray from the normal charging system, but RCS pressure reduction, following a load rejection, would not occur in sufficient time to preclude a pressure relief valve challenge. This option is covered by procedure AOP-019 "Malfunction of RCS Pressure Control".

Two sets of diverse pressure relief valves also provide pressure control. These pressurizer relief valves provide adequate capacity to mitigate any pressure transient as a result of a loss of offsite power. The pressurizer power-operated relief valves (PORVs) are considered the primary means of pressure control with the secondary path involving the pressurizer safety relief valves. Two PORVs are provided with each having sufficient capacity to mitigate the pressure rise associated with a loss of offsite power. The PORVs require dc power and either compressed nitrogen or air to function. Two sources of compressed gas are available, instrument air and nitrogen. Following a loss of power the instrument air system is lost and the containment air supply line is isolated. Accumulators present on the PORV supply lines maintain adequate pressure to allow PORV operation (98 psi). This provides a more than adequate supply of compressed gas for RCS pressure challenges following reactor trip.

Three spring-loaded safety relief valves are present. The safety relief valves are designed to function at a preset pressure and do not require any support system in order to function. By code requirements, the safety relief valves do not have an associated block valve.

The PORVs have the benefit of an ac-powered block valve which may be closed if a PORV fails to reclose. Due to the presence of block valves and that the PORVs will be utilized for other functions, the PORVs are chosen as the preferred path with the safety relief valves being considered as the alternative path. The two success paths for this function are identified in Figure B-3.

1.5 DECAY HEAT REMOVAL

The final function addressed is decay heat removal. Generally, decay heat removal following a reactor trip with loss of offsite power is provided using the steam generators as directed by EPP-005 "Natural Circulation Cooldown." The auxiliary feedwater system provides steam generator makeup and steam is removed using the steam generator safety relief valves or steam line PORVs.

An RCS cooldown using the secondary side requires two elements: a source of feedwater to the steam generators and a means to remove the steam from the steam generators. The mission time for evaluation is 72 hours. Based on the inventory present in the condensate storage tank (CST), the tank will empty prior to meeting 72 hours of decay heat removal. Two options are available to extend cooling for 72 hours. The operators can maintain hot shutdown conditions by aligning the service water header to the suction of the AFW pumps as directed by in the EPP Foldouts. Operating procedure OP-402 covers the need to switch AFW suction to alignment SW. This will provide a source of water which will exceed that required for 72 hours of core cooling but requires manual action. As an option, the operators can perform RCS cooldown and establish cold shutdown using RHR cooling (using the RHR pumps and heat exchangers) prior to CST depletion. The RHR system circulates RCS inventory through the RHR heat exchanger and transfers RCS decay heat to the component cooling water system. Since it is a closed loop arrangement, it does not require additional water sources.

In addition to a supply source, the steam generators must be able to dump steam. Given that the condenser is isolated following the event, steam can be relieved by either the steam generator safety relief valves or by manually and locally opening the main steam before and after seat drain isolation valves. This manual action is directed in procedure AOP-17, Attachment 3 if the steam generator PORVs are not available due to a loss of instrument air and backup nitrogen. The auxiliary feedwater pumps (two motor driven and one steam driven) have sufficient head to inject into the steam generators at the safety relief valve set lift pressure.

Given a lack of secondary-side heat removal, an alternative method is possible. The high head SI pumps and pressurizer PORVs may be used to establish feed-and-bleed cooling. The SI pumps inject cool water into the RCS and the PORVs relieve heated water to the containment which transfers the decay heat from the RCS to the containment. This option is addressed in the Critical Safety Function Status Tree and FRP-H.1.

The inventory used during feed-and-bleed comes from the RWST. Once this inventory is depleted, the operators must establish high pressure recirculation. This involves the alignment of the RHR pumps to the containment sump, establishing CCW flow to the RHR heat exchangers, and swapping the high head SI suction from the RWST to the discharge of the RHR pumps. The RHR heat exchangers provide necessary cooling to maintain RCS cooling. Required operator actions are identified in procedure EPP-009.

For the pressurizer PORVs to function, dc power and a compressed gas supply must be present. Two sources of compressed gas are available, instrument air and nitrogen. Following a loss of power the instrument air system is lost and the nitrogen supply line is isolated. Accumulators present on the PORV supply lines maintain adequate pressure to allow PORV operation for a sufficient period to allow the RHR system to be placed in recirculation mode.

The operators are trained to use the steam generators for the preferred control of decay heat removal. As such, this path is the primary method for heat removal. This involves the use of AFW, steam generators, and the CST for the initial period and then swapping the source to the service water system upon the depletion of the CST. The alternative method for decay heat removal is the use of feed-and-bleed cooling as described by FRP-H.1. This requires the operation of the high head SI pumps and the pressurizer PORVs to remain open. The use of RHR cooling is included as a long-term alternative once cold shutdown conditions are achieved. Figure B4 presents the success logic associated with decay heat removal.

1.6 OVERALL SUCCESS PATH LOGIC DIAGRAM

The individual functions are combined to develop an overall success path logic diagram. Two diagrams (Figures B-5 and B-6) are used to address the two different cases. The first case addresses the transient and assumes that no leakage is present. The second logic diagram is provided for the LOCA case. For both cases, the first issue addresses the status of the support systems. This allows the assumption of all support systems functioning to be made. The next function addresses the ability to control reactivity. This block is equivalent for both cases.

The RCS inventory control block is somewhat different for the two cases. In the transient case success requires that makeup using charging flow rates be provided. The LOCA case requires that additional RCS makeup be provided using either safety injection or low pressure injection. In the case of the transient event, a need for RCS pressure control exists and the RCS pressure control block is included. Following a 1" LOCA, the RCS pressure will decrease and no pressure challenge sufficient to lift the RCS relief valves is expected. The pressure control block is, therefore, not included. Finally, the decay heat removal function required for both cases is the same.

In examining the SPLD it is important to realize that the earlier failure of some paths may preclude success of other functions. As an example, if the SI pumps fail RCS inventory control path the path for feed-and-bleed cooling will also be lost. In other words, the loss of particular paths within functions due to a system failure may result in other paths also being failed for other functions. The converse of this is also true in that the failure of a particular path may not preclude any paths for other functions. As an illustration, if the SI pumps succeed in maintaining RCS inventory following a LOCA, the operators may have successful heat removal by either the AFW system or feed-and-bleed cooling. The success of SI injection for the RCS inventory function does not require that only bleed-and-feed cooling can be used for decay heat removal.

2.0 OPERATIONAL REVIEW OF SUCCESS PATH LOGIC DIAGRAM

The draft SSEL was provided to HBR2 operations personnel to provide a plant review of the success paths and the required equipment. The reviewer examined the paths and components for correctness, applicability to plant procedures, and to identify any other alternatives which might not have been addressed. The SSEL was submitted to the plant for review as part of USI A-46 [3]. Reviewer concurrence is documented in letter File: NF-907.12 Serial: NF-94A-0971.

3.0 SAFE SHUTDOWN EQUIPMENT LISTING

Based on the requirements for maintaining a safe stable state following a seismic event the required frontline and support systems are identified. The equipment required for the system to function is evaluated to ensure a high degree of confidence of a seismic capacity which exceeds the review earthquake value. Although not required, a form similar to that specified by the Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment [6] is utilized for recording the equipment necessary for system function.

For HBR2 a total of 13 systems are included in the search to identify required equipment. Components included in the listing were chosen based on one-line flow diagrams, which were used to identify the available paths, and on guidance contained in EPRI report NP-6041 [2]. NP-6041 suggests that the components be specified by system and that the following information be made available:

- General description (e.g., 6 inch MOV)
- Component identification (tag identifier)
- General building location (reactor auxiliary building)
- Additional comments

This information is provided in the document "Identification of Safe Shutdown Equipment for USI A-46" [3]. In addition to the required information the drawing number, normal state, desired state, and if power is required are also included. The expected type of evaluation is also included. Table 3-1 references the required functions with the tables in the A-46 document that are applicable. In addition Appendix C of the A-46 document contains a composite list of all A-46/IPEEE equipment.

Table 3-1

SSEL Reference to USI A-46 Document [3]

Function	Systems to Support	A-46
Reactivity Control:	RPS/CRD/NIS	Table A-1
	CVCS	Table A-2
RCS Inventory Control:	CVCS	Table A-2
	Pressurizer PORVs	Table A-3.1
	HHSI	Table A-4
RCS Pressure Control:	Pressurizer PORVs	Table A-3.1
	Pressurizer SRVs	Table A-3.2
Decay Heat Removal:	Pressurizer PORVs	Table A-3.1
	RHR	Table A-5
	AFW	Table A-6
	SG PORVs	Table A-7
	SG SRV	Table A-8
Support Systems:	Service Water	Table B-1
	CCW	Table B-2
	AC Power	Table B-3
	DC Power	Table B-4
	HVAC	Table B-5

<u>Nuclear Steam Supply System (NSSS) Components</u>

In addition to the safe shutdown equipment lists, NP-6041 requires that other components be evaluated. This included an evaluation of major NSSS components, or more specifically the component supports, to ensure that component failure will not occur following the seismic margin earthquake. Thus, the following components should be examined and screened if possible:

- Reactor vessel and supports
- Steam generators and supports
- Pressurizer and supports
- Reactor coolant pumps and supports
- Reactor coolant piping

In addition to these components, the reactor internal package and the control rod drive packages must be examined. A comparison of prior plants has been compiled and is documented in Reference 7. This information indicates that an average expected median component capacity would be greater than 1.2 g and that the HCLPF value should exceed the SME acceleration level. Based on the plant vintage and design considerations it is unlikely that these components will provide plausible "weak-links." A screening evaluation to ensure this is, however, necessary.

System Piping

A general rule was applied when piping was considered. Safety-related piping which is adequately anchored has been shown to be very rugged and would not be challenged by an earthquake level postulated for the SMA. A high seismic capacity is anticipated and the piping is expected to be screened from assessment. A verification walkdown will be required to ensure this conclusion which will include examination of instrumentation and other connections to the piping path for potential vulnerabilities. Since a verification walkdown will be necessary, it was deemed more appropriate and efficient to examine piping concerns during the walkdown and not to include each connection in the component listing. Further, a specific listing is not required by the procedure and this approach does not represent a deviation from the methodology. The walkdown will ensure that the preliminary conclusion about the piping strength will be verified.

Containment Isolation and Mitigation Systems

As stated earlier, the containment isolation function is expected to be rugged with respect to seismic events. A detailed listing of containment penetrations is not considered appropriate. As an alternative, the walkdown will examine representative containment penetrations to ensure that the HBR2 design does not have any unique configurations that would lead to conclusions which are contrary to those found in prior studies.

A review of seismic capacities for containments of similar design to HBR2 indicates that the containment structure is expected to have a seismic capacity far above the review level earthquake [8]. Based on this information the major cause for containment isolation failure would be expected to be due to relay failures or spurious operation due to chatter. Low ruggedness relays "bad actors list" are examined separately [9] to ensure that no problems are present.

Containment heat removal is an important aspect in evaluating containment performance. If heat is not adequately removed from the containment, the containment pressure may increase to the containment failure pressure. Two mechanisms can lead to energy being transmitted to the containment. The first is due to the small LOCA. As RCS inventory is lost through the break it carries stored energy, which is then release to the containment and pressurization occurs. Feed-and-bleed cooling also results in energy being transferred to the containment. Containment fan coolers can reduce the pressurization due to these mechanisms.

Based on MAAP analyses performed for the IPE, the containment pressure is not expected to increase to the design limit as long as the RHR heat exchangers are available to remove heat. Thus, the fan coolers represent an additional heat removal mechanism but are not required for successful containment cooling as long as the RHR heat exchangers are present. Failure of this heat removal function will result in containment heat up and pressurization. The pressurization, however, is predicted to occur over many hours and would not result in an early, rapid containment over pressurization. It is concluded that containment fan coolers are not needed to ensure early containment integrity. As a result, containment issues are not addressed in this report and are expected to be addressed through a mixture of the seismic relay review and walkdown.

Seismic Induced Fire and Flood Evaluation

On April 4, 1994 a walk-down was conducted at the H. B. Robinson Unit 2 to specifically look for seismic-induced fire vulnerability issues. A list of areas which contain fire suppression systems and combustible material, including hydrogen, was developed by NUS (the fire analysts) in support of the walk-down.

The walk-down was conducted in accordance with NUREG-1407 requirements. The fire engineers identified all fire water piping and suppression systems. Also identified was any fuel source for a potential fire caused by a seismic event. A summary of the walk-down in presented in Appendix B.7 of Reference 3. Table B-7 of that document lists the equipment identified as needing a seismic fire interaction review.

Random Component Failures and Human Errors

As introduced in Section 1.0, the EPRI methodology [2] provides qualitative guidance for the consideration of random component failures and human errors. The reliability of components identified in the success paths were considered to ensure that only the more reliable systems and components were included. Where more than one system was available to meet a particular function, the most reliable components were chosen. In addition, system alignments which required considerable operator action or were not well documented in procedures were avoided.

Some operator action is required in order to provide identified safety functions. The operator actions identified are proceduralized and the operators receive training as to their implementation. The major actions are identified in Table 3-2 along with procedures which direct the action.

Table 3-2

Major Operator Actions Required in the SPLD

Operator Action	Procedure
Emergency boration	FRP-S.1
Feed-and-bleed cooling	FRP-H.1
RCS cooldown (non-LOCA)	EPP-005
RCS cooldown (LOCA)	EPP-008
Swap over of AFW source	EPP Foldout A
Recirculation swapover	EPP-009
Manual Steam Dump	EPP-1 / AOP-17

Instrumentation and Control

The identification of component control is limited to identifying only those signals which are needed to operate equipment. Since HBR2 is an A-46 plant, NUREG-1407 [3] requires a full scope relay evaluation. Seven hundred and eighty nine relays were identified as essential relays. All seven hundred and eighty nine relays have been accepted by either capacity screening or by system consequence screening. No relays of the low ruggedness type were identified.

Instrumentation identification is limited to that which is required to maintain system function and does not include local indicators which are not required. Two conditions are considered and reflect assumptions used in the HBR2 IPE. Instrumentation is addressed only if it (1) controls active components or (2) if it is required by the operators in order to maintain system operation.

It is believed that this level of investigation meets the intent of the IPEEE for a full scope plant and provided the necessary information required for the walkdown associated with EPRI report NP-6041.

4.0 REFERENCES

1. Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR 50.54(f), Generic Letter No. 88-20, Supplement 4, USNRC, June 28, 1991.

2. Reed, J. W., et al, A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1), Electric Power Research Institute, NP-6041, August 1991.

3. H. B. Robinson Unit 2 Identification of Safe Shutdown Equipment for USI A-46, Final Report, Carolina Power & Light Company, December 1994.

4. Chen, J. T., et al, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, USNRC, NUREG-1407, June 1991.

5. Prassinos, P. G., et al, Seismic Margin Review of the Maine Yankee Atomic Power Station, Vols. 1-3, Lawrence Livermore National Laboratory, NUREG/CR-4826, March 1987.

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7. Campbell, R. D., et al, Compilation of Fragility Information from Available Probabilistic Risk Assessments, Lawrence Livermore National Laboratory, UCID-20571, September 1985.

8. Summitt, R. L., Assessment of Containment Capacity for the ALWR Based on Prior PRA Assessments, USDOE Advanced Reactor Severe Accident Program, April 1990.

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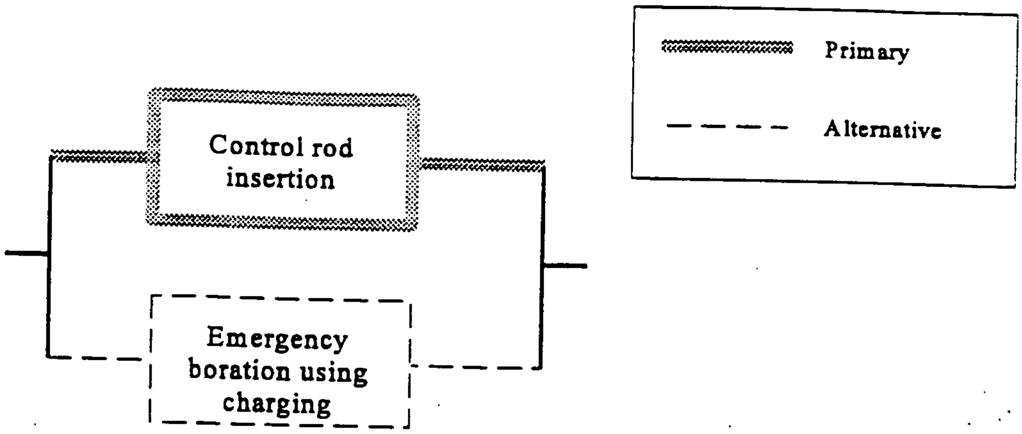

Figure B-1: Success Path Logic Diagram: Reactivity Control Block

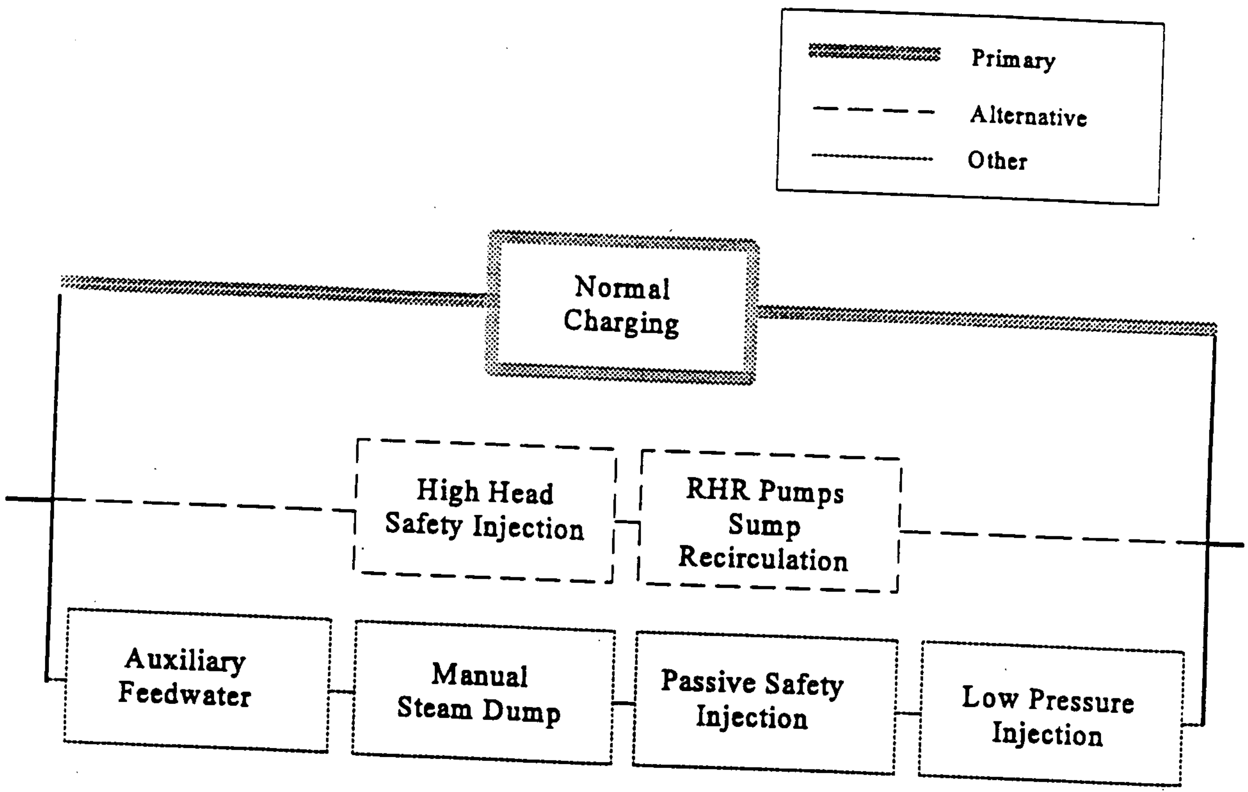

Figure B-2: Success Path Logic Diagram: RCS Inventory Control

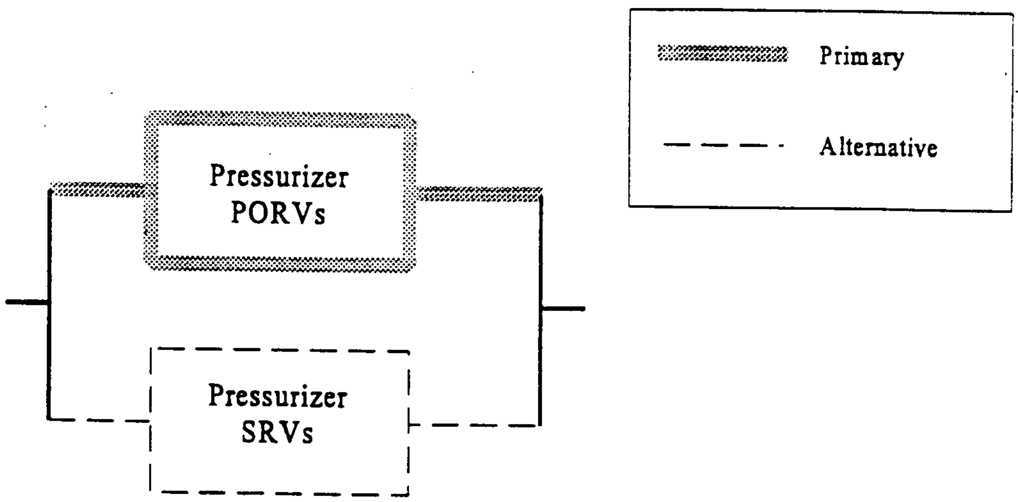

Figure B-3: Success Path Logic Diagram: RCS Pressure Control

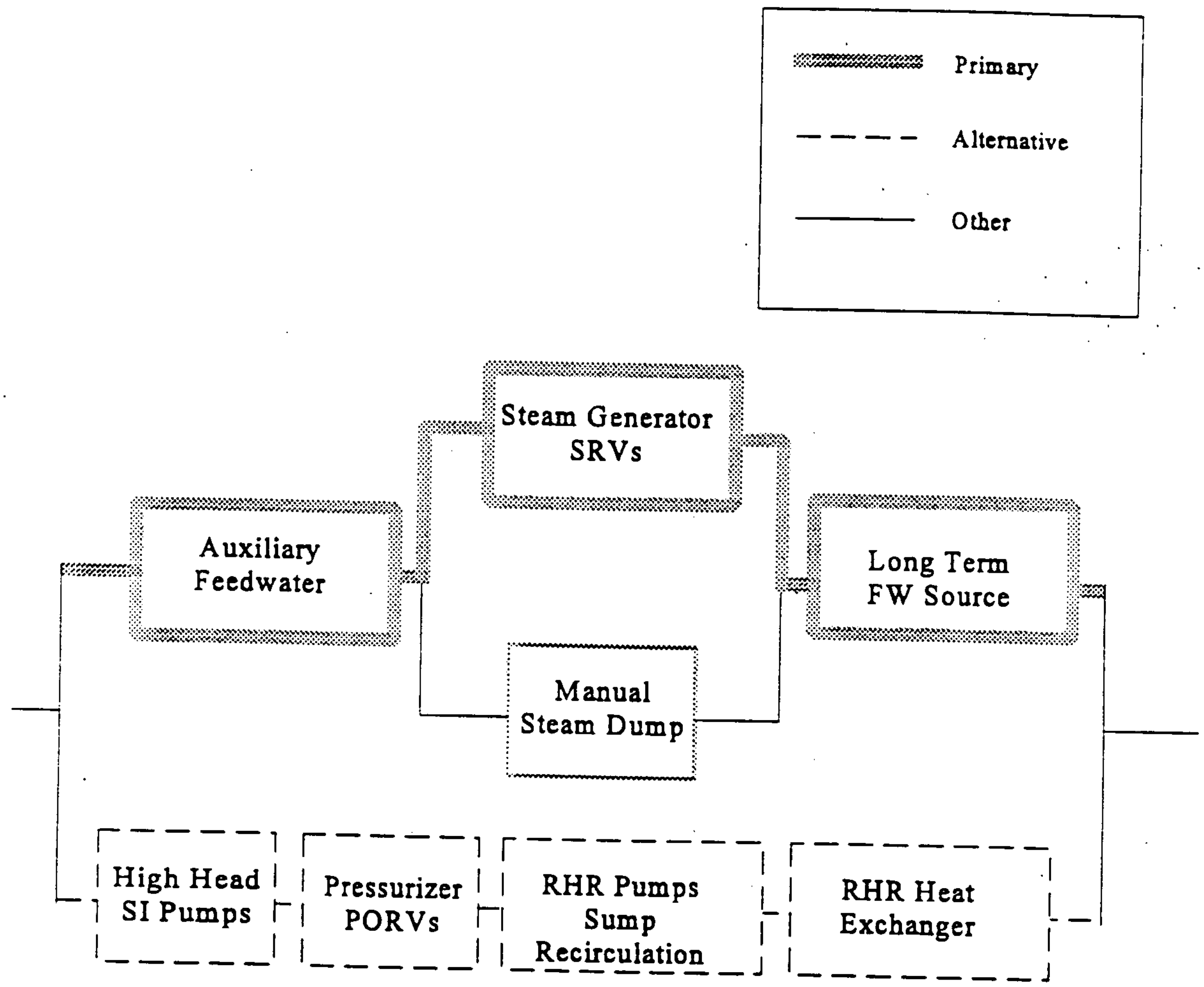

Figure B-4: Success Path Logic Diagram: Decay Heat Removal

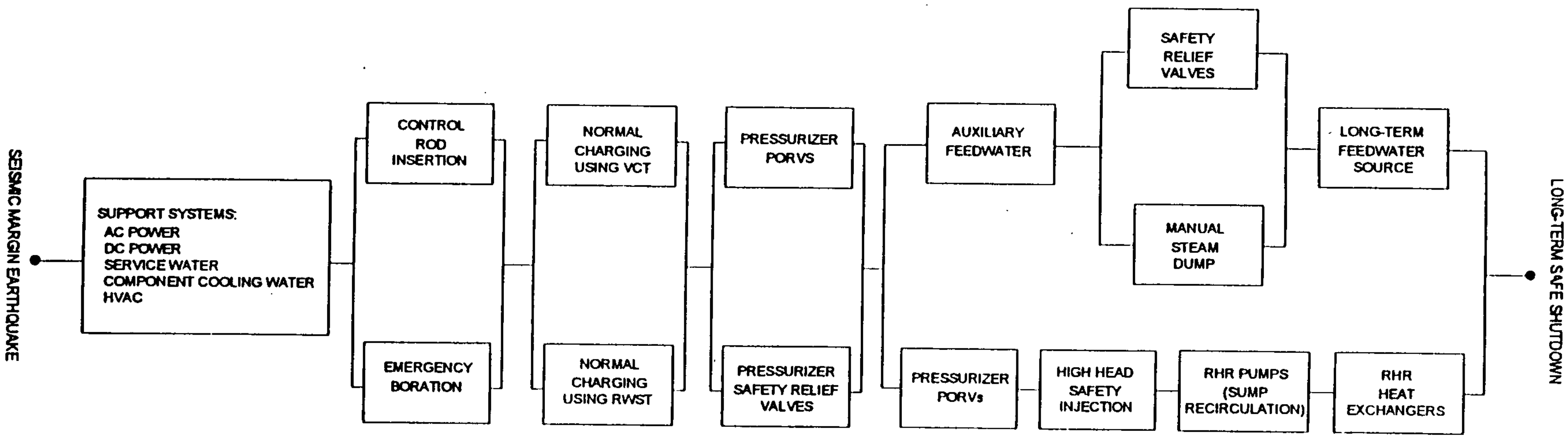

Figure B-5: HBR Seismic Margins Success Path Diagram for Intact Reactor Coolant System Pressure Boundary

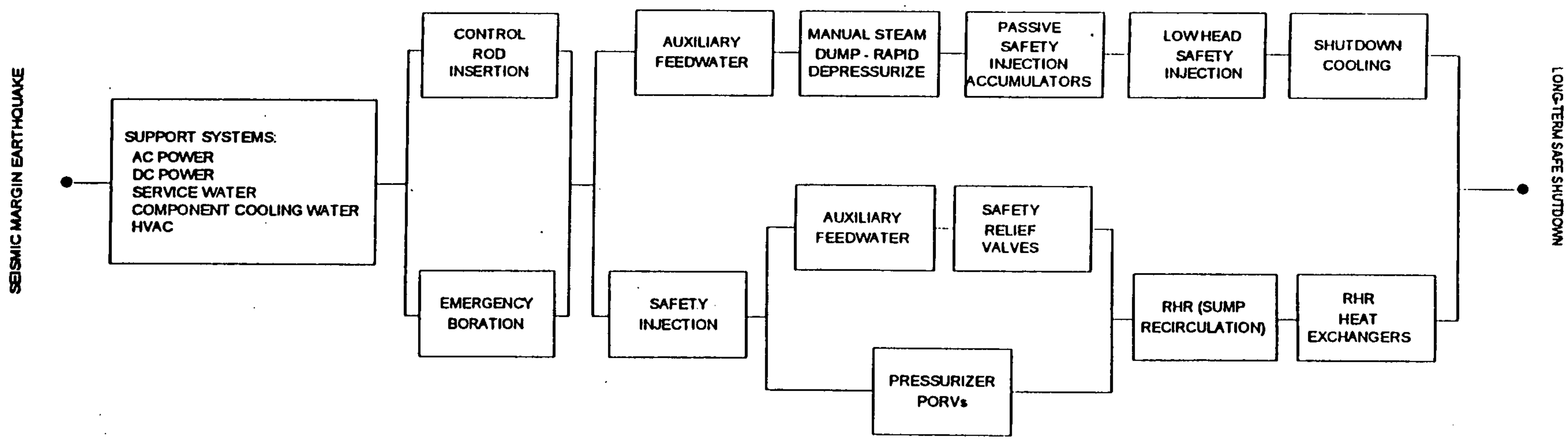

Figure B-6: HBR Seismic Margins Success Path Diagram for Small LOCA

APPENDIX C

Safe Shutdown Equipment List

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CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
00	CR-D7	VENTILATION DAMPER	G-190304(4)(R1)	TB	251	MEZZANINE	HVAC
00	L-19	DAMPER FOR OUTSIDE AIR	G-190304(4)(R1)	TB	251	MEZZANINE	HVAC
00	RC-551A	SAFETY RELIEF VALVE (SRV-1)	5379-1971(1)(R31)	RC	275	CONTAINMENT	SRVs
00	RC-551B	SAFETY RELIEF VALVE (SRV-2)	5379-1971(1)(R31)	RC	275	CONTAINMENT	SRVs
00	RC-551C	SAFETY RELIEF VALVE (SRV-3)	5379-1971(1)(R31)	RC	275	CONTAINMENT	SRVs
00	S6-1A	SW STRAINER A	G-190199(2)(R44)	SW	216	SWP PIT	SW
00	S6-1B	SW STRAINER B	G-190199(2)(R44)	SW	216	SWP PIT	SW
00	SI-857A	RELIEF VALVE	5379-1082(1)(R31)	RAB	226	BIT ROOM	SI
00	SV1-1A	RELIEF VALVE FOR SG A	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-1B	RELIEF VALVE FOR SG B	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-1C	RELIEF VALVE FOR SG C	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-2A	RELIEF VALVE FOR SG A	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-2B	RELIEF VALVE FOR SG B	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-2C	RELIEF VALVE FOR SG C	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-3A	RELIEF VALVE FOR SG A	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-3B	RELIEF VALVE FOR SG B	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-3C	RELIEF VALVE FOR SG C	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-4A	RELIEF VALVE FOR SG A	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-4B	RELIEF VALVE FOR SG B	G-190196(1)(R35)	TB	262	OPERATING	SG
00	SV1-4C	RELIEF VALVE FOR SG C	G-190196(1)(R35)	TB	262	OPERATING	SG
01	DC-MCC-A	125 VDC MCC-A	G-190626(R20)	RAB	248	BATTERY ROOM	DC
01	DC-MCC-B	125 VDC MCC-B	G-190626(R20)	RAB	248	BATTERY ROOM	DC
01	MCC-10	MOTOR CONTROL CENTER	G-190626(R20)	RAB	226	AUX BLDG	AC
01	MCC-16	MOTOR CONTROL CENTER	G-190626(R20)	RAB	246	AUX BLDG	AC
01	MCC-18	MOTOR CONTROL CENTER	G-190626(R20)	RAB	246	CCW PUMP	AC
01	MCC-5,5A	MOTOR CONTROL CENTER	G-190626(R20)	RAB	226	AUX BLDG	AC
01	MCC-6,6A	MOTOR CONTROL CENTER	G-190626(R20)	RAB	246	E1/E2 ROOM	AC
01	MCC-9	MOTOR CONTROL CENTER	G-190626(R20)	RAB	246	E1/E2 ROOM	AC
02	EMER-BUS-E1	480 V EMERGENCY BUS E1	G-190626(R20)	RAB	246	E1/E2 ROOM	AC
02	EMER-BUS-E2	480 V EMERGENCY BUS E2	G-190626(R20)	RAB	246	E1/E2 ROOM	AC
04	TRAN-1	CONSTANT VOLTAGE	G-190626(R20)	RAB	242-6	E1/E2 ROOM	AC
04	TRAN-4	CONSTANT VOLTAGE	G-190626(R20)	RAB	242-6	E1/E2 ROOM	AC
05	AFW	AUX FEEDWATER MOTOR	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
05	AFW	AUX FEEDWATER MOTOR	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW

CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
05	AFW	AUX FEED WATER STEAM	G-190196(1)(R35)	TB	226	AFW SDP	AFW
05	BAT	BORIC ACID TANK TRANSFER	5379-685(3)(R24)	RAB	226	CCW PUMP	CVCS
05	BAT	BORIC ACID TANK TRANSFER	5379-685(3)(R24)	RAB	226	CCW PUMP	CVCS
05	CCW	CCW PUMP A	5379-376(1)(R26)	RAB	226	CCW PUMP	CCW
05	CCW	CCW PUMP B	5379-376(1)(R26)	RAB	226	CCW PUMP	CCW
05	CCW	CCW PUMP C	5379-376(1)(R26)	RAB	226	CCW PUMP	CCW
05	CP-B	CHARGING PUMP B AND COOLER	5379-685(2)(R38)	RAB	226	CHARGING	CVCS
05	CP-C	CHARGING PUMP C AND COOLER	5379-685(2)(R38)	RAB	226	CHARGING	CVCS
05	FOTP-A	FUEL OIL TRANSFER PUMP A	G-190204D(2)(R12)	YARD	N/A	DIESEL OIL	AC
05	SIP-A	SI PUMP A	5379-1082(2)(R32)	RAB	226	SI PUMP ROOM	SI
05	SIP-B	SI PUMP B	5379-1082(2)(R32)	RAB	226	SI PUMP ROOM	SI
05	WST-GAS-COMP-A	WASTE GAS COMPRESSOR A	5379-376(4)(R26)	RAB	246	WASTE GAS	CCW
05	WST-GAS-COMP-B	WASTE GAS COMPRESSOR B	5379-376(4)(R26)	RAB	246	WASTE GAS	CCW
06	FOTP-B	FUEL OIL TRANSFER PUMP B	G-190204D(2)(R12)	YARD	N/A	DIESEL OIL	AC
06	RHRP-A	RHR PUMP A UNIT	5379-1484(R25)	NW OF	203	RHR PUMP PIT	RHR
06	RHRP-B	RHR PUMP B UNIT	5379-1484(R25)	NW OF	203	RHR PUMP PIT	RHR
06	SWPA	SW PUMP A	G-190199(2)(R44)	SW	216	SWP PIT	SW
06	SWPB	SW PUMP B	G-190199(2)(R44)	SW	216	SWP PIT	SW
06	SWPC	SW PUMP C	G-190199(2)(R44)	SW	216	SWP PIT	SW
06	SWPD	SW PUMP D	G-190199(2)(R44)	SW	216	SWP PIT	SW
07	CR-D2A	AO DAMPER FROM OUTSIDE AIR	G-190304(4)(R1)	TB	251	MEZZANINE	HVAC
07	CR-D2B	AO DAMPER FROM OUTSIDE AIR	G-190304(4)(R1)	TB	251	MEZZANINE	HVAC
07	CVC-303A	RCP A SEAL DISCH AOV	5379-685(1)(R35)	RC	251-6	CONTAINMENT	CVCS
07	CVC-303B	RCP B SEAL DISCH AOV	5379-685(1)(R35)	RC	251-6	CONTAINMENT	CVCS
07	CVC-303C	RCP-C SEAL DISCH AOV	5379-685(1)(R35)	RC	251-6	CONTAINMENT	CVCS
07	CVC-310A	LOOP 1 HOT LEG INJECTION AOV	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
07	CVC-310B	LOOP 2 COLD LEG INJECTION	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
07	FCV-1424	FL CUT VALVE MDP-A	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
07	FCV-1425	PISTON OP VALVE MDP-B	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
07	FCV-1608A	SW-A FLOW CONTROL VALVE	G-190199(2)(R44)	SW	216	SWP PIT	SW
07	FCV-1608B	SW-B FLOW CONTROL VALVE	G-190199(2)(R44)	SW	216	SWP PIT	SW
07	FCV-1625A	FLOW CONTROL VALVE SCR A	G-190199(1)(R42)	SW	216	SWP PIT	SW
07	FCV-1625B	FLOW CONTROL VALVE SCR B	G-190199(1)(R42)	SW	216	SWP PIT	SW
07	FCV-1625C	FLOW CONTROL VALVE SCR C	G-190199(1)(R42)	SW	216	SWP PIT	SW

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CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
07	FCV-6416	FL CONTROL VALVE	G-190197(4)(R29)	TB	226	AFW SDP	AFW
07	HCV-121	CHARGING ISOLATION LINE AOV	5379-685(1)(R35)	RAB	226	CHARGING	CVCS
07	PCV-1091	STM DMP N2 ACC INLT PCV AT N2	HBR2-8606(2)(R7)	N2 SHED	YARD	YARD	SG
07	PCV-1093A	STEAM DUMP N2 DISCH PRES	HBR2-8606(2)(R7)	TB	242.5	MEZZANINE	SG
07	PCV-1093B	STEAM DUMP N2 DISCH PRES	HBR2-8606(2)(R7)	TB	242.5	MEZZANINE	SG
07	PCV-1093C	STEAM DUMP N2 DISCH PRES	HBR2-8606(2)(R7)	TB	242.5	MEZZANINE	SG
07	PCV-455C	PORV-1	5379-1971(2)(R11)	RC	275	PZR CUBICAL	PORVs
07	PCV-456	PORV-2	5379-1971(2)(R11)	RC	275	PZR CUBICAL	PORVs
07	PRV-1806	PRESSURE REGULATOR INLET	G-190200(5)(R13)	TB	265	OPERATING	SG
07	PRV-1807	PRESSURE REGULATOR INLET	G-190200(5)(R13)	TB	265	OPERATING	SG
07	PRV-1808	PRESSURE REGULATOR INLET	G-190200(5)(R13)	TB	265	OPERATING	SG
07	RV1-1	SG PORV FOR SG A	G-190196(1)(R35)	TB	265	OPERATING	SG
07	RV1-2	SG PORV FOR SG B	G-190196(1)(R35)	TB	265	OPERATING	SG
07	RV1-3	SG PORV FOR SG C	G-190196(1)(R35)	TB	265	OPERATING	SG
07	SI-856A	AOV RWST/SI RETURN	5379-1082(2)(R32)	RAB	226	SI PUMP ROOM	SI
07	SI-856B	AOV RWST/SI RETURN	5379-1082(2)(R32)	RAB	226	SI PUMP ROOM	SI
07	TCV-1660	TEMP CONTROL VALVE (AOV) DG-A	G-190199(6)(R31)	RAB	226	EDG ROOM A	SW
07	TCV-1661	TEMP CONTROL VALVE (AOV) DG-B	G-190199(6)(R31)	RAB	226	EDG ROOM B	SW
07	TCV-1902A	TEMP CONT VALVE (AOV)	G-190199(10)(R31)	TB	226	AFW SDP	SW
07	TCV-1903A	MDP-A TEMP CNT VALVE (AOV)	G-190199(9)(R36)	RAB	226	AFW MDP ROOM	SW
07	TCV-1903B	MDP-B TEMP CNT VALVE (AOV)	G-190199(9)(R36)	RAB	226	AFW MDP ROOM	SW
07	TCV-659B	AOV TEMPERATURE CONTROL	5379-376(1)(R26)	RAB	226	CHARGING	CCW
07	TCV-659C	AOV TEMPERATURE CONTROL	5379-376(1)(R26)	RAB	226	CHARGING	CCW
08	AFW-V2-14A	MOV COMMON	G-190197(4)(R29)	TB	242-6	SEC CNT PR	AFW
08	AFW-V2-14B	MOV COMMON	G-190197(4)(R29)	TB	242-6	SEC CNT PR	AFW
08	AFW-V2-14C	MOV - COMMON	G-190197(4)(R29)	TB	242-6	SEC CNT PR	AFW
08	AFW-V2-16A	MOV COMMON HEADER MDP	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
08	AFW-V2-16B	MOV MDP-A	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
08	AFW-V2-16C	MOV MDP-B	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
08	AFW-V2-20A	MOV COMMON HEADER	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
08	AFW-V2-20B	MOV COMMON HEADER MDP	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
08	CC-716A	MOV FOR CCW/RCP INLET	5379-376(3)(R20)	RAB	226	PIPE TUNNEL	CCW
08	CC-716B	MOV FOR CCW/RCP INLET	5379-376(3)(R20)	RAB	226	PIPE TUNNEL	CCW
08	CC-735	MOV CCW/THERMAL BARRIER	5379-376(2)(R26)	RAB	226	PIPE TUNNEL	CCW

CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
08	CC-749A	MOV RHR HX A DISCHARGE TO	5379-376(2)(R26)	RAB	226	RHR HX ROOM	CCW
08	CC-749B	MOV RHR HX B OUTLET	5379-376(2)(R26)	RAB	226	RHR HX ROOM	CCW
08	CVC-350	BORIC ACID ISOLATION VALVE	5379-685(2)(R38)	RAB	226	CCW PUMP	CVCS
08	CVC-381	MOV SEAL WATER RETURN LINE	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
08	EV-1702	SOLENOID VALVE TO RVI-1	G-190200(5)(R13)	TB	265	OPERATING	SG
08	EV-1708	SOLENOID VALVE TO RVI-2	G-190200(5)(R13)	TB	265	OPERATING	SG
08	EV-1711	SOLENOID VALVE TO RVI-3	G-190200(5)(R13)	TB	265	OPERATING	SG
08	EV-1963B-1	SOLENOID VALVE	G-190204D(2)(R12)	RAB	226	EDG ROOM B	AC
08	EV-1963B-2	SOLENOID VALVE	G-190204D(2)(R12)	RAB	226	EDG ROOM B	AC
08	FCV-626	CCW RETURN HEADER MOV	5379-376(3)(R20)	RAB	226	PIPE TUNNEL	CCW
08	IA-488	SOLENOID INLET VALVE TO RVI-1	G-190200(5)(R13)	TB	265	OPERATING	SG
08	IA-490	SOLENOID INLET VALVE TO RVI-3	G-190200(5)(R13)	TB	265	OPERATING	SG
08	IA-492	SOLENOID VALVE TO RVI-2	G-190200(5)(R13)	TB	265	OPERATING	SG
08	IA-633	SOLENOID INLET VALVE TO RVI-1	G-190200(5)(R13)	TB	265	OPERATING	SG
08	IA-634	SOLENOID INLET VALVE TO RVI-3	G-190200(5)(R13)	TB	265	OPERATING	SG
08	IA-635	SOLENOID VALVE TO RVI-2	G-190200(5)(R13)	TB	265	OPERATING	SG
08	LCV115C	VCT ISOLATION MOV	5379685(2)D5	RAB	226	CHARGING	CVCS
08	MS-V1-8A	(MOV) 2A MS SUPPLY	G-190196(1)(R35)	TB	226	AFW SDP	AFW
08	MS-V1-8B	(MOV) TR B MS SUPPLY (SDP)	G-190196(1)(R35)	TB	226	AFW SDP	AFW
08	MS-V1-8C	(MOV) TR C MS SUPPLY (SDP)	G-190196(1)(R35)	TB	226	AFW SDP	AFW
08	PCV-3	PRESS CONTROL VALVE ACC A	G-190200(9)(R9)	RC	275	CONTAINMENT	PORVs
08	PCV-4	PRESS CONTROL VALVE ACC B	G-190200(9)(R9)	RC	275	CONTAINMENT	PORVs
08	RC-535	MOV TO PORV 456 (BLOCK	5379-1971(2)(R11)	RC	275	PZR CUBICAL	PORVs
08	RC-536	MOV TO PORV 455C (BLOCK	5379-1971(2)(R11)	RC	275	PZR CUBICAL	PORVs
08	RHR-744A	RHR/SI COLD LEG JUNCTION MOV	5379-1484(R25)	RC	226	CONTAINMENT	RHR
08	RHR-744B	RHR/SI COLD LEG JUNCTION MOV	5379-1484(R25)	RC	226	CONTAINMENT	RHR
08	RHR-750	RHR LOOP 2 HOT LEG ISOLATION	5379-1484(R25)	RC	226	CONTAINMENT	RHR
08	RHR-751	RHR SUCTION LINE MOV	5379-1484(R25)	RC	226	CONTAINMENT	RHR
08	RHR-752A	RHR PUMP INLET - PUMP A	5379-1484(R25)	NW OF	203	RHR PUMP PIT	RHR
08	RHR-752B	RHR PUMP INLET - PUMP B	5379-1484(R25)	NW OF	203	RHR PUMP PIT	RHR
08	RHR-759A	RHR HX OUTLET MOV - PUMP A	5379-1484(R25)	RAB	226	RHR HX ROOM	RHR
08	RHR-759B	RHR HX OUTLET MOV - PUMP B	5379-1484(R25)	RAB	226	RHR HX ROOM	RHR
08	SI-860A	RHR/CONTAINMENT SUMP	5379-1082(5)(R28)	NW OF	203	RHR PUMP PIT	RHR
08	SI-860B	RHR/CONTAINMENT SUMP	5379-1082(5)(R28)	NW OF	203	RHR PUMP PIT	RHR

CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
08	SI-861A	RHR/SI CONT SUMP ISOL MOV	5379-1484(R25)	NW OF	203	RHR PUMP PIT	RHR
08	SI-861B	RHR/SI CONT SUMP ISOL MOV	5379-1484(R25)	NW OF	203	RHR PUMP PIT	RHR
08	SI-862A	RWST/RHR ISOL MOV	5379-1484(R25)	NW OF	203	RHR PUMP PIT	RHR
08	SI-862B	RWST/RHR ISOLATION MOV	5379-1484(R25)	NW OF	203	RHR PUMP PIT	RHR
08	SI-863A	MOV SI/RHR BOUNDARY	5379-1082(2)(R32)	RAB	226	SI PUMP ROOM	RHR
08	SI-863B	MOV SI/RHR BOUNDARY	5379-1082(2)(R32)	RAB	226	SI PUMP ROOM	RHR
08	SI-864A	MOV RWST DSCH LINE	5379-1082(2)(R32)	RAB	226	SI PUMP ROOM	SI
08	SI-864B	MOV RWST DSCH LINE	5379-1082(2)(R32)	RAB	226	SI PUMP ROOM	SI
08	SI-867A	MOV SI/BIT INLET VALVE	5379-1082(1)(R31)	RAB	226	BIT ROOM	SI
08	SI-867B	MOV SI/BIT INLET VALVE	5379-1082(1)(R31)	RAB	226	BIT ROOM	SI
08	SI-870A	MOV BIT OUTLET VALVES	5379-1082(1)(R31)	RAB	226	BIT ROOM	SI
08	SI-870B	MOV BIT OUTLET VALVES	5379-1082(1)(R31)	RAB	226	BIT ROOM	SI
08	SI-878A	MOV SI DSCH PATH FOR PUMPS	5379-1082(2)(R32)	RAB	226	SI PUMP ROOM	SI
08	SV-1	PORV 456 SOLENOID VALVES	G-190200(9)(R9)	RC	275	CONTAINMENT	PORVs
08	SV-2	SOLENOID VALVES TO PORV 455C	G-190200(9)(R9)	RC	275	CONTAINMENT	PORVs
08	SV-3	PORV 456 SOLENOID VALVES	G-190200(9)(R9)	RC	275	CONTAINMENT	PORVs
08	SV-4	SOLENOID VALVES TO PORV 455C	G-190200(9)(R9)	RC	275	CONTAINMENT	PORVs
08	V6-12A	ISOL VALVE TO SOUTH SUPPLY	G-190199(2)(R44)	SW	216	SWP PIT	SW
08	V6-12B	SW DSCH HDR X-CONNECTION	G-190199(2)(R44)	SW	216	SWP PIT	SW
08	V6-12C	SW DSCH HDR X-CONNECTION	G-190199(2)(R44)	SW	216	SWP PIT	SW
08	V6-12D	ISOL VALVE TO NORTH SUPPLY	G-190199(2)(R44)	SW	216	SWP PIT	SW
08	V6-16A	MOV BUTTERFLY TB ISOL	G-190199(10)(R31)	RAB	226	CCW PUMP	SW
08	V6-16B	SW/TB ISOL MOVS	G-190199(10)(R31)	RAB	226	CCW PUMP	SW
08	V6-16C	SW/TB ISOL MOVS	G-190199(10)(R31)	RAB	226	CCW PUMP	SW
09	HVE-17	EXHAUST FAN FOR EDG-B	G-190304(2)(R26)	RAB	226	EDG ROOM B	HVAC
09	HVE-18	EXHAUST FAN EDG-A HVAC	G-190304(2)(R26)	RAB	226	EDG ROOM A	HVAC
09	HVH-6A	SI/CS PUMP RM HVAC	G-190304(2)(R26)	RAB	226	SI PUMP ROOM	HVAC
09	HVH-6B	SI/CS PUMP RM HVAC	G-190304(2)(R26)	RAB	226	SI PUMP ROOM	HVAC
09	HVH-7A	AFW PUMP RM HVAC	G-190304(2)(R26)	RAB	226	AFW MDP ROOM	HVAC
09	HVH-7B	AFW PUMP RM HVAC	G-190304(2)(R26)	RAB	226	AFW MDP ROOM	HVAC
09	HVH-8A	RHR PUMP RM HVAC	G-190304(2)(R26)	NW OF	203	RHR PUMP PIT	HVAC
09	HVH-8B	RHR PUMP RM HVAC	G-190304(2)(R26)	NW OF	203	RHR PUMP PIT	HVAC
09	HVS-5	SUPPLY FAN	G-190304(2)(R26)	RAB	226	EDG ROOM B	HVAC
09	HVS-6	SUPPLY FAN	G-190304(2)(R26)	RAB	226	EDG ROOM B	HVAC

CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
10	EDG A AIR DRYER	EDG A AIR DRYER SW SIDE	G-190199(6)(R31)	RAB	226	EDG ROOM A	SW
10	EDG B AIR DRYER	EDG B AIR DRYER SW SIDE	G-190199(6)(R31)	RAB	226	EDG ROOM B	SW
10	HVH-CR	CNTRL RM AIR HANDLNG UNIT &	G-190304(4)(R1)	RAB	242-6	H&V	HVAC
10	HVS-CR	AIR CLEANING UNIT	G-190304(4)(R1)	TB	251	MEZZANINE	HVAC
10	WCCU-1A	H&V EQUIPMENT ROOM COOLER	G-190199(9)(R36)	RAB	242-5	H&V	SW
10	WCCU-1B	H&V EQUIPMENT ROOM COOLER	G-190199(9)(R36)	RAB	242-5	H&V	SW
14	INSTR-BUS-1	118V INSTRUMENT BUS 1	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
14	INSTR-BUS-2	118V INSTRUMENT BUS 2	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
14	INSTR-BUS-3	118V INSTRUMENT BUS 3	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
14	INSTR-BUS-4	118V INSTRUMENT BUS 4	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
14	INSTR-BUS-6	118V INSTRUMENT BUS 6	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
14	INSTR-BUS-7A	118V INSTRUMENT BUS 7A	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
14	INSTR-BUS-7B	118V INSTRUMENT BUS 7B	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
14	INSTR-BUS-8	118V INSTRUMENT BUS 8	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
14	INSTR-BUS-9A	118V INSTRUMENT BUS 9	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
14	INSTR-BUS-9B	118V INSTRUMENT BUS 9B	G-190626(R20)	RAB	242-6	SAFEGUARDS	AC
15	BATTERY-A	STATION BATTERY "A"	G-190626(R20)	RAB	246	BATTERY ROOM	DC
15	BATTERY-B	STATION BATTERY "B"	G-190626(R20)	RAB	246	BATTERY ROOM	DC
16	BAT-CHGR-A	BATTERY CHARGER "A"	G-190626(R20)	RAB	248	BATTERY ROOM	DC
16	BAT-CHGR-B	BATTERY CHARGER "B"	G-190626(R20)	RAB	248	BATTERY ROOM	DC
16	BAT-CHRG-A1	BATTER CHARGER "A1"	G-190626(R20)	RAB	248	BATTERY ROOM	DC
16	BAT-CHRG-B1	BATTER CHARGER "B1"	G-190626(R20)	RAB	248	BATTERY ROOM	DC
16	INVERTER-A	INVERTER-A	G-190626(R20)	RAB	246	E1/E2 ROOM	AC
16	INVERTER-B	INVERTER-B	G-190626(R20)	RAB	246	E1/E2 ROOM	AC
17	EDG-A	EMERGENCY DIESEL GENERATOR	G-190204A(1)(R18)	RAB	226	EDG ROOM A	AC
17	EDG-B	EMERGENCY DIESEL GENERATOR	G-190204A(1)(R18)	RAB	226	EDG ROOM B	AC
18	A1-E1/2	PRESSURE SWITCH FOR FP/A		TGB	242-6	NEAR HALON	SIESMIC FIRE INTERACTION
18	A1-NCV	PRESSURE SWITCH FOR FP/A		RAB	226	SOUTH PIPE	SIESMIC FIRE INTERACTION
18	A2-CSR	PRESSURE SWITCH FOR FP/B		TGB	242-6	NEAR HALON	SIESMIC FIRE INTERACTION
18	A2-SCV	PRESSURE SWITCH FOR FP/A		RAB	226	SOUTH PIPE	SIESMIC FIRE INTERACTION
18	B1-E1/2	PRESSURE SWITCH FOR FP/B		TGB	242-6	NEAR HALON	SIESMIC FIRE INTERACTION

CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
18	B1-NCV	PRESSURE SWITCH FOR FP/B		RAB	226	SOUTH PIPE	SIESMIC FIRE INTERACTION
18	B2-CSR	PRESSURE SWITCH FOR FP/B		TGB	242-6	NEAR HALON	SIESMIC FIRE INTERACTION
18	B2-SCV	PRESSURE SWITCH FOR FP/B		RAB	226	SOUTH PIPE	SIESMIC FIRE INTERACTION
18	DPS-1608A	DIFF PRESS SWITCH SW-A	G-190199(2)(R44)	SW	216	SWP PIT	SW
18	DPS-1608B	DIFF PRESS SWITCH SW-B	G-190199(2)(R44)	SW	216	SWP PIT	SW
18	ERFIS-MUX-3	ERFIS MULTIPLEXER 3	5379-3503(R13)	RAB	242-6	CABLE SPREAD	CAB
18	FDPS-A1	FIRE DAMPER POWER SUPPLY		RAB	226	OUTSIDE CCW	SIESMIC FIRE INTERACTION
18	FIC-626	FLOW INDICATOR CONROL -	5379-376(2)(R26)	RAB	226	PIPE TUNNEL	CCW
18	FIC-637	FLOW INDICATOR CONTROLLER -	5379-376(4)(R26)	NW OF	203	RHR PUMP PIT	CCW
18	FIC-638	FLOW INDICATOR CONTROLLER -	5379-376(4)(R26)	NW OF	203	RHR PUMP PIT	CCW
18	FIC-657	FLOW INDICATING CONTROL-CSP DISCHARGE	5379-376(4)(R26)	RAB	226	SI PUMP ROOM	CCW
18	FIC-658	FLOW INDICATING CONTROLLER-	5379-376(4)(R26)	RAB	226	SI PUMP ROOM	CCW
18	FT-110	FLOW TRANSMITTER FOR BAT TO CHARGING PUMP	5379-685(2)(R38)	RAB	226	CCW PUMP	CVCS
18	FT-122	CHARGING FLOW TRANSMITTER	5379-685(1)(R35)	RAB	226	CHARGING	CVCS
18	FT-1424	FL TRANS MDP-A	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	FT-1425	FL TRANS MDPB	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	FT-1425A	AFW MDPS TO SG-A FLOW TRANS	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	FT-1425B	AFW MDPS TO SG-B FLOW TRANS	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	FT-1425C	AFW MDPS TO SG-C FLOW TRANS	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	FT-1426A	AFW SDP TO SG-A FLOW TRANS	G-190197(4)(R29)	TB	226	AFW SDP	AFW
18	FT-1426B	AFW SDP TO SG-B FLOW TRANS	G-190197(4)(R29)	TB	226	AFW SDP	AFW
18	FT-1426C	AFW SDP TO SG-C FLOW TRANS	G-190197(4)(R29)	TB	226	AFW SDP	AFW
18	FT-154A	RCP-C SEAL LEAK-OFF HI RANGE FLOW TRANS	5379-685(1)(R35)	RC	251-6	CONTAINMENT	CVCS
18	FT-154B	RCP-C SEAL LEAK-OFF LO RANGE FLOW TRANS	5379-685(1)(R35)	RC	251-6	CONTAINMENT	CVCS
18	FT-155A	RCP-B SEAL LEAK-OFF HI RANGE FLOW TRANS	5379-685(1)(R35)	RC	251-6	CONTAINMENT	CVCS
18	FT-155B	RCP-B SEAL LEAK-OFF LO RANGE FLOW TRANS	5379-685(1)(R35)	RC	251-6	CONTAINMENT	CVCS
18	FT-156A	RCP-A SEAL LEAK-OFF HI RANGE FLOW TANS	5379-685(1)(R35)	RC	251-6	CONTAINMENT	CVCS
18	FT-156B	RCP-A SEAL LEAK-OFF LO RANGE FLOW TRANS	5379-685(1)(R35)	RC	251-6	CONTAINMENT	CVCS
18	FT-613	FLOW TRANSMITTER CCW	5379-376(1)(R26)	RAB	226	CCW PUMP	CCW
18	FT-6416	COMMON FL TRANS	G-190197(4)(R29)	TB	226	AFW SDP	AFW
18	FT-943	SI FLOW TRANSMITTER	5379-1082(1)(R31)	RAB	226	SI PUMP ROOM	SI

CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
18	FY-1425A	AFW MDPS TO SG-A SQUARE	A-190197(1425A)	RAB	226	OUTSIDE AFW	AFW
18	FY-1425B	AFW MDPS TO SG-B SQUARE	A-190301(1425B)	RAB	226	OUTSIDE AFW	AFW
18	FY-1425C	AFW MDPS TO SG-C SQUARE	A-190197(1425C)	RAB	226	OUTSIDE AFW	AFW
18	FY-1426A	AFW SDP TO SG-A SQUARE ROOT EXTRACTOR	B-190301(1426A)	TB	226	AFW SDP	AFW
18	FY-1426B	AFW SDP TO SG-B SQUARE ROOT EXTRACTOR	A-190301(1426B)	TB	226	AFW SDP	AFW
18	FY-1426C	AFW SDP TO SG-C SQUARE ROOT EXTRACTOR	A-190301(1426C)	TB	226	AFW SDP	AFW
18	IR-PT-950	INSTRUMENT RACK FOR PT-950,		RAB	246	RCA ACCESS	PORVs
18	IR-PT-951	INSTRUMENT RACK FOR PT-951,		RAB	246	RCA ACCESS	PORVs
18	LI-614A	CCW SURGE TANK LOCAL LEVEL INDICATOR	5379-3507(R13)	RAB	267	CCW SRG TNK	CCW
18	LIS-1966	DOST LEVEL INDICATOR SWITCH	G-190204D(2)(R12)	YARD	N/A	DIESEL OIL	AC
18	LM-1454A	CST LEVEL SIGNAL ISOLATOR	A-190301(1453)	RAB		AUX PNL DE	AFW
18	LM-1454B	CST LEVEL SIGNAL ISOLATOR	A-190197(1453)	RAB		LM-1454B	AFW
18	LQ-948	POWER SUPPLY	5379-3512(R16)	NORTH	226	RWST	SI
18	LT-106	LEVEL TRANSMITTER BAT-A	5379-685(3)(R24)	RAB	226	CCW PUMP	CVCS
18	LT-108	LEVEL TRANSMITTER BAT-B	5379-685(3)(R24)	RAB	226	CCW PUMP	CVCS
18	LT-115	VOLUME CONTROL TANK LEVEL TRANSMITTER	5379-685(2)(R38)	RAB	246	VOLUME	CVCS
18	LT-1454A	CST LEVEL TRANS	G-190197(1)(R49)	TB	226	SOUTH WEST	AFW
18	LT-1454B	CST LEVEL TRANS	G-190197(1)(R49)	TB	226	SOUTH WEST	AFW
18	LT-614	CCW SURGE TANK LEVEL	5379-376(1)(R26)	RAB	267	CCW SRG TNK	CCW
18	LT-948	RWST LEVEL TRANS	5379-1082(2)(R32)	NORTH	226	RWST	SI
18	LT-969	RWST LEVEL TRANS	5379-1082(2)(R32)	NORTH	226	RWST	SI
18	PC-611	PRESSURE CONTROL - COMMON	5379-376(1)(R26)	RAB	226	CCW PUMP	CCW
18	PI-125B	REMOTE RCP-C TB DIFF PRESS	5379-3478(R13)	RAB	226	SOUTH CABLE	CVCS
18	PI-128B	REMOTE RCP B TB DIFF. PRESS.	5379-3478(R13)	RAB	226	SOUTH CABLE	CVCS
18	PI-131B	REMOTE RCP-A T.B. DIFF. PRESS. INDICATOR	5379-3478(R13)	RAB	226	SOUTH CABLE	CVCS
18	PI-154B	REMOTE RCP-C SEAL DISCH	5379-3475(R15)	RC	226	CONTAINMENT	CVCS
18	PI-155B	REMOTE RCP-B DISCH PRESS	5379-3475(R15)	RC	226	CONTAINMENT	CVCS
18	PI-156B	REMOTE RCP-A SEAL DISCH	5379-3475(R15)	RC	226	CONTAINMENT	CVCS
18	PIC-1393	PIC FOR SDP TURBINE (GOVERNS)	G-190196(1)(R35)	TB	226	AFW SDP	AFW
18	PIC-157	SEAL INJECT FILTER PRES IND	5379-685(1)(R35)	RAB	226	CHARGING	CVCS
18	PIC-477	PRESS INDICAT CONTROLLER	G-190196(1)(R35)	TB	251	MEZZANINE	SG
18	PIC-487	PRESS INDIC CONTLR FOR SG B	G-190196(1)(R35)	TB	251	MEZZANINE	SG
18	PIC-497	PRESS INDIC CONTR FOR SG C	G-190196(1)(R35)	TB	251	MEZZANINE	SG
18	PS-4500A	DG-A PRESSURE SWITCH	G-190204A(2)(R10)	RAB	226	EDG ROOM A	AC

CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
18	PS-4500B	DG-B PRESSURE SWITCH	G-190204A(3)(R10)	RAB	226	EDG ROOM B	AC
18	PS-4509A	OUTLET PRESSURE SWITCH	G-190204A(2)(R10)	RAB	226	EDG ROOM A	AC
18	PS-4509B	OUTLET PRESSURE SWITCH	G-190204A(3)(R10)	RAB	226	EDG ROOM B	AC
18	PS-68A	PRESSURE SWITCH FOR FP/A		RAB	226	OUTSIDE OF	SIESMIC FIRE INTERACTION
18	PS-68B	PRESSURE SWITCH FOR FP/B		RAB	226	OUTSIDE OF	SIESMIC FIRE INTERACTION
18	PS-69A	PRESSURE SWITCH FOR FP/A		RAB	226	OUTSIDE OF	SIESMIC FIRE INTERACTION
18	PS-69B	PRESSURE SWITCH FOR FP/B		RAB	226	OUTSIDE OF	SIESMIC FIRE INTERACTION
18	PSH/L-602A	PRESSURE SWITCH (H/L)	A-190301(602A)	RAB	226	SOUTH CABLE	RHR
18	PSL-1474A-1	MDP-A TRIPS MDP-A	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	PSL-1474A-2	MDP-A TRIPS MDP-A	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	PSL-1474B-1	PR SW TRIP MDP-B	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	PSL-1474B-2	LOW PRESS	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	PSL-1476-1	SDP LINE TRIPS	G-190197(4)(R29)	TB	226	AFW SDP	AFW
18	PSL-1476-2	SDP LINE TRIPS	G-190197(4)(R29)	TB	226	AFW SDP	AFW
18	PT-117	VCT PRESSURE TRANSMITTER	5379-685(2)(R38)	RAB	246	VOLUME	CVCS
18	PT-121	CHARGING PRESSURE	5379-685(1)(R35)	RAB	226	CHARGING	CVCS
18	PT-125	PRESSURE TRANS RCP-C	5379-685(1)(R35)	RC	228	CONTAINMENT	CVCS
18	PT-128	PRESSURE TRANS RCP-B	5379-685(1)(R35)	RC	228	CONTAINMENT	CVCS
18	PT-131	PRESSURE TRANS RCP-A	5379-685(1)(R35)	RC	228	CONTAINMENT	CVCS
18	PT-1421A	MDP COMMON HEADER PRESS TR	G-190197(4)(R29)	RAB	226	AFW MDP ROOM	AFW
18	PT-1421B	COMMON PR TRANS	G-190197(4)(R29)	TB	226	AFW SDP	AFW
18	PT-154	RCP-C SEAL DISCH PRESS TRANS	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
18	PT-155	RCP B SEAL DISCH PRESS TRANS	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
18	PT-156	RCP-A SEAL DISCH PRESS TRANS	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
18	PT-1616	NORTH SW SUPPLY HDR PRESS	G-190199(9)(R36)	RAB	226	AFW MDP ROOM	SW
18	PT-1684	SOUTH SW SUPPLY HDR PRESS	G-190199(10)(R31)	RAB	226	CCW PUMP	SW
18	PT-474	SG A MAIN STEAM LINE PRES	G-190196(1)(R35)	TB	251	MEZZANINE	SG
18	PT-484	SG B MAIN STEAM LINE PRESS	G-190196(1)(R35)	TB	251	MEZZANINE	SG
18	PT-494	SG C MAIN STEAM LINE	G-190196(1)(R35)	TB	251	MEZZANINE	SG
18	PT-602A	PRESSURE TRANSMITTER FOR	5379-1484(R25)	RAB	226	PIPE TUNNEL	RHR
18	PT-934	PRESS TRANS BIT	5379-1082(1)(R31)	RAB	226	BIT ROOM	SI

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CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
18	PT-943	SI/BIT PRESS TRANS	5379-1082(1)(R31)	RAB	226	SI PUMP ROOM	SI
18	TIC-107	TEMPERATURE INDICATOR	5379-685(3)(R24)	RAB	226	CCW PUMP	CVCS
18	TIC-109	TEMPERATURE INDICATOR	5379-685(3)(R24)	RAB	226	CCW PUMP	CVCS
18	TIC-625	TEMP IND CNTLLR - THERMAL	5379-376(2)(R26)	RAB	226	PIPE TUNNEL	CCW
18	TM-410	LOW LEVEL AMPLIFIER	5379-3527(R23)	RAB	226	NORTH CABLE	RPS/CRD/NIS (SCRAM)
18	TM-413	LOW LEVEL AMPLIFIER	5379-3502(R20)	RAB	226	SOUTH CABLE	RPS/CRD/NIS (SCRAM)
18	TM-413B	SIGNAL ISOLATOR	5379-3502(R20)	RAB	226	SOUTH CABLE	RPS/CRD/NIS (SCRAM)
19	TE-116	VOLUME CONTROL TANK TEMP	5379-685(2)(R38)	RAB	246	VOLUME	CVCS
19	TE-123	HX DISCHARGE TEMPERATURE	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
19	TE-126	TEMP ELEMENT RCP-C	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
19	TE-129	TEMP ELEMENT RCP-B	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
19	TE-132	TEMP ELEMENT RCP-A	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
19	TE-133	TEMP ELEMENT CVC-SEAL	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
19	TE-410	LOOP 1 TEMPERATURE ELEMENT	5379-3527(R23)	RC	231	CONTAINMENT	RPS/CRD/NIS (SCRAM)
19	TE-413-1	LOOP 1 TEMPERATURE ELEMENT	5379-3502(R20)	RC	231	CONTAINMENT	RPS/CRD/NIS (SCRAM)
19	TE-413-2	LOOP 1 TEMPERATURE ELEMENT	5379-3502(R20)	RC	231	CONTAINMENT	RPS/CRD/NIS (SCRAM)
19	TE-420	LOOP 2 TEMPERATURE ELEMENT	5379-3527(R23)	RC	231	CONTAINMENT	RPS/CRD/NIS (SCRAM)
19	TE-423	LOOP 2 TEMPERATURE ELEMENT	5379-3502(R20)	RC	231	CONTAINMENT	RPS/CRD/NIS (SCRAM)
19	TE-430	LOOP 3 TEMPERATURE ELEMENT	5379-3527(R23)	RC	231	CONTAINMENT	RPS/CRD/NIS (SCRAM)
19	TE-433	LOOP 3 TEMPERATURE ELEMENT	5379-3502(R20)	RC	231	CONTAINMENT	RPS/CRD/NIS (SCRAM)
19	TE-463	PORV DISCH TEMP ELEMENT	5379-1971(2)(R32)	RC	275	PZR CUBICAL	PORVs
19	TE-465	TEMPERATURE ELEMENT SRV-3	5379-1971(1)(R31)	RC	275	PZR CUBICAL	SRVs
19	TE-467	TEMPERATURE ELEMENT SRV-2	5379-1971(1)(R31)	RC	275	PZR CUBICAL	SRVs
19	TE-469	TEMPERATURE ELEMENT SRV-1	5379-1971(1)(R31)	RC	275	PZR CUBICAL	SRVs
19	TE-607	TEMPERATURE ELEMENT CCW	5379-376(1)(R26)	RAB	226	CCW PUMP	CCW
20	A-65V	PANEL A-65V: PSL-1616 & PSL-1684	A-190301(1616)(R0)	RAB	242-6	CABLE SPREAD	CAB
20	AUX-RLY-RKS A-F	AUX RELAY RKS: A-F (RELAYS		RAB	242.5	CABLE SPREAD	CAB

CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
20	AUX-RLY-RKS G-M	AUX RELAY RKS: G-M (NO I)		RAB	242-6	CABLE SPREAD	CAB
20	BOX-208	RELAY BOX		RAB	246	HVS FAN ROOM	CAB
20	BOX-209	RELAY BOX		RAB	246	HVS FAN ROOM	CAB
20	CET	CET PANEL INCLUDES TM-577 &	5379-5302	RAB	249-6	ROD DRIVE	RPS/CRD/NIS (SCRAM)
20	EAST	HAGAN RACKS 1-13,26	5379-02045-77	RAB	254	HAGAN ROOM	CAB
20	EDG-A-480V-PNL	DG-A MOTOR 480V POWER BOX A		RAB	226	EDG ROOM A	AC
20	EDG-A-CON-SWTCHBRD A	DGA-CONTROL SWITCHBOARD A	WALK	RAB	226	EDG ROOM A	AC
20	EDG-A-CT-CUB	DG-A VOLTAGE REGULATOR	5379-2103(R0)	RAB	226	EDG ROOM A	AC
20	EDG-A-DIESEL-CP	DG-A MOTOR CONTROL PANEL	WALK	RAB	226	EDG ROOM A	AC
20	EDG-A-JBOX	DG-A EXPANSION TANK		RAB	226	EDG ROOM A	AC
20	EDG-B-480V-PNL	DG-B MOTOR 480V POWER BOX B		RAB	226	EDG ROOM B	AC
20	EDG-B-CON-SWTCHBRD B	DGB-CONTROL SWITCHBOARD B	WALK	RAB	226	EDG ROOM B	AC
20	EDG-B-CT-CUB	DG-B VOLTAGE REGULATOR	5379-2103(R0)	RAB	226	EDG ROOM B	AC
20	EDG-B-DIESEL-CP	DG-B MOTOR CONTROL PANEL	WALK	RAB	226	EDG ROOM B	AC
20	EDG-B-JBOX	DG-B EXPANSION TANK		RAB	226	EDG ROOM B	AC
20	ERFIS	ERFIS MULTIPLEXER 2 Q-LIST	5379-3507(R13)	RAB	242.5	CABLE SPREAD	CCW
20	FDAP-A1	FIRE DETECTOR ACTUATION		RAB	226	OUTSIDE CCW	SIESMIC FIRE INTERACTION
20	FDAP-A2	FIRE DETECTOR ACTUATION		RAB	246	E1/E2 ROOM	SIESMIC FIRE INTERACTION
20	FDAP-B1	FIRE DETECTOR ACTUATION		RAB	226	OUTSIDE CCW	SIESMIC FIRE INTERACTION
20	FDAP-B2	FIRE DETECTOR ACTUATION		RAB	246	E1/E2 ROOM	SIESMIC FIRE INTERACTION
20	FDPS-B1	FIRE DAMPER POWER SUPPLY		RAB	226	OUTSIDE CCW	SIESMIC FIRE INTERACTION
20	ICCM-86I	INADEQUATE CORE COOLING		RAB	249-6	ROD DRIVE	CAB
20	ICCM-86II	INADEQUATE CORE COOLING		RAB	249-6	ROD DRIVE	CAB
20	IR-1B	INSTRUMENT RACK: PT-131,156, PI-156B & LT-474		RC	226	CONTAINMENT	CVCS
20	IR-2B	INSTRUMENT RACK: PT-128,155, PI-155B & LT-484		RC	226	CONTAINMENT	CVCS
20	IR-3B	INSTRUMENT RACK: PT-125,154, PI-154B & LT-494		RC	226	CONTAINMENT	CVCS
20	LI-477A	SG-A LEVEL INDICATOR	5379-3578(R0)	TB	242.5	SECONDARY	AFW
20	LI-477B	SG-A LEVEL INDI AFWP RM PNL	5379-3578(R0)	RAB	226	AFW MDP ROOM	AFW
20	LI-487A	SG-B LEVEL INDICATOR	5379-3516(R17)	TB	242.5	SECONDARY	AFW

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CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
20	LI-487B	SG-B LEVEL INDIC AFWP ROOM	5379-3516(R17)	RAB	226	AFW MDP ROOM	AFW
20	LI-497A	SG-C LEVEL INDI MEZZANINE	5379-3517(R20)	TB	242.5	SECONDARY	AFW
20	LI-497B	SG-C LEVEL INDICATOR AFW	5379-3517(R20)	RAB	226	AFW MDP ROOM	AFW
20	NUC-INST-PROTECT CA	NUCLEAR INSTRUMENTATION		RAB	254	CONTROL ROOM	RPS/CRD/NIS (SCRAM)
20	PAM I	POST ACCIDENT MONITOR PANEL		RAB	254	CONTROL ROOM	CAB
20	PAMII	POST ACCIDENT MONITOR PANEL		RAB	254	CONTROL ROOM	CAB
20	PXMTR-1	PRESS CAB: LT-459, PT-455	5379-0169(R0)	RC	226	CONTAINMENT	CVCS
20	PXMTR-2	PRESS CAB: LT-460, PT-456	5379-0169(R0)	RC	226	CONTAINMENT	CVCS
20	PXMTR-3	PRESS CAB: LT-461, PT-457	5379-0169(R0)	RC	226	CONTAINMENT	CVCS
20	PXMTR-4	PRESS CAB: LT-462, PT-444,445,500	5379-0169(R0)	RC	226	CONTAINMENT	CVCS
20	RACK-29	HAGAN RACK 29	HBR2-11267	RAB	254	HAGAN ROOM	CAB
20	RACK-30	HAGAN RACK 30	HBR2-11268	RAB	254	HAGAN ROOM	CAB
20	RACK-50	MISC. RELAY RACK 50 - RELAYS		RAB	242-6	SAFEGUARDS	CAB
20	RACK-51,52	SAFEGUARDS RACK 51 - RELAYS	5379-03237	RAB	242-6	SAFEGUARDS	CAB
20	RACK-53-57	RPS RACK 53 - 57 RELAYS	5379-03134	RAB	242-6	SAFEGUARDS	CAB
20	RACK-58-62	RPS RACK 58 - 62 REALYS	5379-03134	RAB	242-6	SAFEGUARDS	CAB
20	RACK-63,64	SAFEGUARDS RACK 63 & 64		RAB	242-6	SAFEGUARDS	CAB
20	REACTOR-BRK-CAB	CABINET FOR REACTOR		RAB	226	SOUTH CABLE	RPS/CRD/NIS (SCRAM)
20	RMS-CONSOLE	RADIATION MONITORING SYSTEM		RAB	254	CONTROL ROOM	CAB
20	RPI-RACK-1	ROD POSITION TRANSLATING	B-190628(78)(R12)	RAB	242-6	CABLE SPREAD	RPS/CRD/NIS (SCRAM)
20	RPI-RACK-2	ROD POSITION TRANSLATING	B-190628(79)(R12)	RAB	242-6	CABLE SPREAD	RPS/CRD/NIS (SCRAM)
20	RPI-RACK-3	ROD POSITION TRANSLATING	B-190628(80)(R10)	RAB	242-6	CABLE SPREAD	RPS/CRD/NIS (SCRAM)
20	RPI-RACK-4	ROD POSITION TRANSLATING	B-190628(81)(R9)	RAB	242-6	CABLE SPREAD	RPS/CRD/NIS (SCRAM)
20	RTGB	RTGB		RAB	254	CONTROL ROOM	CAB
20	RVLIS	RVLIS INSTRUMENTATION RACK		RAB	249-6	ROD DROVE	CAB
20	TM-410B	SIGNAL ISOLATOR	5379-3527(R23)	RAB	226	NORTH CABLE	RPS/CRD/NIS (SCRAM)
20	TM-577	SIGNAL PROCESSOR CABINET CH	5379-3502(R20)	RAB	254	CET PANEL	RPS/CRD/NIS (SCRAM)
20	TM-578	SIGNAL PROCESSOR CABINET CH	5379-3502(R20)	RAB	254	CET PANEL	RPS/CRD/NIS (SCRAM)

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CAT	EQUIP ID	DESCRIPTION	DWG_NO	BLDG	FL_EL	ROOM	SYSTEM
20	WEST	HAGAN RACKS 14-25,27,28	5379-02045-77	RAB	254	HAGAN ROOM	CAB
21	ART-A	AIR RECEIVER A TANK	G-190204A(1)(R18)	RAB	226	EDG ROOM A	AC
21	ART-B	AIR RECEIVER B TANK	G-190204A(1)(R18)	RAB	226	EDG ROOM B	AC
21	BAT-A	BORIC ACID TANK A	5379-685(3)(R24)	RAB	226	CCW PUMP	CVCS
21	BAT-B	BORIC ACID TANK B	5379-685(3)(R24)	RAB	226	CCW PUMP	CVCS
21	CCW	CCW HX A	5379-376(1)(R26)	RAB	226	CCW PUMP	CCW
21	CCW	CCW HX B	5379-376(1)(R26)	RAB	226	CCW PUMP	CCW
21	CCW	COMPONENT COOLING WATER	5379-376(1)(R26)	RAB	267	CCW SRG TNK	CCW
21	CST	CONDESATE STORAGE TANK	G-190197(1)(R49)	TB	226	SOUTH WEST	AFW
21	CVC	REGENERATIVE HX	5379-685(1)(R35)	RC	226	CONTAINMENT	CVCS
21	DAY	FUEL OIL DAY TANK A	G-190204D(2)(R12)	RAB	226	EDG ROOM A	AC
21	DAY	FUEL OIL DAY TANK B	G-190204D(2)(R12)	RAB	226	EDG ROOM B	AC
21	DOST	DIESEL OIL STORAGE TANK	G-190204D(2)(R12)	YARD	N/A	DIESEL OIL	AC
21	EXCS-LTDWN-HX	EXCESS LETDOWN HEAT EXCH-	5379-376(3)(R20)	RC	228	CONTAINMENT	CCW
21	EXPANSION TNK-A	JACKET WATER EXPANSION	G-190204A(1)(R18)	RAB	226	EDG ROOM A	AC
21	EXPANSION TNK-B	JACKET WATER EXPANSION	G-190204A(1)(R18)	RAB	226	EDG ROOM B	AC
21	N2-ACC-A	N2 ACCUMULATOR A	G-190200(9)(R9)	RC	275	PZR CUBICLE	PORVs
21	N2-ACC-B	N2 ACCUMULATOR B	G-190200(9)(R9)	RC	275	PZR CUBICLE	PORVs
21	NON-REG-HX-CCW	NON-REGEN. HX CCW SHELL SIDE	5379-376(2)(R26)	RAB	226	NON-REGEN HX ROOM	CCW
21	PRZ	PRESSURIZER STEAM SAMPLE HX	5379-376(1)(R26)	RAB	226	SAMPLE ROOM	CCW
21	PZR	PRESSURIZER LIQUID SAMPLE HX	5379-376(1)(R26)	RAB	226	SAMPLE ROOM	CCW
21	RHR	RHR HEAT EXCHANGER A	5379-1484(R25)	RAB	226	RHR HX ROOM	RHR
21	RHR	RHR HEAT EXCHANGER B	5379-1484(R25)	RAB	226	RHR HX ROOM	RHR
21	RWST	RWST TANK W/VENT	5379-1082(2)(R32)	NORTH	226	RWST	SI
21	RX	REACTOR COOLANT SAMPLE HX	5379-376(1)(R26)	RAB	226	SAMPLE ROOM	CCW
21	SG	SG BLOWDOWN SAMPLE HX A	5379-376(1)(R26)	RAB	226	SAMPLE ROOM	CCW
21	SG	SG BLOWDOWN SAMPLE HX B	5379-376(1)(R26)	RAB	226	SAMPLE ROOM	CCW
21	SG	SG BLOWDOWN SAMPLE HX C	5379-376(1)(R26)	RAB	226	SAMPLE ROOM	CCW
21	SPENT	SPENT FUEL PIT HEAT	5379-376(4)(R26)	FHB	226	NEXT TO SI	CCW
21	STM	STEAM DUMP N2 ACCUMULATOR	HBR2-8606(2)(R7)	TB	242.5	MEZZANINE	SG
21	SW HX	SEAL WATER HX	5379-685(2)(R38)	RAB	226	SEAL WATER HX ROOM	CVCS
21	VCT	VOLUME CONTROL TANK	5379-685(3)(R24)	RAB	246	VOLUME	CVCS

APPENDIX D

Peer Review Report

VECTRA

June 23, 1995
0132-00175.000-001

Carolina Power & Light Company
Nuclear Engineering Department
One Hanover Square, 8th Floor
Raleigh, NC 27602-1551

Attention: Mr. Ronald L. Knott
Principal Civil Engineer

Subject: Carolina Power & Light Company
Robinson Nuclear Plant - SQUG A-46/IPEEE Peer Reviews

Dear Mr. Knott:

This letter is intended to document our detailed comments and recommendations from the review of the Robinson A-46 and IPEEE program.

Attachment A contains the observations related to the seismic evaluations: these have been communicated to you during my visit to your offices earlier in May.

Attachment B consolidates comments for the Success Path Development, and the Relay Report. Detailed comments had been forwarded to Mr. Bostian in late April and were also discussed during my visit in early May.

As discussed previously, the A-46 and IPEEE efforts for the Robinson Plant were found to have been conducted in a very thorough and competent manner. The Peer reviewers found that the programs are being performed in accordance to the guidance of the SQUG GIP and EPRI NP-6041, in addition the seismic reviews met the stated objectives of NUREG-1407. The results and findings from the program appear to be reasonable and are consistent with expectations for a plant of this vintage. A number of voluntary upgrades to equipment were noted during the plant walk-through which have resulted in improved seismic ruggedness; in addition a number of the outliers that were noted by the SRT were in the process of being upgraded during the outage indicating good initiative and responsiveness to seismic issues.

VECTRA Technologies, Inc. · 600 Worcester Road · Framingham, MA 01701-5360 · Tel: (508) 370-3266 · Fax: (508) 370-3223

VECTRA

June 23, 1995
0132-00175.000-001

If you have any questions or comments, please do not hesitate to contact me at (508) 370-3391.

Very truly yours,

C. M. Abou Jaoude

Charbel M. Abou-Jaoude, P.E.
Project Manager

Attachments

cc: Mr. M.F. Page
Mr. S.R. Bostian
Mr. M.D. Engelman

Seismic Peer Review of the HBR2 A-46/IPEEE Program

This attachment provides a summary of the seismic reviews for the A-46 / IPEEE program at the Robinson Nuclear Plant.

A two day plant visit of all accessible areas, excluding containment and high radiation or dress-out areas was conducted. SEWS for each of the equipment classes and data packages were sampled subsequent to the walkdowns to compare field notes with the SRT recorded observations and conclusions; a brief review of a number of back-up evaluations and anchorage analyses was performed at the CP&L offices. In addition the SMA spectra report and drafts of the IPEEE and A-46 reports and other HCLPF calculations were subsequently transmitted for peer evaluation.

The seismic effort for the two programs has been well coordinated, the findings appear to be consistent with the observations and expectations of the peer reviewer. The vast majority of the conditions that were identified by the reviewer, as requiring further evaluations or upgrades, had been previously noted by the SRT. The completed evaluations were thorough and well documented. During the plant visit a number of equipment anchorage upgrades that had been implemented in the late 80's were noted, also some of the outlier dispositions were already initiated and being implemented.

Based on the conducted reviews, it is evident that the A-46 and IPEEE efforts at Robinson were conducted in a very thorough and competent manner. The completed evaluations follow the guidance of the SQUG GIP for A-46 and the EPRI NP-6041 report and NUREG-1407 for IPEEE. The SRT's have exercised appropriate judgments and the overall conclusions are reasonable.

The following is a listing of the areas and equipment that were covered during the plant visit; in addition a brief discussion of observations or comments is provided.

PLANT VISIT

The walk through of representative components in the accessible areas of the power block provided a good sampling of various equipment types, distributed systems, housekeeping practice, and II/I considerations. The areas and items reviewed included:

Turbine Building (elev. 226', 242', and 262')

- Turbine Driven Aux Feed Pump and associated Controllers and Valves
- Miscellaneous Main Steam Valves and Instrument Racks

Reactor Building

- Motor Driven Aux Feed Pumps, Safety Injection Pumps, Charging Pumps, and associated control and isolation valves, local instruments and room coolers
- CCW pumps and Heat Exchangers
- Boric Acid Storage Tank
- Miscellaneous Relay Racks, Distribution Panels, and Cabinets in the Cable Spreading Safeguards Room, Control Room, and Hagan Room
- Batteries, Chargers/Inverters, MCC's and Emergency Buses.
- Diesel Generators

Yard and Service Water Intake

- Service Water Pumps and Valves
- Large Flat Bottom Storage Tanks

OBSERVATIONS / COMMENTS:

The vast majority of conditions that were noted during the plant visit had previously been identified by the SRT. The following are the additional observations from the peer reviewer:

- Jacket Water and Lube Oil Heat Exchangers do not have the axial restraining shipping cables installed. It is recommended to install the cables or evaluate the load path for the stacked configuration.

- Non-Safety batteries and chargers adjacent to the diesel generator skid may cause a II/I interaction with a newly installed transmitter on the diesel skid. The safety function of the transmitter needs to be reviewed.

- The existing conservative spectra does not envelop the SSRAP reference spectrum (1.5 x Bounding Spectrum) for elevations 242' and above of the turbine building. The SEWS for corresponding components needed to be revised to use the 40' rule for comparison of Capacity vs. Demand.

Peer Review of the HBR2 SSEL and Relay Review for A-46/IPEEE

REVIEW OF THE H.B. ROBINSON NUCLEAR PLANT A-46 PROGRAM

The following documents were reviewed by Mr. Steve Reichle for the purposes of examining the H.B. Robinson Unit 2 (HBR2) A-46 Program:

- H.B. Robinson Unit 2, "Identification of Safe Shutdown Equipment" Final Report, dated December 1994.
- H.B. Robinson Nuclear Power Plant, "Relay Evaluation Report for Carolina Power and Light Co.", dated April 28, 1995.
- CP&L Drawings (flow diagrams):

G-190196, Sh. 1	5379-1082, Sh. 4
G-190197, Sh. 1	5370-1484, Sh. 1
G-190197, Sh. 4	

The peer review of the safe shutdown equipment selection and relay review work completed for the HBR2 was performed against the guidance provided in the SQUG Generic Implementation Procedure (GIP) and EPRI NP-6041. The methodology utilized to select and document the safe shutdown paths and equipment selection, as documented in the safe shutdown final report, fully meets the intent of the GIP and EPRI methodology.

In addition to reviewing the above reports against the referenced guidance documents, a detailed check of the AFW and RHR systems and component selection process was performed with their respective flow diagrams. This review was made to determine if all applicable components were identified, and whether the correct review types (i.e. seismic and/or relay) were specified.

As a result of these reviews, several observations and comments were made by VECTRA and were provided to CP&L in April of 1995. CP&L reviewed each specific comment provided by VECTRA and provided a detailed response to each comment. These comments and resolutions are documented in an internal correspondence. Based on follow-up conversations and CP&L's detailed comment resolutions, all action items which were identified as a result of the peer review have been addressed.

The comments made by the VECTRA Peer Reviewer were mostly questions presented to the preparer of the documents, and do not necessarily indicate that an error or omission had been made. On the contrary, the majority of comments were concerning clarifications which VECTRA believed would make these documents more explicit and precise. Following the completion of the recommendations, the Peer Reviewer finds that the SSEL and Relay Review work followed the guidance of the GIP and NP-6041, and are complete and acceptable.

Peer Review of the HBR2 SSEL and Relay Review for A-46/IPEEE

Some of the issues addressed by VECTRA in it's peer review include the following:

- Indication requirements
- HVAC requirements
- Make-up capacities
- Boundary paths
- Component fail positions

It should be noted that CP&L initially provided verbal clarifications of many of the comments provided by the peer reviewer and, in addition, followed-up these verbal clarifications with written responses addressing each and every issue as well as incorporating some of these clarifications directly into the SSEL report. This effort produced a much clearer and more easy to read report.

Similar to the SSEL, detailed questions and documented responses were prepared for the relay review.

The relay review for HBR2 was performed on a draft version of the report referenced above. A revision of the document has since been made which incorporates all comments which were identified as a result of the peer review. The relay study for the Robinson plant is very thorough and complete and the process performed by CP&L is consistent with the methodology and procedures prescribed in the SQUG GIP, EPRI NP-7148 and NUREG-1407.

In general, the results of the SSEL and Relay peer review have indicated that individuals involved in this effort had a very thorough knowledge of the A-46 and IPEEE program, as well as a clear understanding of the safe shutdown methodology and the development of an essential relay listing.