DETROIT EDISON

Fermi 2

INDIVIDUAL PLANT EXAMINATION
(External Events)

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List of Abbreviations

Abbreviation	Full Name
AB	Auxiliary Building
AC	Alternating Current
ACI	American Concrete Institute
ADS	Automatic Depressurization System
AISC	American Institute of Steel Construction
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOV	Air-Operated Valve
AP	Annulus Pressurization
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	Boiler and Pressure Vessel Code
BOP	Balance of Plant
BWR	Boiling Water Reactor
CCDF	Conditional Core Damage Frequency (same as CCDP)
CCDP	Conditiona Core Damage Probability (same as CCDF)
CCHVAC	Control Center Heating, Ventilating, and Air Conditioning
CDFM	Conservative Deterministic Failure Margin
CECO	Central Component (database)
COP	Combination Operating Panels
CR	Control Room
CRD	Control Rod Drive
CS	Core Spray
CST	Condensate Storage Tank
CT	Current Transformer (Also CST event tree top in PSA model)
CTG	Combustion Turbine Generator
DBE	Design Basis Earthquake
DC	Design Calculation or Direct Current
DECo	Detroit Edison Company
DER	Deviation Event Report
DHR	Decay Heat Removal
DOF	Degree of Freedom
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EDP	Engineering Design Package
EECW	Emergency Equipment Cooling Water
EESW	Emergency Equipment Service Water
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
FCIA	Fire Compartment Interaction Analysis
F_i	Fire Ignition Frequency
FIE	Fire Initiated Event
FIVE	Fire-Induced Vulnerability Evaluation

List of Abbreviations

Abbreviation	Full Name
FHA	Fire Hazard Analysis
FOS	Functional Operating Sketch
FRSS	Fire Risk Scoping Study
GE	General Electric
GIP	Generic Implementation Procedure
GL	Generic Letter
HCLPF	High Confidence of Low Probability of Failure
HCU	Hydraulic Control Unit
HP	Horse Power (Also HPCI event tree top in PSA model)
HPCI	High Pressure Core Injection
HVAC	Heating, Ventilating, and Air Conditioning
HWC	Hydrogen Water Chemistry
I & C	Instrumentation and Controls
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular Stress Corrosion Cracking
IN	Information Notice
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
LERF	Large Early Release Frequency
LLNL	Lawrence Livermore National Laboratories
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LVS	Low Voltage Switchgear
MAAP	Modular Accident Analysis Program
MCC	Motor Control Center
MOV	Motor-Operated Valve
MPU	Modular Power Unit
MSIV	Main Steam Isolation Valve
MVS	Medium Voltage Switchgear
NEMA	National Electrical Manufacturers Association
NIAS	Non-Interruptible Air Supply
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
P & ID	Piping and Instrumentation Drawing
PGA	Peak Ground Acceleration
PIS	Plant Identification System
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
RB	Reactor Building
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling

List of Abbreviations

Abbreviation	Full Name
RFO	Refueling Outage
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RLE	Review Level Earthquake
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRS	Required Response Spectrum
RTD	Resistance Temperature Detector
RWCU	Reactor Water Cleanup
SCFM	Standard Cubic Feet per Minute
SEWS	Screening and Evaluation Worksheets
SGTS	Standby Gas Treatment System
SMA	Seismic Margin Assessment
SME	Seismic Margin Earthquake
SMM	Seismic Margins Methodology
SOP	System Operating Procedure
SPLD	Success Path Logic Diagram
SQUG	Seismic Qualification Users Group
SRSS	Square Root of the Sum of the Squares
SRT	Seismic Review Team
SRV	Safety Relief Valve
SSD	Safe Shut Down
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
STA	Shift Technical Advisor
TRS	Test Response Spectrum
TSR	Technical Service Request
UBC	Uniform Building Code
UFSAR	Updated Final Safety Analysis Report
USI	Unresolved Safety Issue
UTS	Ultimate Tensile Strength
WR	Work Request
ZPA	Zero-Period Acceleration

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WR	Work Request
ZPA	Zero-Period Acceleration

SECTION 1

EXECUTIVE SUMMARY

1.0 INTRODUCTION

In response to the NRC's request in Generic Letter 88-20, Supplement 4 **[1.1]** Detroit Edison has completed an individual plant examination of external events (IPEEE) for severe accident vulnerabilities for Fermi 2. External events are transient initiators external to the plant systems and include such events as fire, earthquakes, floods, high winds, and transportation and nearby facility accidents. This study complements the previously submitted individual plant examination (IPE) that treated severe accident vulnerabilities associated with internal events, typically initiated by equipment failures.

The IPE and IPEEE taken together fulfill the portion of the 1985 Severe Accident Policy Statement **[1.2]** that expected licensees "to perform a limited scope accident safety analysis designed to discover instances (i.e., outliers) of particular vulnerability to core melt or to unusually poor containment performance, given core melt accidents."

1.1 BACKGROUND AND OBJECTIVES

During the past 15 to 20 years, the NRC has placed increasing emphasis on the evaluation and use of severe accident information in regulating the current generation of nuclear power plants. This regulatory emphasis has manifested itself in a number of NRC actions which extend beyond the design basis of the current generation of plants. Such actions include the Anticipated Transient Without Scram (ATWS) Rule, the Severe Accident Policy Statement and the Station Blackout Rule.

In August of 1989, the NRC issued Supplement 1 to Generic Letter 88-20 requesting each utility to perform an Individual Plant Examination for severe accident vulnerabilities. The scope of such an effort involved the integrated analysis of plant and system response to a wide spectrum of internal, randomly initiated events such as reactor scram, loss of off-site power and loss of coolant accidents (LOCAs) with an emphasis on quantification of plant core damage frequency and evaluation of containment performance with regard to the release of radionuclides. The events analyzed in the IPE are, in many cases, far beyond the original design basis of the plant and extremely unlikely and are not expected to occur within the life of the plant. Nevertheless, the performance of such an effort provided new insight into system and plant capability and provided a tool for the quantitative evaluation of potential plant improvements and prioritization of plant activities.

The effort involved the use of a Level 1 Probabilistic Safety Assessment (PSA) and the performance of a Level 2 containment performance analysis aimed at identifying the dominant core damage risk contributors for the plant and the dominant potential causes of an off-site release of radioactivity. However, the IPE effort did not include external event initiators principally because at that time acceptable examination methods had not been identified nor had the scope of external events been determined. Subsequently, in June 1991 Supplement 4 to Generic Letter 88-20 was issued requesting licensees to now add external events to the severe accident examination process. While PSAs were cited as acceptable approaches to the fire and seismic portions of such an examination, other options were also described as viable alternatives. As discussed below, Detroit Edison elected not to use the formal PSA approach for the fire and seismic portions.

Detroit Edison has undertaken the performance of the Fermi 2 IPEEE in a manner that in conjunction with the IPE fulfills the following NRC objectives **[1.1]**:

1. Develop an appreciation for severe accident behavior;

2. Understand the most likely severe accident sequences that could occur under full power operating conditions;

3. Gain a qualitative understanding of the overall likelihood of core damage and release; and

4. If necessary, reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

1.2 PLANT FAMILIARIZATION

The Fermi 2 plant is located on the western shore of Lake Erie approximately 30 miles from Detroit, Michigan. Including licensed power uprate provisions, it is a 3430 MW(t) BWR-4 plant with a pressure suppression inerted Mark I containment.

In general, Fermi 2 can be considered similar in design to many other BWR-4 plants licensed in the U.S. However, Fermi 2 has a number of plant unique features which have been shown on balance in the IPE to favorably affect plant risk. These features include a standby, high pressure coolant inventory control system called standby feedwater; four emergency diesel generators coupled with intra-divisional cross-tie capability at the 480V level;; the availability of an on-site, blackstart combustion turbine generator; and the use of two completely independent "divisional" offsite power distribution systems and switchyards to provide offsite power to the plant.

The Fermi 2 Site is located in one of the most seismically stable regions in the United States. No earthquake epicenter has been located closer than about 25 miles, and only seven earthquakes have been reported within 50 miles of the site since the beginning of the 19th century. Category I structures at the plant are founded on bedrock and are designed so that the plant can be safely shut down in the event ground accelerations exceed those that are operationally tolerable. A peak ground acceleration of 0.15g was used for the safe shutdown design earthquake at Fermi 2.

The Fermi 2 fire protection program is designed to provide adequate fire protection for all potential fire hazards. Supplementing the basic fire protection design concepts of low combustible fire loading and separation of fire areas, the fire protection provisions include:

- fire protection water supply and distribution (includes electric pump and diesel driven fire pump as backup)
- fire detection and alarm
- gaseous suppression
- fixed water spray
- automatic sprinklers
- manual hose stations
- nitrogen system for containment inerting during operation

As an alternate to complete divisional fire protection in the control room, an alternate shutdown capability for the control room is provided.

Plant and site characteristics relative to the other external events covered by the IPEEE (high winds/tornadoes, external floods, transportation and nearby facility accidents) are within the guidelines provided in the Standard Review Plan even though Fermi 2 is not a "standard review plan plant".

1.3 OVERALL METHODOLOGY

Acceptable methods for performing the IPEEE are outlined in Generic Letter 88-20, Supplement 4 **[1.1]**, and the IPEEE guidance document NUREG-1407 **[1.3]**. The acceptable fire options were supplemented by NRC acceptance of EPRI's "Fire Vulnerability Evaluation" (FIVE) methodology **[1.4]**, and the seismic options were supplemented by Supplement 5 to Generic Letter 88-20 **[1.5]**. The methods utilized for Fermi 2 are outlined below.

Seismic - The EPRI Seismic Margin Assessment (SMA) approach **[1.6]** as specified for a focused scope plant in NUREG-1407 **[1.3]** is used as the framework for the seismic portion of the Fermi 2 IPEEE. The structures, systems and components required to be operable to validate two safe shutdown paths is evaluated against a review level (0.3g) earthquake, which is of greater magnitude than the design basis earthquake. The methodology provides screening techniques for this evaluation; items that do not screen out are evaluated in more detail to determine their seismic capacity. Since the Fermi 2 site seismic hazard is quite low based on both EPRI **[1.7]** and revised Livermore **[1.8]** hazard curves, the SMA approach for Fermi 2 was modified in February 1995 **[1.9]** to reduce the evaluation effort for some of the equipment that does not screen out against the review level earthquake; however, anchorage evaluation has been completed using the 0.3g earthquake requirements. This modified approach is very similar to the modified seismic scope subsequently provided by the NRC in September 1995 **[1.5]**.

- Fire - The EPRI Fire-Induced Vulnerability Evaluation (FIVE) technique **[1.4]** is used for the fire portion of the Fermi 2 IPEEE. This technique identifies fire initiators by compartment and then uses a screening process to ascertain if the probability of going to core damage is less than 1.0E-06/yr for each identified fire compartment. This screening effort includes a walkdown to verify assumptions credited in the screening process. Those compartments that do not screen out are then evaluated as potential vulnerabilities. The overall effort includes consideration of the six issues included in the Sandia/NRC Fire Risk Scoping Study **[1.10]**.

- Other External Events - The evaluations for other external events specified in GL 88-20 Supplement 4 (high winds and tornadoes, external floods, transportation and nearby facility accidents) utilize the progressive screening approach given in NUREG-1407 **[1.3]**. The basic approach is to show that the plant conforms to the 1975 Standard Review Plan criteria for these events.

1.4 SUMMARY OF MAJOR FINDINGS

A summary of the major findings of this IPEEE for each of the major external event categories is presented below.

1.4.1 Seismic Events

The plant was found to be seismically rugged in that upon completion of the few plant modifications and corrective maintenance activities discussed below, all structures, systems and components required for the two identified safe shutdown paths met the seismic capacity requirements of the 0.3g review level earthquake. No seismic vulnerabilities were identified. There were several observations made and insights gained that led to corrective action and planned future actions.

For example, some minor mounting hardware deficiencies were identified and have been corrected or will be corrected through work requests by the completion of the fall 1996 refueling outage (RFO5). The potential for four modest hardware changes were identified. One involves the bolting together of some adjacent relay panels to reduce the probability of relay chatter during a seismic event. The second involves replacing four low-ruggedness relays in the emergency diesel generator voltage sensing circuits. The third provides for evaluation of additional seismic restraint to a large air dryer on the second floor of the reactor building. Fourth deals with evaluating a weakness in the seismic load path for two CCHVAC instrument panels on the fifth floor of the reactor building.

Three insights of interest were gained during the seismic evaluation.

A large fraction of the minor hardware deficiencies found were believed to be associated with maintenance activities rather than original installations. Additional training will be incorporated in the continuing maintenance training program to increase the awareness level and emphasize the importance of mounting hardware installation and restoration during and after maintenance activities.

Operations training does not include a sustained loss of offsite power and combustion turbine generator Unit 1 (CTG 11-1) scenario as may result from a severe seismic event. Current simulator training assumes CTG 11-1 is restorable within the first 30 to 60 minutes after a loss of offsite power. A new simulator drill will be incorporated in the operator training program to address this scenario.

- During a severe seismic event, it is expected that many spurious alarms will be received in the control room due to low seismic ruggedness relay chatter. Although this may not have a direct effect on safe plant shutdown, it may cause some confusion in the control room. This item will also be included in the new seismic simulator training event.

Additional description and references to the associated documentation are provided in Sections 3 and 8.

1.4.2 Fire Events

The progressive screening process employed in the FIVE methodology led to six fire compartments that did not meet the screening criterion of less than 1.0E-6/yr core damage frequency (CDF). Since the screening criterion was only modestly exceeded (largest computed CDF was 4.5E-06/yr) and in view of the recognized conservatisms in the FIVE methodology as applied by Fermi 2, this result is considered to represent an acceptably low risk to fire induced damage and thus presents no vulnerabilities.

Five of the unscreened six compartments are control center compartments including the relay and control rooms, the switchgear rooms, and the Division 1 portion of the miscellaneous room, which is a finding consistent with other plants. The sixth compartment is the second floor of the reactor building. This latter unscreened compartment leads to the single fire insight in that the dominating contributors are cabinets used for dedicated shutdown whose loss would isolate the affected equipment from the main control room thereby causing loss of the equipment function. While this loss potential is adequately covered by current operator training, additional Fire Brigade drills in the vicinity of these cabinets are planned to increase the awareness of the brigade members to the need to quickly isolate and extinguish such cabinet fires.

Additional description and references to the associated documentation are provided in Sections 4 and 8.

1.4.3 Other Events (High Winds, Floods, and Transportation and Nearby Facility Accidents)

The site review and design comparison relative to the 1975 Standard Review Plan revealed no vulnerabilities or insights relative to these other external events. The potential, however, for a common cause failure of diesel generator cooling function due to ice formation in the pump column was recently recognized. This event is the subject of LER 96-001 **[1.11]**, and is currently under evaluation.

1.5 REFERENCES

1.1 "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" - 10CFR50.54(f)," Generic Letter 88-20, Supplement 4, June 28, 1991.

1.2 "Severe Reactor Accident Design; Policy Statement and Withdrawal of Proposed Rulemaking," Federal Register, Volume 50, p. 32138, August 8, 1985.

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

1.4 "Fire-Induced Vulnerability Evaluation (FIVE)," EPRI-TR-100370, Final Report, April 1992, and Revision 1, September 29, 1993.

1.5 "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," - 10 CFR 50.54(f)," Generic Letter 88-20, Supplement 5, September 8, 1995.

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

1.7 "Probabilistic Seismic Hazard Evaluations at Plant Sites in the Central And Eastern United States: Resolution of the Charleston Earthquake Issue," Appendix E (Results for 57 Sites), EPRI NO-6395-D, April 1989.

1.8 P. Sobel, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," NUREG-1488, April 1994.

1.9 Detroit Edison letter to NRC, "Revision to IPEEE Seismic Scope", NRC-95-0003, February 9, 1995.

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

1.11 "Emergency Diesel Generator Cooling Water Function Potentially Lost Due to Ice Formation in the Pump Column", LER 96-001, Event Date February 6, 1996, Report Date March 6, 1996.

SECTION 2

EXAMINATION DESCRIPTION

2.0 INTRODUCTION

This Individual Plan Examination for External Events (IPEEE) for the Fermi 2 nuclear power plant was conducted to meet the objectives stated in Supplement 4 to NRC Generic Letter 88-20 **[2.1]**. This report documents that examination. Display of the information generally follows the suggested table of contents given in the IPEEE guidance document **[2.2]** with some additional subsections. Minor changes have been made in some of the section titles, and portions of Section 4 (fire) have been partitioned into two parts (A and B) for convenience of display.

It is evident that this external events examination is essentially a snapshot in time of plant conditions. Thus, walkdowns, review of plant drawings and procedures, etc. used in the examination process pertain to the plant at a given time. Some of this activity took place over a year prior to issuance of this report. Thus, any recent changes in plant conditions and documentation, except as noted, would not be reflected in this external events examination.

2.1 GENERAL PLANT DESCRIPTION

The Fermi 2 plant is located on the western shore of Lake Erie approximately 30 miles from Detroit, Michigan. Including licensed power uprate provisions, it is a 3430 MW(t) BWR-4 plant with a pressure suppression Mark I containment. A summary of some of the key design features of the plant is provided in Table 2-1.

In general, Fermi 2 can be considered similar in design to many other BWR-4 plants licensed in the U.S. However, the Fermi 2 design has a number of plant unique features which have been shown in the performance of the IPE to influence plant risk. These features include a standby, high pressure coolant inventory control system called standby feedwater; four emergency diesel generators coupled with intra-divisional cross-tie capability; the availability of an on-site, blackstart combustion turbine generator (CTG 11-1); and the use of two completely independent offsite power distribution systems and switchyards to provide offsite power to the plant.

The standby feedwater system is a non-safety grade system consisting of two motor driven pump trains, each capable of providing up to 600 gpm of coolant makeup to the reactor pressure vessel (RPV) at high pressure from the condensate storage tank through the feedwater lines. The standby feedwater system is incorporated into the Fermi 2 emergency operating procedures and is available as a coolant makeup source for transient and small LOCA events.

On-site emergency AC power is provided at Fermi 2 by four emergency diesel generators (EDGs). Power distribution is divided into two redundant divisions with two EDGs per division. Each division can provide power to necessary shutdown cooling and control power systems to ensure long term operation and control. In addition, if an EDG fails, the 480V buses in the same division can be cross-connected to provide power to necessary low voltage plant loads, and by taking actions such as interlock defeats, the 4160V buses can also be cross-connected.

In addition to the four EDGs, the Fermi 2 site has four combustion turbine generators (18.8 MW each) located adjacent to the decommissioned Fermi 1 plant. These generators are used by Detroit Edison for peaking loads and can be connected to the Division 1 offsite power feed to Fermi 2. One of the combustion turbine generators (CTG 11-1) can be started from the Fermi 2 control room and is capable of starting without power from the grid or Fermi 2 (i.e., it is blackstart capable). CTG 11-1 in conjunction with the standby feedwater system make up part of the Appendix R Alternate Shutdown System and are included in the Technical Specifications.

The Fermi 2 plant has a unique offsite power supply configuration, consisting of independent switchyards, as well as multiple, diverse onsite AC power supplies. Division 1 offsite power is provided through a connection to a 120kV portion of the Detroit Edison grid through a switchyard located near the Fermi 1 plant. This switchyard is connected to the Fermi 2 plant through underground lines. Division 2 power is provided to Fermi 2 through an independent switchyard which is connected to a 345kV portion of the Detroit Edison grid. The Division 2 switchyard is connected to the Fermi 2 plant through overhead transmission lines. No connections exist between the switchyards. The only connection between the two divisions of power is between the 4160V ESF buses through a maintenance cross-tie which by administrative controls is not allowed to be in-service during normal power operations. However, this bus tie is available and can provide additional flexibility in recovery from beyond design basis events.

The Fermi 2 site is located in one of the most seismically stable regions in the United States. No earthquake epicenter has been located closer than about 25 miles and only seven earthquakes have been reported within 50 miles of the site since the beginning of the 19th century. None of these was greater than Intensity V on the Modified Mercalli Scale. Category I structures at the plant are founded on rock and are designed so that the plant can be safely shut down in the event ground accelerations at the site exceed those that are operationally tolerable. The Seismology Division of the National Ocean Survey, NRC staff's seismological advisor, concluded that an acceleration of 0.15g resulting from a strong intensity earthquake would be adequate for representing the ground motion from the maximum earthquake likely to affect the site. This acceleration was used for the seismic design of Fermi 2 **[2.10]**.

The Fermi 2 fire protection program is designed to provide adequate fire protection for all potential fire hazards. Supplementing the basic fire protection design concepts of low combustible fire loading and separation of fire areas, the fire protection provisions include:

- fire protection water supply and distribution (includes electric pump and diesel driven fire pump as backup to GSW supply)
- fire detection and alarm
- gaseous suppression
- fixed water spray
- automatic sprinklers
- manual hose stations
- nitrogen system for containment inerting during operation

As an alternate to complete divisional fire protection in the control room, a dedicated alternate shutdown capability for the control room is provided. This Appendix R alternate shutdown system is separate and remote from the control center complex (control, relay, and cable-spreading rooms) and is designed to (1) achieve and maintain the reactor in a subcritical condition; (2) maintain reactor coolant inventory; (3) achieve and maintain hot shutdown; (4) achieve cold shutdown within 72 hours; and (5) maintain the reactor in a cold shutdown condition thereafter. In the event of a fire which would prevent achieving a safe shutdown from the control room, the only required operator action in the control room is a manual reactor scram if an automatic trip has not already occurred.

Plant and site characteristics relative to the other external events covered by the IPEEE (high winds/tornadoes, external floods, transportation and nearby facility accidents) are within many of the guidelines provided in the Standard Review Plan even though Fermi 2 is not a "standard review plan" plant. The external event in this category actually experienced at the site was the 1989 low lake level caused by a sustained strong offshore wind compounded by silt buildup in the general service water (GSW) intake canal that led to a loss of GSW. However, loss of GSW is included as part of the design basis.

2.2 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

The Fermi 2 IPEEE meets the objectives stated in Section 1 of the IPEEE Generic Letter **[2-1]** and generally follows the stipulations of that letter and of the related submittal guidance **[2.2]**. The objectives are restated below.

The general purpose of this IPEEE is for the Fermi 2 staff, in conjunction with the IPE to:

1. Develop an appreciation for severe accident behavior;

2. Understand the most likely severe accident sequences that could occur under full power operation conditions;

3. Gain a qualitative understanding of the overall likelihood of core damage and release; and

4. If necessary, reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

To meet these objectives, a methodology was employed that is generally consistent with the options described in the IPEEE guidance document NUREG-1407 **[2-2],** using the non-PSA options for both the fire and seismic portions. The overall methodology used in this IPEEE is described in Section 2.3 as well as in the individual analysis sections.

The evaluation was performed under the control of Detroit Edison using Fermi 2 plant personnel supplemented by contractor support. Walkdowns employed both Fermi 2 and contractor personnel as did the peer reviews. Senior consultants for selected applications were also utilized. Details on the makeup of the IPEEE evaluation team are included in Section 6.

The Fermi 2 IPEEE is documented in a traceable manner that provides the basis for the findings in a tiered approach. Specifically, there are two tiers of documentation to support the Level 2 IPEEE:

- Documentation submitted to the NRC, referred to as Tier 1.
- Documentation developed and retained as formal in-house documents to provide additional support for the submittal, referred to as Tier 2.

2.3 OVERALL METHODOLOGY

Acceptable methods for performing the IPEEE are outlined in Generic Letter 88-20, Supplement 4 **[2.1]**, and the IPEEE guidance document NUREG-1407 **[2.2]**. The fire options were supplemented by NRC acceptance of the "Fire Vulnerability Evaluation" (FIVE) methodology **[2.3]**, and the seismic options were supplemented by Supplement 5 to Generic Letter 88-20 **[2.4]**. The methods utilized for Fermi 2 are outlined below.

2.3.1 Seismic

The EPRI Seismic Margin Assessment (SMA) approach **[2.5]** as specified for a focused scope plant in NUREG-1407 **[2.2]** is used as the framework for the seismic portion of the Fermi 2 IPEEE. The basic approach is to identify two shutdown success paths (one capable of handling a small break LOCA) that can bring the plant to hot or cold shutdown following a pre-selected postulated seismic event. The structures, systems and components required to be operable to validate these success paths is evaluated against a review level (0.3g) earthquake (RLE), which is of greater magnitude than the design basis earthquake. The methodology provides screening techniques to be used by a seismic review team to screen structures and equipment during plant walkdowns. Items that do no screen out are evaluated in more detail to determine their high-confidence-low-probability-of-failure (HCLPF) capacity. Since the Fermi 2 site seismic hazard is quite low based on both EPRI **[2.6]** and revised Livermore **[2.7]** hazard curves, the SMA for Fermi 2 was modified in February 1995 **[2.8]** to reduce the evaluation effort for some of the equipment that does not screen out against the review level earthquake. Namely, highly sophisticated HCLPF calculations would not be completed for non-screened items. Instead, simple evaluations would be attempted to demonstrate RLE ruggedness; otherwise, only compliance to the seismic design basis is demonstrated. It should be noted that anchorage evaluations would be completed using the 0.3g earthquake for all components involved. This approach is very similar to the modified seismic s[illegible] subsequently provided by the NRC in September 1995 **[2.4]**. Additional detail is given in Section 3.0.

2.3.2 Internal Fires

The EPRI Fire-Induced Vulnerability Evaluation (FIVE) technique **[2.3]** is used for the fire portion of the Fermi 2 IPEEE. This technique identifies fire initiators by compartment and then uses a multi-step screening process to ascertain if the probability of going to core damage is less than 1.0E-06/yr for each identified fire compartment. This screening effort includes a walkdown to verify assumptions credited in the screening process. Those components that do not screen out are then evaluated as potential vulnerabilities. The process is structured to take advantage of existing plant specific fire protection programs and 10CFR50 Appendix R evaluations. The overall effort also includes consideration of the six issues included in the Sandia/NRC Fire Risk Scoping Study **[2.9]**. Additional detail on the fire methodology is given in Section 4.

2.3.3 Other External Events

The other external events specified in GL 88-20 Supplement 4 (high winds and tornadoes, external floods, transportation and nearby facility accidents) utilize the progressive screening approach given in NUREG-1407 **[2.2]** that centers around conformance with the 1975 Standard Review Plan (SRP). While the Fermi 2 construction permit preceded issuance of the SRP, many of the plant design features had been brought into compliance with the SRP because of the extended time over which the plant was designed and constructed. The basic approach is to show that the plant still conforms to the 1975 Standard Review Plan criteria for these events.

2.4 INFORMATION ASSEMBLY

The information utilized in this IPEEE came from a myriad of sources. Classification of sources consist of written documentation (Detroit Edison and external), Fermi 2 drawings, individual expertise (Fermi 2 and consultant), and walkdowns. Major examples of these classifications are given below. Specific information sources are referenced elsewhere in the report within the context that they are used.

1. UFSAR
2. Standard Review Plan
3. Design calculations
   - Fire (e.g., "Appendix R" calculations)
   - Seismic (seismic and anchorage calculations)
4. Fire Hazards Analysis (Appendix 9A, Fermi 2 UFSAR)
5. Alarm Response Procedures (fire)
6. Standard Operating Procedures
7. Abnormal Operating Procedures
8. Central Component Data Base (CECO)
9. Automated Records Management Systems
10. Vendor Manuals
11. Seismic Qualification Reports
12. SQUG/EPRI Seismic Reports (e.g., EPRI FIVE and SMA reports)
13. Fermi 2 Cable Data Base
14. Design Basis Documents
15. Fermi 2 IPE Report
16. Miscellaneous Drawings
    - Schematics
    - Functional Operating Sketches
    - Equipment
    - Anchorage
17. Shift Technical Advisors and Operation Personnel
18. Seismic Senior Consultants
19. Walkdowns
    - Preliminary Seismic
    - Seismic Capability
    - Subsequent Seismic Evaluation
    - Fire Information Gathering
    - FIVE Confirmatory

2.5 REFERENCES

2.1 "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" - 10CFR50.54(f)," Generic Letter 88-20, Supplement 4, June 28, 1991.

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

2.3 "Fire-Induced Vulnerability Evaluation (FIVE)," EPRI-TR-100370, Final Report, April 1992, and Revision 1, September 29, 1993.

2.4 "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," - 10CFR50.54(f)," Generic Letter 88-20, Supplement 5, September 8, 1995.

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

2.6 "Probabilistic Seismic Hazard Evaluations at Plant Sites in the Central And Eastern United States: Resolution of the Charleston Earthquake Issue," Appendix E (Results for 57 Sites), EPRI NO-6395-D, April 1989.

2.7 P. Sobel, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," NUREG-1488, April 1994.

2.8 Detroit Edison letter to NRC, "Revision to IPEEE Seismic Scope", NRC-95-0003, February 9, 1995.

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

2.10 "Safety Evaluation Report related to the operation of Enrico Fermi Atomic Power Plant Unit No. 2," NUREG-0798, July 1981.

Table 2-1 Summary Of Fermi 2 Design Features

SAFETY FUNCTION	KEY SYSTEMS/FEATURES
Coolant Inventory Makeup	1) High Pressure Systems: - Main Feedwater - High Pressure Coolant Injection (HPCI) - Reactor Core Isolation Cooling (RCIC) - Standby Feedwater - Control Rod Drive Hydraulic System (CRD)
	2) Medium Pressure System: - Condensate
	3) Low Pressure Systems: - Low Pressure Coolant Injection (LPCI) - Low Pressure Core Spray (LPCS) - RHR Service Water (RHRSW)
Containment Heat Removal	1) Main Condenser
	2) Residual Heat Removal (RHR) via - Suppression Pool Cooling Mode - Shutdown Cooling Mode - Drywell Spray Mode
	3) Containment Vent
Reactivity Control	1) Control Rods
	2) Standby Liquid Control (SLCS)
	3) Alternate Boron Injection via - Standby Feedwater System (EOP directed) - Condensate/Feedwater System - Reactor Water Cleanup System

Table 2-1 Summary Of Fermi 2 Design Features (Cont'd.)

SAFETY FUNCTION	KEY SYSTEMS/FEATURES
Key Support System	1) DC Power System
	2) Emergency On-Site AC Power From Four EDGs
	3) Alternate AC Power From Onsite Blackstart Combustion Turbine Generator
	4) Offsite Power Provided To Each Division through Separate, Independent Switchyards
	5) RHR Service Water (RHRSW), EDG Service Water (EDGSW) and Emergency Equipment Service Water (EESW) reject heat to Dedicated Heat Sink
Primary Containment Structure	1) BWR Mark 1
	2) 0.29 million cu. ft.
	3) 56 psig Design Pressure/140 psig Ultimate Pressure
Containment Protection	1) Containment Hard Pipe Venting From Wetwell or Drywell
	2) RHR in Containment Spray Mode
	3) Nitrogen for Containment Inerting During Operation

SECTION 3

SEISMIC ANALYSIS

3.0 METHODOLOGY SELECTION

In accordance with NUREG-1407 **[3.1]**, two methodologies are considered acceptable to identify potential seismic vulnerabilities at nuclear power plants. The first is a seismic probabilistic risk assessment (PRA) method and the second is a seismic margins methodology (SMM). Two different SMMs are considered acceptable: an NRC-developed method based on an event/fault tree approach for delineating accident sequences, and an EPRI-developed method based on a systems "success path" approach. For these methodologies to be acceptable, NUREG-1407 requested that certain methodology enhancements be included.

As stated in Detroit Edison's response **[3.2]** to Generic Letter (GL) 88-20, Supplement 4 **[3.3]**, the EPRI SMM approach described in report NP-6041-SL **[3.4]**, with the enhancements requested in NUREG-1407, was selected for implementation of the seismic IPEEE program at Fermi 2.

In April 1994, the NRC issued Information Notice (IN) 94-32 **[3.5]** on the subject of revised seismic hazard estimates published by the Lawrence Livermore National Laboratory (LLNL) in NUREG-1488 **[3.6]**. The NRC suggested in the IN that licensees who had not completed their seismic IPEEE may submit a request to adjust their schedule commitment to allow the NRC staff review of the revised seismic hazard estimates and take the revised information into account in their IPEEE programs.

On January 30, 1995, the NRC published a proposed Supplement 5 to Generic Letter 88-20 in the Federal Register for public comment. GL 88-20, Supplement 5 **[3.7]**, was subsequently issued in September 1995 with the resolution of the public comments. The supplement acknowledged that the results of the revised LLNL seismic estimates indicate that the perceived risk has been reduced for most plant sites in the central and eastern U.S. Therefore, the NRC proposed reducing the scope of the seismic IPEEE programs for licensees with focused-scope and full-scope plants.

For plants performing a reduced-scope seismic margin assessment (SMA), supplement 5 indicates that the seismic capacities for reactor internals and soil-related failures need not be evaluated for the seismic IPEEE. Supplement 5 also states that "*modifying the scope of the seismic IPEEE for focused-scope plants in this manner will make these evaluations equivalent to those for the reduced-scope plants, with additional evaluations of a few known weaker, but critical, components or items*."

These additional weaker items were further defined in Attachment 1 to GL 88-20, supplement 5. They include masonry and block walls, flat-bottom tanks, relay chatter, and other items. The other items "*pertain to inadequate anchorage and bracing, adverse physical interactions, building impact, or pounding. These items include the weaker components of the diesel generators or pumps. However, the licensee's seismic review team should determine whether seismic capacities of these components need to be evaluated in the seismic review.*"

In supplement 5, the NRC staff reemphasized that the guidance in the generic letter and NUREG-1407 does not preclude the use of well-based expert judgment and efficient approaches that minimize the effort of conducting an IPEEE. GL 88-20, supplement 4, in describing the acceptable IPEEE examination methods, states that "*the application of the above approaches involves considerable judgment with regards to the requested scope and depth of the study, level of analytical sophistication, and level of effort to be expended.*"

Prior to final issuance of Supplement 5 to GL 88-20, Detroit Edison notified the NRC by letter dated February 9, 1995 **[3.8]** of a revision in the Fermi 2 IPEEE seismic scope. The revision was structured based on the stage of completion of the Fermi 2 seismic IPEEE program, the revised perceived seismic risk for the Fermi 2 site, the general industry direction for completing the IPEEE, and the preliminary information in GL 88-20, supplement 5. Moreover, the Fermi 2 scope revision philosophy was not far different than that encompassed by the subsequently issued Supplement 5 to GL 88-20. Fermi 2 had completed an evaluation of the reactor internals and relay chatter effects. All safety-related buildings are founded on bed rock; thereby, eliminating potential soil-related failures. Furthermore, Fermi 2 had completed a significant portion of the seismic IPEEE study.

The revision described in Detroit Edison scope change letter to the NRC stated that the Fermi 2 SMA will proceed along the focused-scope evaluation path as originally stated in the response to GL 88-20, supplement 4. However, for components that do not screen out using the EPRI screening methodology, and in lieu of performing highly sophisticated high-confidence-of-low-probability-of-failure (HCLPF) calculations, simplified evaluations will be performed to demonstrate ruggedness to the review level earthquake. Any components failing the simplified evaluation check will only be evaluated to demonstrate compliance with the design basis. The letter also stated that the Fermi 2 scope change may amount to no reduction in scope, if using the revised approach demonstrates compliance to the Review Level Earthquake (RLE) for all items included in the SMA. It should be noted that anchorage evaluation, which typically limits component seismic capacity, has been completed using the focused scope requirements of 0.3g ground acceleration.

3.1 SEISMIC MARGIN METHOD

As stated above, the methodology selected for completing the seismic IPEEE for Fermi 2 is the EPRI-developed seismic margin assessment (SMA) method outlined in NP-6041. The philosophy from which seismic margin methods emerged was based on increasing knowledge in the geoscience field, which led to the belief that it is possible for a nuclear plant site to be subjected to earthquake ground motion greater than its design basis. The SMM utilizes inherent conservatism in current design practices along with more realistic seismic capacity evaluations to calculate the margin above the seismic design basis that exists in operating nuclear plants. Only those systems and components required to bring the plant to, and maintain a safe shutdown condition following the seismic event need to be examined.

To define this margin, it is practical to select a conservatively high earthquake level and use it as a benchmark to demonstrate sufficient margin above the plant design basis and develop a high confidence in the seismic capacity of structures, systems and components. The earthquake level chosen for the review is called the RLE. For Fermi 2, this review level earthquake has been defined by the NRC in NUREG-1407 as a NUREG/CR-0098 **[3.9]** median spectrum anchored at 0.3g. EPRI NP-6041 refers to the earthquake level against which the plant is evaluated as the Seismic Margin Earthquake (SME). For purposes of the Fermi 2 seismic margin assessment, the RLE and the SME are set equal, and terms are interchangeable. The primary purpose of margin reviews is to demonstrate sufficient margin over the safe shutdown earthquake (SSE) to assure plant safety and to find any "weak links" which might limit the plant capability to safely shutdown after a seismic event bigger than the SSE. The EPRI methodology concentrates on demonstrating that a reliable operational path exists to shut down and maintain the plant in either hot or cold shutdown mode for a minimum of 72 hours following the seismic event.

Two alternate safe shutdown paths, called "success paths," are selected. In NUREG-1407, the NRC requested that, to the maximum extent possible, the alternate path involve operational sequences, systems, piping runs, and components different from the preferred path. Only plant components required for the operation of systems on the success paths are seismically evaluated for the RLE. The seismic margin assessment method relies heavily on earthquake experience data, generic equipment qualification, fragility test data, and seismic PRA results.

EPRI NP-6041 provides a set of screening guidelines to be used by the seismic review team (SRT) to screen structures and equipment, against the RLE, during plant walkdowns. The screening also relies on the judgment and the experience of the SRT. More detailed evaluations may be required to establish the seismic capability of items that do not meet the screening criteria or are judged by the SRT to warrant further reviews.

Detroit Edison notified the NRC by letter **[3.8]** of a change in scope in the Fermi 2 seismic IPEEE program, as a result of the revised seismic hazard estimates in NUREG-1488 **[3.6]** and the general nuclear industry reaction to the revised estimates. The Fermi 2 program revision was structured based on the significance of different elements and evaluations in the SMM as perceived by experts in the seismic and risk areas, and based on the stage of completion of the Fermi 2 program. Detroit Edison stated in their letter that highly sophisticated HCLPF calculations will not be completed for items that do not screen out. Instead, simple evaluations will be attempted to demonstrate RLE ruggedness; otherwise, only compliance to the seismic design basis will be demonstrated.

3.1.1 Review of Plant Information, Screening and Walkdown

Fermi 2 uses a General Electric (GE) Company, single cycle, forced circulation Boiling Water Reactor (BWR) of the BWR 4 Class, with a pressure-suppression Mark I containment. The containment is a steel plate pressure vessel consisting of a light-bulb-shaped drywell and a torus-shaped pressure suppression chamber. The plant is similar in design to Browns Ferry Nuclear Plant Units 1, 2, and 3, Cooper Nuclear Station, Edwin I. Hatch Unit 1, and Brunswick Steam Electric Plant Units 1 and 2. The uprated licensed design power rating for Fermi 2 is 3430 MWt, with a turbine generator design electrical output of 1203 MWe and a rated net electrical output of 1154 MWe.

The Fermi 2 site is located on the shore of the western end of Lake Erie at Lagoona Beach in Frenchtown Township, Monroe County, Michigan. The site is approximately six miles northeast of Monroe, Michigan, 30 miles southwest of downtown Detroit, and 25 miles northeast of Toledo, Ohio. The site consists of approximately 1120 acres.

The site is located within the Central Stable Region tectonic province of the North American Continent. Some regional faulting and seismic activity is known, but the region is characterized as one of relative stability. There are no known faults within 25 miles of the site and there are no capable faults within 200 miles of the site. The site is located in one of the most seismically stable regions in the United States.

The area is characterized by glacial landforms and by beach and lacustrine deposits formed during the fluctuations of the Great Lakes. The glacial deposits overlie maturely dissected bedrock and broad areas of relatively flat-lying bedrock. All major Fermi 2 Category I structures are supported in the Bass Islands dolomite (bedrock).

The principle structures located on the plant site include the following:

- The reactor building, which houses the drywell, the suppression pool, the Nuclear Steam Supply System (NSSS), the engineered safety features, some auxiliary system equipment, and the fuel storage and shipping area;
- The turbine building, which houses the power conversion equipment, the off-gas system, and plant auxiliaries;

- The auxiliary building, which houses the main control room, the computer facility, electrical equipment, and HVAC equipment;
- The radwaste building, which houses the radioactive waste treatment facilities for liquid and solid waste;
- The RHR complex, which houses the emergency diesel generators (EDGs), the RHR cooling towers, the RHR service water reservoirs, and the RHR service water, emergency equipment service water, and EDG service water pumps; and
- Two natural draft hyperbolic circulating water cooling towers and corresponding intake and discharge structures.

General plant building arrangement is shown in Figure 3-1.

This section briefly describes the seismic design basis of the plant and identifies the analytical methods, codes and standards, and other design requirements. In addition, this section briefly describes the screening criteria used for the Seismic Margins Assessment of plant structures and equipment and the walkdown criteria and methods.

Throughout this report, the terms Safe Shutdown Earthquake (SSE) and Design Basis Earthquake (DBE) represent the same earthquake and are used interchangeably.

3.1.1.1 Plant Seismic Design Basis

The seismic design classification of Fermi 2 structures, systems, and components is established in accordance with the requirements of Regulatory Guide 1.29 **[3.10]**. Plant structures, systems, and components which are important to safety and must be designed to maintain their safety function in the event of a safe shutdown earthquake are designated seismic Category I. They are classified as safety-related if they are necessary to ensure the integrity of reactor coolant pressure boundary, the capability to shut down the reactor and to maintain it in the safe shutdown condition, and the capability to prevent or mitigate the consequences of an accident that could result in potential offsite exposure beyond 10CRF100 **[3.11]** limits.

3.1.1.1.1 Seismic Design

The dynamic response and design of the Fermi 2 structures and components due to earthquake loading was divided into two broad areas of analysis. First, major buildings and structures which house or support Category I systems and components were modeled and analyzed. Second, the results of the building analyses were used as forcing functions in the dynamic analyses of smaller Category I systems and components.

3.1.1.1.1.1 Ground Response Spectra

The site ground response spectra for the operating-basis earthquake (OBE) and the safe shutdown earthquake (SSE), in the horizontal direction, are shown in Figures 3-2 and 3-3, respectively. Vertical ground acceleration for the OBE and SSE is two-thirds of the horizontai ground acceleration. The Fermi 2 Category I structures are designed to withstand a maximum horizontal ground acceleration of 0.15g and a maximum vertical acceleration of 0.10g for the SSE. For the OBE, maximum horizontal ground acceleration is 0.08g and maximum vertical ground acceleration is 0.05g. The shapes of the OBE and SSE spectra essentially conform to the 1940 El Centro earthquake with minor adjustments to account for the 1949 Olympia, Washington and the 1935 Helena, Montana earthquakes.

3.1.1.1.1.2 Acceleration Time Histories

Dynamic response of the Fermi 2 building structures was determined from detailed time history analyses of representative models subjected to four time history base excitations. Scaled actual earthquake records were used in addition to those used to describe the basic ground spectra to ensure a broad-band frequency content for equipment seismic qualification purposes. The input spectra were generated from the following four earthquakes and their horizontal time history records:

Direction	Earthquake
N-S	El Centro, CA., 5-18-40
N-S	El Centro, CA., 12-30-34
S-80-W	Olympia, WA., 4-13-49
N-21-E	Taft, CA., 7-21-52

Ground response spectra for a system with two percent critical damping were generated for each of the earthquake records shown above. Sixty periods from 0.1 sec. to 1.0 sec. were used in the generation of the ground spectra for each record. The duration of each record required to give maximum responses in the period range of interest was determined. Each earthquake record was scaled so that the area under the acceleration response spectra equaled the area under the recommended OBE spectra. Vertical OBE ground accelerations were obtained by scaling the horizontal ground accelerations. Maximum horizontal and vertical ground accelerations for SSE were obtained by multiplying the OBE values by a factor of two.

3.1.1.1.1.3 Seismic Models

The seismic response of all Category I structures was determined by applying earthquake ground motions to appropriate dynamic models. The lumped mass dynamic analysis approach was used for the Fermi 2 structures in which it was assumed that the entire mass of the structure is concentrated at a number of discrete points.

The horizontal dynamic analyses were performed using a shear structure system, a frame structure system, or a combined shear-frame structure system. The massive, stiff floor slab/shear wall configurations of the reactor/auxiliary building and the RHR building were modeled as spring-slab systems. The slabs were treated as infinitely rigid in their own planes and were interconnected by weightless linear elastic springs used to simulate the stiffness of shear walls. A three-dimensional frame analytical model was used to represent the containment shield, containment vessel, RPV and internals, reactor support pedestal, and sacrificial shield. The lumped masses in this portion of the model were allowed two translational and three rotational degrees of freedom and were interconnected with the frame members. Since the slabs were considered infinitely rigid in their own planes, their resulting rigid body motions had three degrees of freedom: horizontal translation in two perpendicular directions and rotation about a vertical axis. The slab model and the frame model were connected by axial springs at various elevations to represent the behavior of the structure more accurately. This model is shown in Figure 3-4.

In order to study the interaction between the RPV and the rest of the structure, the model for the RPV and its internals was included in the analysis and is shown in Figure 3-5. The RPV was supported by the reactor pedestal at its base (skirt) and laterally by the stabilizer and refueling bellows near the top.

The crane bridge and associated steel structures were included in the horizontal dynamic model as shown in Figure 3-6. The model was based on the assumption that the crane would be parked at the end bay during a seismic event.

The horizontal model developed for the RHR building was similar to that for the reactor/auxiliary building and is shown in Figure 3-7.

The vertical dynamic model of the reactor/auxiliary building was developed on the basis that the amplification in the vertical direction was a function of the axial stiffness of the walls and bending stiffness of the beam-slab system. The vertical stiffness was due mainly to two structural systems in the model: the reactor containment shield and the reactor/auxiliary building walls. The two wall systems were connected by the reactor building floor slab at all the floor elevations. The auxiliary building floor slab was represented by a single-degree-of-freedom system connected to the joints of the reactor/auxiliary building wall system at each elevation. In the dynamic model, the masses were allowed to displace relative to one another, with one degree-of-freedom in the vertical direction. The vertical dynamic model is shown in Figure 3-8.

The vertical model developed for the RHR building was similar to that for the reactor/auxiliary building and is shown in Figure 3-9. The vertical stiffness of the RHR building was also due mainly to two structural systems: the cooling tower walls and the RHR building walls.

Buried Category I electrical ducts and piping run between the reactor building and RHR building. The ducts and piping were analyzed for seismic wave propagation in the soil and for relative seismic displacement between the duct and piping anchor points and the buildings. The ducts and piping were evaluated using the "beam on elastic foundation" concept.

3.1.1.1.1.4 Soil-Structure Interaction

Category I structures at Fermi 2 are founded on bedrock. A study **[3.12]** was completed for Fermi 2 structures founded on rock in which it was shown that soil-structure interaction was insignificant. Therefore, the spectra developed for the bedrock represent the response to the base excitation.

3.1.1.1.1.5 Structural Damping

All seismic Category I structures consist of reinforced concrete and welded/bolted structural steel. The Fermi 2 Updated Final Safety Analysis Report (UFSAR) **[3.13]** Table 3.7-2 provides the following damping values for Fermi 2 representative structural items:

	Percent of Critical	
Item	OBE	SSE
Welded and high strength bolted steel framed structures	2.0	5.0
Bolted steel framed structures	5.0	10.0
Welded structural assemblies	2.0	4.0
Reinforced concrete structures	2.0	5.0

Damping values of two and five percent were used in the design basis seismic analyses for OBE and SSE, respectively. Also, other special damping values for RPV components are shown in UFSAR Figure 3.7-16.

3.1.1.1.1.6 Development of Floor Response Spectra

The in-structure response spectra were arrived at by averaging the results obtained from the four time history input excitations discussed in Section 3.1.1.1.1.2. To establish slab or wall motions, the time-history forcing functions were used to excite the building models used in the system analyses. Resulting time-history slab or wall motions were used to generate floor response spectra for the analysis of subsystems (i.e., components supported in the buildings). At each spectra period for a given spectra damping, the average response from the four earthquake excitations was calculated and plotted. The plotted spectra curves were smoothed by enveloping the peaks with a smooth curve which extended ten percent to either side of the peak for horizontal and 20 percent for vertical. In the vertical direction, response spectra were generated at two elevations; the spectra at other than the two elevations were not generated, but were classified in one of the two elevations.

To determine the response spectrum of a slab at a particular level, the vertical model for that structure was modified at that level to include the multi-degree behavior of the slab system. The modified vertical model for reactor/auxiliary building elevation 613'-6" is shown in Figure 3-10. The slab system consisted of five masses, and the springs on each side and was connected to the same wall joint. Similarly, the stiffness parameters of the auxiliary building slab system were determined.

3.1.1.1.2 Site Specific Earthquake

In response to requests for information from the NRC Geosciences Branch, a site-specific ground response spectrum was developed in 1981. The site-specific spectrum exhibited a significantly higher ground response than the DBE ground response spectrum. The site specific earthquake was not to be considered as a new design basis earthquake; however, it was used to determine the adequacy of a selected shutdown path components as described in Section 3.1.1.1.4.4.

Based on two site specific response spectra for nearby magnitude 5.3 earthquake records developed by Lawrence Livermore Laboratory and Weston Geophysical, it was concluded that the 84th percentile (specified by the NRC) of these spectra was similar to the Regulatory Guide 1.60 **[3.14]** shaped spectrum in the higher frequency range. The site specific response spectrum was conservatively assumed to have the shape of Regulatory Guide 1.60 spectrum anchored at 0.15g in that higher frequency range. The low frequency portion of the site specific response spectrum was controlled by the large, distant earthquakes. The lower frequency range of the DBE spectrum adequately reflects the influence of large earthquakes in the New Madrid area and in the Western Quebec seismic zone, both more than 300 miles from the Fermi 2 site.

The generation of the vertical site specific floor response spectra conservatively was based on multiplying the vertical DBE response spectra by a factor of 2.4.

3.1.1.1.3 Spatial Systems Interaction

During a seismic event, it is possible that the resulting displacements of plant components which are too close to each other could result in interaction between the two components, compromising the safety-related functional integrity of one or both of the components. In order to preclude such an occurrence, a rattlespace program was established to assure adequate distance between adjacent components and to justify existing, less-than-minimum distances between the components. During original plant construction, there was no rigorous criteria which was used to address proximity effects and separation issues between plant components. The purpose of this section is to describe the resulting program which was established to address this issue and seismic Category II/I criteria.

3.1.1.1.3.1 Non-Seismic-Category I Structures

The non-seismic-Category I turbine/radwaste building is adjacent to the seismic Category I auxiliary building. A four-inch gap separates the two buildings. The turbine building was originally designed in accordance with the Uniform Building Code (UBC) **[3.15]** Subsequently, it was seismically analyzed **[3.16]** by the response spectrum method to assure the seismic integrity of the main steam line valves in the turbine building.

Two hyperbolic natural draft cooling towers made from reinforced concrete are located north of the safety-related buildings. The towers act as large heat exchangers in which air blowing through the towers removes heat from the circulating water cascading down through the towers. The minimum distance between the cooling towers and Category I structures is about 800 feet.

Reinforced concrete structures and elements have been designed in accordance with the requirements of ACI 318-63 **[3.17]**.

3.1.1.1.3.2 Rattlespace Program

A rattlespace program was established in the early-1980s to address spatial interaction problems (i.e., to identify and resolve any rattlespace violations which occur between two interacting components, at least one of which is safety-related). Rattlespace is defined as the distance between two components in close proximity to one another. Bounding allowable rattlespace values were established **[3.18]** for different types of components. The minimum rattlespace between two components is the distance necessary to assure that the safety-related component maintains its structural and functional integrity during and after a seismic event. If the actual distance between the components is less than the required rattlespace, an evaluation is performed to justify the actual distance. In most cases, the actual distance is acceptable based on analytical displacements being smaller than the available rattlespace. In other cases, the effect of an impact between the two components is evaluated. If necessary, field modifications are implemented to prevent the components from interacting.

The first phase of the rattlespace program was implemented in the early- to mid-1980s, following plant construction. The purpose of the program was to identify and resolve existing rattlespace problems resulting from plant construction. From 1986 to the present, the focus of the rattlespace program has been to assure that adequate spatial separation is maintained during the implementation of plant design modifications and to resolve any newly discovered rattlespace violations in existing construction.

3.1.1.1.3.3 Seismic II/I Criteria

Non-safety-related components in safety-related buildings are designated as Seismic Category II/I. Components that do not have safety functional requirements but are located in safety-related buildings fall into this category. The continued functioning of these items is not required, but their failure could adversely affect the functioning of plant Category I items. Seismic category II/I components are either designed to maintain structural integrity under SSE excitation or it is demonstrated that their failure would not affect safety related components. Seismic category II/I items can be qualified by analysis, test, or a combination of both test and analysis. Most seismic category II/I items at Fermi 2 are qualified by analysis.

3.1.1.1.4 Seismic Qualification

The seismic qualification of the Fermi 2 power plant structures and components was divided into two main categories. The first was the analysis of the Category I structures. The second was the qualification of Category I systems and components housed in Category I structures. The Fermi 2 seismic qualification program is described in Section 3 of the UFSAR.

3.1.1.1.4.1 Structures

The seismic Category I structures include the primary containment, reactor/auxiliary building, and the RHR building. The structures were dynamically represented in lumped-mass stick models. Response spectrum analyses were performed on the models to determine displacements and accelerations at the mass points. Member forces were determined and distributed to the building walls and slabs in proportion to their stiffness and distance from the center of rigidity. The distributed forces were combined with other required design forces. The structural members were designed to adequately resist the appropriate design forces.

A brief description of these structures is provided below.

__**Primary Containment**__

The primary containment (Figure 3-11) is a leak-tight, steel-plate containment vessel consisting of a light bulb-shaped drywell and a torus-shaped suppression chamber. It houses the reactor vessel, recirculation system, and other primary systems. The drywell is enclosed in a reinforced concrete biological shield and is supported by the drywell pedestal. The basic purpose of the primary containment is to limit the release of fission products to the plant site environment, following a postulated design basis accident (LOCA), so that offsite doses do not exceed legal values.

The reactor/auxiliary building is a single structure enclosing both the reactor building (which includes the reactor), and the auxiliary building (which includes the auxiliary equipment and the control room). See Figures 3-12 and 3-13, respectively.

__**Reactor Building**__

The reactor building completely encloses the drywell and suppression chamber and is supported on a reinforced concrete foundation mat. The building provides secondary containment when the primary containment is closed and primary containment during reactor refueling and maintenance operations when the primary containment is open. The reactor building houses the refueling and reactor servicing equipment, biological shield, and new and spent fuel storage facilities. The building consists of poured-in-place reinforced concrete up to and including the refueling floor. Above the refueling floor, the building structure is steel-framed with insulated metal siding and has a pitch and slag roof over insulated metal.

Auxiliary Building

The auxiliary building is a poured-in-place reinforced concrete building, integrally connected to the reactor building by the common east wall of the reactor building. It is supported by the reinforced concrete foundation mat common with the reactor building. The building is adjacent to but separated from the turbine building by a four-inch seismic rattlespace. The auxiliary building houses the main steam tunnel, main control room, computer room, main battery rooms, switchgear rooms, main ventilation rooms, relay room, off-gas treatment rooms, CRD pumps, HPCI pump and turbine, and main power distribution center.

RHR Building

The RHR building is a poured-in-place reinforced concrete building designed to serve as the ultimate heat sink for the reactor during normal shutdowns and postulated accident conditions. It is supported by a reinforced concrete foundation mat. The building consists of two identical divisions. Each division consists of a water reservoir, pump house, two mechanical draft cooling towers, and two emergency diesel generators. Each division has the capacity to safely and orderly shut down the reactor during normal and/or accident conditions completely independent of the other. See Figures 3-14 to 3-16.

3.1.1.1.4.2 Equipment

Seismic qualification of Fermi 2 equipment was performed by one of three methods: analysis, testing, or a combination of analysis and testing.

Analyses used for equipment qualification were either static or dynamic. Static analysis was used for equipment characterized as a relatively simple structure. The static method involved multiplication of the component dead weight by the applicable accelerations from response spectra curves, including considerations for multi-frequency excitation and multi-mode response, and applying the forces at the component center of gravity. A dynamic analysis was used on equipment for which significant multi-mode response or cross coupling was anticipated or when the results from a static analysis were too conservative. A lumped mass or finite element model was developed and a response spectrum analysis or time history analysis was performed. For both the static and dynamic analysis methods, each of the three directions of earthquake was evaluated separately. The results were combined by the square root of the sum of the squares (SRSS) method.

Testing was used on complex equipment and equipment whose operability verification was required. For such equipment, qualification by analysis was insufficient to determine either its structural or functional adequacy. The dynamic qualification used for equipment and components which were classified as Class 1E, confirmed their ability to perform the required safety function and maintain structural and pressure boundary integrity during and after the postulated seismic forces. Testing was conducted by mounting the specimen to a rigid platform driven by hydraulic actuators to produce the required motion. Sinusoidal, sine beat, and random input tests were acceptable methods of seismic qualification based on the particular component location, structure, and floor response characteristics.

Equipment vendors and suppliers were required to develop test programs for qualifying their equipment in accordance with the conditions specified in the earthquake design requirements. Qualification reports were prepared and submitted by the vendors or suppliers to document the equipment's seismic adequacy.

Equipment and components were grouped into two general categories: NSSS and balance of plant (BOP). The NSSS vendor (General Electric) was responsible for all related components and their seismic qualification, as required. In general, NSSS items were qualified using generic qualification procedures and criteria. For GE-supplied equipment qualified by testing, IEEE 344-1971 **[3.19]** was the applicable standard for all electrical equipment purchased before the issuance of IEEE 344-1975 **[3.2]**. The seismic requirements for Fermi 2 safety-related BOP equipment are governed by either Fermi 2 seismic qualification Specification 3071-296 **[3.21]** or the equipment design specifications.

3.1.1.1.4.3 Distribution Systems

Seismic Category I distribution systems at Fermi 2 include piping, cable trays, electrical conduits, and HVAC ductwork. Various methods of seismic qualification were used for the systems and their supports.

NSSS Primary Coolant System

The Nuclear Steam Supply System primary coolant system at Fermi 2 consists of the reactor pressure vessel, the two recirculation loops with the recirculation pumps and valves, and the main steam piping lines from the RPV to the first isolation valves.

The seismic loads on the RPV and internals were based on the dynamic analysis of the reactor/auxiliary building. A mathematical model was developed to represent the RPV, RPV internals, reactor pedestal, and sacrificial shield wall. The model consisted of lumped masses and springs to idealize the inertial and stiffness properties of the system. Figure 3-5 shows the RPV mathematical model. The seismic analysis was performed by a modal superposition time history analysis. The design of the RPV and internals met the stress criteria of Section III of the ASME code.

The recirculation and main steam piping systems were elastically analyzed in accordance with the requirements of ASME Section III **[3.22]** and were constructed per the ANSI B31.7 Nuclear Power Piping Code **[3.23]**, Class 1 requirements.

Piping

Category I piping was seismically analyzed by either a simplified analysis or a multi-degree-of-freedom analysis depending on its quality group and nominal size. The simplified method was generally used for field designed piping and typically for ASME Class 2 or 3 piping of size two inch and under with design temperature of 575°F or less. For dynamic analysis, the lumped mass, response spectrum method was used. The responses from different modes were combined in accordance with Regulatory Guide 1.92 **[3.24]**. Damping values used were 0.5 percent for OBE and 1.0 percent for SSE. ASME piping was analyzed and qualified to the requirements of the ASME Boiler and Pressure Vessel Code (BPVC) **[3.22]**. In the re-analysis of selected piping systems for snubber reduction, ASME Code Case N411-1 **[3.25]** damping values were applied in accordance with Regulatory Guide 1.84 **[3.26]**. Non-ASME-BPVC piping was analyzed to the requirements of ANSI B31.1 **[3.27]**. Each pipeline was idealized as a mathematical model consisting of lumped masses connected by elastic members.

Supports for Category I piping systems were analyzed and designed to withstand the resulting pipe loading from the piping analysis. Loading conditions were appropriately considered in accordance with applicable ASME code sections. Pipe supports generally met the stress requirements of the AISC Specification for Structural Steel Buildings **[3.28]**.

Cable Trays

The cable trays at Fermi 2 are either of the ladder-bottom type or solid-bottom type with covers. Tray widths vary from six to 36 inches and outside depths from three to six inches. The ladder-bottom cable trays are a prefabricated sheet metal structure consisting of two galvanized channels placed face-to-face and connected transversely by hat section members spaced at about nine-inch centers. The cable trays are rigid enough to support their own weight and other design loads for about eight feet; therefore, cable tray hanger type supports are placed at about eight-foot intervals.

The trays were analyzed and designed for combinations of deadload, live load, and seismic loads. Their design was based on the Specification for the American Iron and Steel Institute (AISI) Design of Cold-Formed Steel Structural Members **[3.29]**. For dynamic seismic analysis, the cable trays and supports were modeled as a multi-degree-of-freedom system in which the mass of the cables and tray are lumped at their support level. The response spectrum method of analysis was used. The response spectra curves were obtained from the building analysis. For both horizontal and vertical excitations, ten percent damping for DBE was used in the design.

Electrical Conduits

The electrical conduits used at Fermi 2 are either rigid or flexible. Rigid conduit consists of galvanized steel pipe sections connected by couplings. They are supported at about eight-foot intervals to prevent any failure due to the combined dead weight and seismic excitation of the conduits and conduit support system. Seismic loads for the conduit support design were generally obtained by selecting the peak accelerations from the response spectra curves at the support elevation or higher and multiplying the accelerations by 1.5 to account for multi-mode response. Damping values of two percent were used for OBE and five percent for SSE. Flexible conduit is used to route and support electrical cable between cable trays and rigid conduit and between rigid conduit and pad-mounted electrical equipment. Conduit support steel members met the stress criteria of the AISC Specification for the Design of Structural Steel for Buildings.

HVAC Ductwork

HVAC ducts used at Fermi 2 are either circular or rectangular in cross section. They are constructed from galvanized sheet steel to a specified gauge thickness and stiffness. The ducts are supported at specified spacing to restrain their movements. Duct design criteria was based on buckling of the duct walls under axial loads and bending. The ducts were designed for the combined loading of dead weight plus the seismic load. Temperature and pressure loading were also addressed for the drywell ducts. Seismic analysis of the ducts was either the simplified static or dynamic type. The dynamic analysis was performed using the modal response spectrum method in which the duct was considered as a series of lumped-masses connected by mass-less elastic members. Damping values used were two percent for OBE and five percent for SSE. Modal responses in the three seismic input directions were combined by the SRSS method. The simplified analysis approach considered the duct system as a simply supported or continuous beam model supported by hangers. Duct hanger steel members met the stress criteria of the AISC Specification for the Design of Structural Steel for Buildings.

Instrument Tubing

Instrument tubing was either analyzed using the same computerized stress analysis techniques used for large bore piping, per ASME Section III, Class 2 or 3, or by using simplified analysis. The simplified technique was based on the use of pre-developed design tables which limit stresses and deflections of the tubing to conservative allowable limits. Tubing supports were designed per the AISC code requirements.

3.1.1.1.4.4 Site Specific Earthquake Reassessment

Detroit Edison performed a detailed reassessment **[3.30]** of the structures, systems and components in one safe shutdown path using the site specific earthquake response spectra. Two acceleration time histories, north-south and east-west, matching the 5%

damped site spectrum were generated for use as horizontal forcing functions. These time histories were used to generate response spectra which enveloped the ground response spectra discussed earlier. The time histories were then used to generate internal equipment floor response spectra upon which equipment validation was based.

The structural reassessment was performed using the same models as were used in the original seismic dynamic analyses. Detailed analyses were performed on selected structures in the drywell, reactor/auxiliary building, and RHR building. Examples of such structures included the reactor pedestal, sacrificial shield wall, biological shield wall, spent fuel pool, mat foundation, shear walls, superstructure steel, cable trays and hangers, and torus and torus supports.

The reassessment of equipment was based on the selection of a scenario, based on loss of offsite power, characterized by early automatic control of reactor level and pressure by the RCIC and SRV systems, respectively. Following stabilization of reactor vessel level and pressure, operator action was assumed to be taken to cool down and depressurize the reactor. A list of the systems necessary to shut down and cool down the reactor, and the requisite components within the systems, was developed. Principal systems included RCIC, NSSS, RHR (Division 2), and CRD. auxiliary systems included RHR service water, EDG, EECW, EDG service water, control air, Control Center HVAC, drywell cooling, and EDG ventilation. Essential equipment included drywell coolers, room coolers, cable trays, conduits, I & C tubing, motor control centers, switchgear, relay room and control room panels, batteries and chargers, diesel generators, underground electrical ducts and piping, valve-operators, pumps and motors, control center ceiling and lights, and valves.

The effects of the postulated site specific earthquake on plant piping systems were evaluated by performing detailed analyses of large-bore piping and I & C piping and tubing, and by generic analysis of small-bore piping and Class 1E conduits. The results of the evaluations demonstrated that these systems have the capability of withstanding the defined site specific earthquake and the ability to subsequently support a cold shutdown of the plant.

These supplementary site specific evaluations reaffirmed the original facility seismic design basis acceptability.

3.1.1.2 Seismic Margin Screening

The EPRI seismic margin methodology utilizes a screening approach to eliminate certain elements from detailed seismic evaluation and to enable the evaluation to concentrate on those elements that may potentially become the "weak links" among the safe shutdown success path components. Screened-out elements are considered adequate to withstand the 0.3g RLE. By minimizing efforts in the evaluation of the components that meet the screening guidelines and concentrating on the ones that may eventually describe the plant HCLPF, the seismic margin evaluation can be completed with reasonable cost and without the risk of overlooking a "weak link" component that could affect the outcome of the plant seismic capability.

Tables 2-3 and 2-4 of EPRI NP-6041 were used as the main source of screening guidelines for the Fermi 2 seismic margin assessment. However, all components on the safe shutdown equipment list (SSEL) were walked down regardless of whether they were considered screened out or screened in. In some cases, the walkdown observations resulted in some concerns for items that met the screening rules but had other anchorage or spatial interaction concerns that required detailed evaluations. Further discussion on structure and component screening is presented in Section 3.1.4.4.

3.1.1.3 Plant Walkdown

A detailed walkdown of the selected structures and components is by far the most important step in the seismic margin program. All the steps preceding the walkdown are preparation steps for this activity. Experience with earthquake damage indicates that thorough field examination, in accordance with prescribed procedures, can detect most seismic weaknesses and vulnerabilities in the plant.

The purpose of the walkdown is to observe the actual in-situ condition of the components and, as appropriate, to screen as many components as possible from the margin evaluation effort, based on the earthquake experience data and the collective seismic experience and judgment of the SRT. Additionally, seismic interaction concerns and the most likely failure modes for items that do not screen out are identified for follow-up work and further evaluations.

The Fermi 2 SRT recognized the importance and significance of the walkdown activity in the margin program; therefore, concentrated effort was expended in preparation for the walkdown to gain the maximum benefit possible during the field detailed inspection. The walkdown focused on equipment anchorage, spatial interactions and component functional capability. More detailed description of the seismic capability walkdown effort is presented in Section 3.1.4.5.

3.1.2 System Analysis

This section presents the systems analysis and component selection process used to develop the safe shutdown equipment list (SSEL) for use in the seismic margins assessment. The systems analysis identifies a finite set of systems that can be used to successfully mitigate a seismic event. As required by NUREG-1407, systems that represent a primary and an alternate shutdown path were selected. To minimize the extent of walkdowns inside of containment, one shutdown path was selected with the capability of coping with a small, seismically induced LOCA. Both shutdown paths include systems that will provide the necessary shutdown functions. These functions are reactivity control, reactor vessel pressure control, reactor coolant inventory supply and decay heat removal. Following selection of the safe shutdown paths, components that are essential for system operability were identified for each system in the shutdown paths. Included in this selection process were systems needed to support the front-line systems.

3.1.2.1 Safe Shutdown Path

The seismic margin assessment requires the identification of systems needed to maintain a safe shutdown condition following a seismic margin earthquake. A preferred and an alternate success path were identified based on operational and system considerations. One of the selected paths must be able to cope with a small, seismically induced leak since there was no attempt to rule out such leakage by plant walkdowns. Implicit in the selection of the safe shutdown paths is that only paths operators are apt to use, are based upon procedures and training, and will have instrumentation and indication available following the seismic event are considered.

The first step in selecting a safe shutdown path was the identification of the front-line systems needed to provide the four safety functions necessary to establish a safe shutdown condition. Once the essential front-line systems were identified, systems needed to support the operability of these front-line systems were determined. Finally, support systems required for the operability of the support systems were identified.

3.1.2.1.1 Background

The success path logic diagram (SPLD) was assembled by a Risk Analysis engineer familiar with the Fermi 2 Internal Events Individual Plant Examination (IPE) **[3.31]**. Information obtained during the IPE effort was used in the selection of the systems in the SPLD. The diagram identifies a finite set of systems that can be used to establish a safe shutdown condition given a seismic margin earthquake (SME). Both front-line and support systems are included in the SPLD. To assess the availability of these systems, flow paths had to be determined and the operability of each component in the flow paths had to be evaluated. The flow paths and the required components are normally obtained by reviewing system P&IDs. To obtain a list of the safe shutdown components for Fermi 2, Functional Operating Sketches (FOSs) were used instead. The FOSs are the preferred drawings of operations personnel and include more familiar valve nomenclature. The method used to generate the list of safe shutdown components follows.

3.1.2.1.2 Methodology

Flow paths required for system success were highlighted on the appropriate FOS drawing. Components on these success paths were then categorized as to the need of verification of their ability to withstand a SME. Those that require a seismic evaluation were highlighted and placed on a list called the safe shutdown equipment list (SSEL). Components that are not considered to be vulnerable to a SME are called rugged. Rugged components were uniquely highlighted on the FOS, but normally were not included on the SSEL. Table 3-1 lists the type of items that were considered to be rugged at Fermi 2.

In addition to the rugged items, motor operated and air operated valves that are not required to change state were typically left off the SSEL.

3.1.2.1.3 Assumptions

EPRI NP-6041 provides ground rules or analytical constraints to allow study of seismic vulnerability using a deterministic SMA instead of a PRA. One of these constraints is that path success is defined as the ability to achieve and maintain a stable hot or cold shutdown condition for at least 72 hours following the seismic event. Following is a list of additional assumptions that went into the development of the Fermi 2 safe shutdown paths and the selection of equipment in the safe shutdown list:

- A 0.3g seismic event occurs. This is twice the Fermi 2 design basis earthquake.
- Due to the seismic event, all off-site power is lost for the duration of the evaluated scenario. This includes the black-start combustion generator since it is not seismically qualified.
- A demand for all four emergency diesels to start and run for 72 hours is received by the diesels.
- The seismic event may result in small line breaks inside of the primary containment. Total leakage, however, is limited to the equivalent of a one inch diameter line break (i.e., slow inventory reduction that does not depressurize the reactor vessel).
- Since RPS is fail-safe, a scram signal is present; however, the hydraulics and CRD housing still must be assessed.
- All injection systems (HPCI, RCIC and LPCI) considered in the safe shutdown path are required to have the capability to start automatically. Standby feedwater, condensate, CRD flow and core spray are not credited in the IPEEE seismic evaluation.
- If necessary for LPCI injection, the operator will manually depressurize the reactor vessel using the SRVs that are equipped with accumulators.
- Operators will align and operate the RHR system in either the torus cooling or shutdown cooling mode. For the types of transients that are expected from a seismic event, decay heat removal could be deferred during the first eight hours of the transient. This is partially based on the Modular Accident Analysis Program (MAAP) results reported in the Fermi 2 IPE submittal **[3.31]** and a review of the Net Positive Suction Head (NPSH) data provided in the Fermi 2 emergency operating procedures (EOPs) that indicate several feet of NPSH would still be available at suppression pool temperatures up to 240°F.

- Equipment in the shutdown path which is expected to survive the effects of an earthquake is assumed to be available for mitigation of the seismic event. This is reasonable since, with the exception of HPCI and RCIC, system success relies on one of two divisions being functionally operable. However, HPCI and RCIC are single train systems and are known to have low availability factors. Due to their poor availability, the high pressure injection function is assumed to require both systems. No other unusually high system or human error rates related to the selected safe shutdown paths were identified in the Fermi 2 IPE **[3.31]**.

- Even though Abnormal Operating Procedure NP 20.300.03 **[3.32]**, "Loss of Offsite Power," provides for restoration of MPUs 3 and 4 through essential busses following a loss of offsite power, no credit was taken for these BOP power supplies.

- The control room alarm system is assumed to remain operable after the seismic event. It is anticipated that a seismic event would result in several spurious alarms. While confirming a plant trip and assessing the status of key plant parameters, the operators will acknowledge and reset/clear both valid and spurious alarms. It is assumed that these actions will correct the consequences of any spurious alarms and that no further actions would be necessary.

- Since the logic circuits of the load sequencer do not include any "bad actor" relays, as confirmed in the relay screening evaluation, it is assumed that the automatic shedding and loading of AC power operates as designed.

- Since a large break LOCA is not postulated, successful operation of the LPCI loop select logic is not necessary for this scenario. Failure of the LPCI loop select logic will not prevent a LPCI injection path to the reactor vessel from being established.

- Reactor building HVAC is not required since EECW will provide cooling to essential equipment. However, operation of the control room HVAC and EDG HVAC is required.

3.1.2.1.4 Success Path Determination

Success path logic diagrams identify systems which can perform the four safety functions that are necessary for establishing and maintaining a long-term safe shutdown condition. The four safety functions are categorized as:

- reactor reactivity control
- reactor coolant system pressure control
- reactor coolant system inventory control
- decay heat removal

The SPLD displays the safe shutdown systems by means of horizontal paths. To ensure a high degree of success, the paths selected use systems that would normally be considered by plant operators based upon procedures, training and available instrumentation and indicators. Front-line systems that perform the safety functions and support systems required for operation of the front-line systems are identified in the SPLD.

For the seismic margin assessment two major assumptions dictate the selection of the safe shutdown systems. The first assumption is that the seismic margin earthquake would result in the loss of off-site power. This assumption requires that all equipment needing electrical power should be capable of being powered by an emergency diesel generator or the essential batteries. The second assumption is that the SME may result in small line failures comparable to a small LOCA. Since a small LOCA is conceivable, systems selected for the SPLD for at least one shutdown path should have the capability of handling this event. To satisfy the SMA requirements, two success paths were selected. Each path has the ability to achieve and maintain a stable shutdown condition for at least a 72-hour period following a SME. The path involving reactor depressurization and LPCI is capable of mitigating the consequences of a small LOCA for at least 72 hours. Both paths can handle the small LOCA during the early stages of the event before depressurization as a result of the LOCA itself occurs. As discussed in Section 6.2.1.3 of the Fermi 2 UFSAR, the heat load due to a small LOCA in the drywell could be adequately handled by the torus cooling mode of RHR. Drywell sprays would not be necessary to maintain the containment within design limits and; therefore, are not included as part of the SPLD.

It should be noted that, due to the potential stress imposed on plant operators by a SME, the selected systems should respond automatically, at least in the short term. The systems on the SPLD are capable of automatic operation. To be assured of automatic operation, the transmitters necessary for system initiation are included on the safe shutdown equipment list.

3.1.2.1.5 Preferred and Alternate Paths

The Fermi 2 SPLD showing both the preferred and alternate success paths is shown in Figure 3-17. Because it is desirable to avoid a deliberate depressurization, the preferred group of systems includes high pressure makeup and is given by the upper path. This path assumes there is no LOCA that would depressurize the vessel within the 72 hour mission time. An alternate set, relying upon low pressure makeup, is represented by the lower path.

The preferred path has RCIC and HPCI in series rather than in parallel. This is the case even though success could be accomplished with either system. These two systems are single train systems and are only moderately reliable. Placing them in series provides reasonable assurance that their safety functions (pressure and inventory control) will be available.

Suction for both HPCI and RCIC is assumed to be from the torus. Normally HPCI and RCIC are aligned to take suction from the condensate storage tank (CST). However, on low CST level both systems are designed to automatically switch their suction to the torus. It is assumed that a SME would fail the CST. The failure would result in loss of inventory in the tank and automatic transfer of HPCI and RCIC suction to the torus. Since the transfer logic would be required for system success, the logic required a seismic evaluation. However, neither the suction line nor the CST is required for system success; therefore, they were not evaluated.

Included in a draft version of the Fermi 2 SPLD were the standby liquid control system, core spray and the drywell sprays. Page B-3 of EPRI NP-6041 contains the following remark regarding the standby liquid control system:

> "*...emergency boration in response to an anticipated transient without scram event is not considered an acceptable means of reactivity control following an SME due to the added stress imposed on the plant operators.*"

The above statement implies that the standby liquid control system should not be a part of the SPLD. Because the standby liquid control system is a manually initiated emergency boration system, successful operation of the system is of questionable value following a SME. Successful operation of the system is questionable because other tasks requiring immediate action following a SME would compete for the operator time and may prevent the system from being initiated. For this reason, the standby liquid control system was removed from the SPLD.

Low pressure makeup can be provided by either the LPCI mode of RHR or core spray. Due to the diversity of the LPCI mode of RHR (two flow paths, each path with two pumps), the additional redundancy provided by the core spray system is not needed. Similarly, drywell sprays are a redundant system to suppression pool cooling. The major difference between drywell sprays and suppression pool cooling is the injection path to the primary containment. In addition, due to the number of shared components, if the torus cooling mode of RHR was not available, it is likely that the drywell sprays would also be unavailable. Thus, to simplify the SMA and because it does not significantly alter the success of either path, both the core spray system and drywell sprays were removed from the final Fermi 2 SPLD.

3.1.2.1.6 Primary and Support Systems

To establish support dependencies for the systems in the Fermi 2 SPLD, support-to-support and support-to-front-line dependency matrices were generated. These dependency matrices identify support requirements for both the front-line and the support systems.

One of the support systems identified in the SPLD is the emergency equipment cooling water system (EECW). This system includes components in the drywell. Although the drywell equipment cooled by EECW is not required for a safe shutdown, isolation capability of EECW drywell loads still requires a seismic evaluation.

The systems selected for the four safe shutdown functions are described in Table 3-**3-2**. This table gives a brief description of the systems selected for both the preferred and alternate success paths.

3.1.2.2 Safe Shutdown Equipment List

The SSEL (Table 3-3) is a list of the equipment required to achieve and maintain a safe shutdown condition given a SME. This list includes mechanical and electrical equipment which should operate to accomplish a safe shutdown function. Equipment such as tanks, heat exchangers, and instrumentation needed for system operation and to confirm plant status is also included on the SSEL. Each component on the SSEL is uniquely identified by its Plant Identification System (PIS) number and a Central Component (CECO) data base sub-unit number. In addition, the following information is stored on the SSEL:

- Location of equipment (building, floor, elevation, and grid coordinates);
- Manufacturer, model, and supplier of equipment;
- Description of equipment including an alternate identification;
- Seismic report number for the equipment and its Fermi 2 file number;
- Number of the drawing on which the equipment can be found;
- QA level; and
- Any other notes pertaining to the equipment or seismic evaluation

An extensive search for relays and equipment that may be affected by relay chatter was not performed during the generation of the Fermi 2 SSEL. This task was completed independently of the SSEL by the Electrical and I&C groups. The steps for screening potential "bad actor" relays were established and are discussed in Section 3.1.2.3. Since this effort was separate from the equipment selection process, relays are not included in the SSEL.

3.1.2.2.1 Component Selection

Information from the SSEL is used to generate a list of equipment that needs to be evaluated during the seismic walkdown. Section 3.3.3 of the GIP **[3.33]**, discusses a concept called the "rule-of-the-box" which allows some leeway in the detail required for the walkdown list. The Fermi 2 SSEL uses a modified version of the "rule-of-the-box". The Fermi 2 database contains a unique PIS and sub-unit number for each component. Therefore, most equipment requiring an evaluation is included as a separate line item in the database. For example, rather than having a motor-operated valve stand for all sub-components of the assembly, the following components are listed separately:

- motor
- motor operator
- valve

The "rule-of-the-box" would have had all of these items identified as a single component. In the process of identifying assembly sub-components from the CECO database, certain sub-units were deliberately excluded either because of their generic seismic ruggedness (e.g., wires, couplings, seals, gaskets) or because they were included in a separate evaluation scope (e.g., relays and switches).

The detail provided on the SSEL lengthened the list but also increased the confidence that most, if not all, of the relevant components were evaluated. A listing of the components that were evaluated during the seismic walkdown is attached as Table 3-3. This list was obtained from the SSEL database by selecting unique PIS numbers only. CECO sub-units are not individually identified in this table.

Valves that provide containment isolation are identified in Table 6.2-2 of the Fermi 2 UFSAR. From this table, isolation valves that are normally open and that are required to close during a shutdown or an accident condition were selected and added to the SSEL. Of the valves added, those that fail closed upon loss of their power source were uniquely identified on the SSEL. Additional details of the containment isolation valves are included in Section 3.1.2.2.4.

3.1.2.2.2 Instrumentation Selection

A review of System Operating Procedures (SOPs) was made for instrumentation that is required during system operation. An assessment was performed which identified electrical components necessary for operation of the identified instruments. Both the instruments and the support components were added to the SSEL.

Some instrumentation that is not identified in the SOPs was included on the SSEL but was noted as only requiring an assessment for pressure boundary integrity. These instruments were later excluded from the SSEL walkdown list based on the fact that earthquake experience data does not include pressure boundary failure of instruments due to seismic excitation. Additionally, these instruments are located in the same areas and cabinets where other SSEL components are located; therefore, the general walkdowns would identify any spatial interaction or other unique problems these instruments may have. Instrumentation required for the automatic operation plus permissives for system operation was also identified and included on the SSEL.

All Division 1 and 2 MCC panels were added to the SSEL. Since MCCs are similar to each other, including all of them on the list did not impose a burden on the walkdown effort. Relay panels were also added to the SSEL. Panels containing relays or other electrical equipment were selected on a system basis. A CECO search for panels associated with systems on the SPLD was performed. Panels identified during the search were added to the SSEL. CECO also contains a "mounted on" field for the components. All panels identified in the "mounted on" field were also added as separate items on the SSEL.

The equipment on the SSEL is grouped by system or functional categories. A line item number places each component into one of these groups. Each system/function and its corresponding line numbers are shown on Table 3-4.

3.1.2.2.3 Instrumentation Dependency

Instrumentation selected during the system and operations review were given to the I&C group for the identification of dependencies. The group review identified all components that could interfere with the operability of the selected instruments. These components were added to the SSEL. For example, an indicator in the control room may be dependent on a power supply, an intermediate instrument to process a signal, and a local sensor. All such items, including any power supplies required for their operation and any isolation devices to prevent circuit malfunction, were added to the SSEL.

3.1.2.2.4 Containment Function Components

In addition to establishing a safe shutdown condition, the requirements for the seismic IPEEE call for the successful isolation of the primary containment. Successful containment isolation would mitigate the consequences of a radiological accident. However, containment isolation would only be necessary if the other components on the SSEL failed to perform as expected. If the components on the SSEL operate successfully, core damage would be prevented and there would not be a radiological release.

It is assumed that successful containment isolation would be assured if all of the normally open valves that receive an isolation signal would close. Several of these valves are designed to close given the loss of power. The power source for these valves need not be evaluated since failure of the power supply would lead to closure of the valves. However, the valves themselves still require an evaluation to assure closure. Table 3-5 lists the normally open valves that need to close upon the receipt of an isolation signal. This valve information is based on Table 2.2-2 of the UFSAR. Two of the valves on the list are not on the SSEL. These valves (PIS numbers E4150F079 and E5150F084) are similar in one respect. They are isolation valves in series with a second valve on lines that discharge to the torus below the water line. The valve in series with the excluded valve is on the SSEL. This configuration in itself was assumed to be adequate isolation for purposes of this study. Thus, the two valves above were not placed on the SSEL.

Each isolation valve on the SSEL receives one or more isolation signals. It is assumed that any one of the signals would result in isolation (closure) of that valve. All of the isolation valves would receive one or more of the following isolation signals:

- Reactor Vessel Low Level 1
- Reactor Vessel Low Level 2
- Reactor Vessel Low Level 3
- High Drywell Pressure

The components corresponding to these signals are already on the SSEL and are evaluated as part of the Emergency Core Cooling Systems (ECCS) actuation instrumentation. Since at least one signal received by each isolation valve came from an instrument that was already evaluated, no additional components were added to the SSEL. Containment isolation is assured with the successful operation of the equipment already on the SSEL.

Instrumentation providing indication of containment status was not considered to be necessary for maintaining the integrity of the plant. If containment indication is not available, it is assumed that operators will take the conservative course of action based on available plant information and operate those systems that are available to mitigate the consequences of the event. Operation of one division of RHR in the torus cooling mode and the other division in the shutdown cooling mode would prevent core damage for the postulated IPEEE scenarios.

Review of the selected instrumentation by operations resulted in some containment instrumentation being added to the SSEL. The instruments provide temperature indication for both the drywell and the torus. Additional details regarding the containment performance evaluation are found in Section 3.1.6.

3.1.2.2.5 Plant Operations Review

The plant operations organization performed a review of the SSEL. This review, in addition to ensuring that the shutdown paths were consistent with operating procedures, focused on assuring that components and instrumentation on the list were adequate f system monitoring and functionality. Given the systems selected for the safe shutdown paths, operation personnel identified the components and instruments needed to operate the systems. Their list was then compared to the SSEL. The result of the initial review was a list of potential components to be added to the SSEL.

The operations list was further reviewed to identify components and instruments that were not essential given the assumptions for the SME. In the end, about 50 components were added to the SSEL as a result of the operations review. Most of the components added were instruments used for monitoring various system parameters.

The following is a brief chronological list of internal correspondence documents that describe the operations review process:

- Reference **[3.34]** initiated the request for operations review of the SSEL. Operations delegated the responsibility to the Shift Technical Advisor (STA) group. The assigned STA reviewed the SPLD, the SSEL, and the marked-up FOS drawings that were used in the component selection process. Other STAs and licensed operators were consulted as needed. The STA provided preliminary comments on the SSEL.
- Reference **[3.35]** documents the disposition of the operations preliminary comments by the SMA system engineer.

- Reference **[3.36]** provides operations concurrence with the comment disposition in Reference **[3.35]**.
- It was later realized that certain components recommended for addition to the SSEL would not be available to the operators given the loss of offsite power (LOOP) scenario assumed in the SMA; therefore, as documented in Reference **[3.37]**, alternate components were used when available. Otherwise, the components were dropped from the list if no alternates exist and the components are not absolutely necessary.
- As a result of the I&C review outlined in Section 3.1.2.2.3, additional components were included on the SSEL to address instrument dependency and other miscellaneous concerns. Reference **[3.38]** documents the components added as a result of this review.

3.1.2.3 Low Ruggedness Relay Screening

According to NUREG-1407, non USI A-46 nuclear plants performing a "Focused Scope" margin study may complete the relay evaluation part of the study by locating and evaluating low-seismic-ruggedness (bad actor) relays, as identified in EPRI report NP-7148-SL **[3.39]**.

The relay evaluation performed as part of the seismic IPEEE program at Fermi 2 utilized information available in the plant Central Component (CECO) database to identify low-seismic-ruggedness relays. All plant safety-related systems were included in the relay screening task. The use of existing database information was chosen over the EPRI NP-7148 methodology because it provided direct access to relay information as identified by manufacturer's name and model number. However, once bad-actor relays were identified, control schematic drawings were reviewed to evaluate the impact of contact chatter on system functions. The scope of the review was divided into two parts: electrical and control.

3.1.2.3.1 Electrical Relays

The scope of the electrical systems relay evaluation included the review of motor control centers, low and medium voltage switchgears, the emergency diesel generator and the fire protection system. The fire protection system was included in the scope of the review to identify any impact on operability of the HVAC systems due to system interfaces.

The Central Component (CECO) database was used as a main source for the relay search. CECO is a site-wide database that contains information on plant components as identified by a unique plant identification system (PIS) number and a sub-unit number. Safety-related relays and switches in the electrical scope are uniquely identified in CECO. In addition to CECO, vendor manuals, wiring and schematic drawings were also reviewed, as required, to locate bad-actor relays and contact devices.

Appendix E of EPRI NP-7148 also identifies mercury switches and sudden pressure switches in the list of low-ruggedness relays. Sudden pressure switches are generally associated with electrical system service transformers. Given the loss of offsite power scenario assumed for the IPEEE, locating these switches was not considered applicable to the evaluation. Mercury switches were located by searching the CECO database for commonly known manufacturers of mercury switches and by reviewing drawings, vendor manuals, and other pertinent documents.

As a result of the screening described above, the only low-ruggedness relays found in the electrical systems are the following:

1. Four "Westinghouse" type SV-1 relays are used in the voltage sensing circuit of the emergency diesel generators (EDGs). One of these relays is located in each of the four EDG control panels in the RHR building. Deviation event report (DER) number 95-0104 **[3.40]** was initiated to address the operability of the system and the long-term corrective action required. Based on the disposition of this DER, the relays will be replaced. Technical service request (TSR) number 27,566 **[3.41]** was approved to search for and design a suitable replacement for the SV-1 relays. An engineering design package (EDP) with the same TSR number will be developed to facilitate replacement of the relays.
2. Two "Westinghouse" type SG relays are installed in the fire protection and miscellaneous A. C. relay cabinet H11P852 located on the second floor of the auxiliary building. Evaluation of the circuits associated with these relays indicates that relay contact chatter would result in spurious fire alarms in the fire control panels in the relay and control rooms. However, there is no effect on the operability of any system; hence, no further action was required regarding these two relays.

No other bad-actor relays, contact devices, or mercury switches were found in the electrical systems reviewed; therefore, with the exception of the "Westinghouse" SV-1 relays described above, all electrical systems are considered acceptable for potential relay chatter effects.

3.1.2.3.2 Control Relays

The CECO database was also used as a main source for locating bad-actor relays in the controls and instrumentation scope of review. Special reports were prepared listing all safety related relays in the database with their manufacturer's name and model number. All relays matching one of the EPRI bad-actors listed in Appendix E of NP-7148 and located in one of the SSEL systems were selected. The primary SSEL systems are HPCI (E41), RCIC (E51), and RHR (E11). The support systems are EECW (E1156), RHRSW (P45), NIAS (P50), CCHVAC (T41), and RHR HVAC (X41).

Matching of the CECO relays to the bad-actor list resulted in the identification of two types of low-ruggedness relays. Sixty-three Westinghouse Type SG relays and 151 GE Type HGA relays were located. Four of the HGA relays were identified by reviewing vendor part lists and schematic drawings. The EPRI list specifically identifies the closed contacts in de-energized operating mode as the only case of low-ruggedness behavior for both of these relays. Therefore, it was necessary to review the control schematic drawings to determine the function associated with the normally closed contacts. The function of the relays was categorized in one of four impacts:

1. Alarm interface;
2. Sequence of events recorder interface;
3. Flasher interface (i.e., contacts that provide on-off power for control panel blinking lights when the associated device is not in service); and
4. Control interface (i.e., contacts that provide logic or permissive function).

The chatter of contacts associated with the annunciator (alarm) system does not have a significant impact on the plant and system operation. The associated alarm would clear at the end of the seismic event. Similarly, chatter associated with the sequence of events recorder is not significant for the plant operation. It would only result in a false log in the recorded event. The third category is chatter associated with contacts that interface with the flasher bus power. This chatter would result in an interruption of power to the control switch blinking light for the duration of the earthquake; therefore, it has no significance on the operation of plant systems. The last category is for contacts used in logic circuits. Three Westinghouse SG and ten GE HGA relays were identified in this category. The circuits were reviewed in detail to determine the effect of contact chatter during a seismic event **[3.42]**. It was concluded that none of these relays would cause a control system operation malfunction; therefore, all identified bad-actor relays were considered acceptable for the SMA review.

The search for mercury switches in the control circuits did not reveal any such devices. The search was accomplished by a combination of reviews of CECO reports, restricted engineering component list, and other system permissive instrumentation literature.

3.1.3 Analysis of Structure Response

EPRI NP-6041 provides two alternatives for the development of in-structure response spectra associated with the RLE: generation of new demand spectra or scaling of available SSE spectra. While scaling of available response spectra provides a very economical way to generate the new RLE demand, it generally results in more conservative spectra. The sources of conservatism include the artificial time history used in the analysis, the damping values, and the broadening and smoothening of response spectra peaks. On the other hand, generating new spectra can be more costly, but is expected to provide savings in the component evaluation and re-qualification process because of the less conservative spectra resulting from the new dynamic analysis of the buildings.

Design basis in-structure response spectra at Fermi 2 were generated by performing time history analyses on mathematical models that consider the building mass and stiffness distribution, including torsional effects **[3.43** and **3.44]**. Accordingly, EPRI NP-6041 suggests spectra scaling as a reasonable alternative worthy of consideration. Therefore, it was initially planned to generate SMA spectra by scaling the DBE and the Site Specific Earthquake spectra. The use of both sets of spectra would provide adequate data points for interpolation between the flat-shaped spectrum used in the DBE and the Regulatory Guide 1.60 shaped Site Specific Earthquake, and it would allow for structural damping adjustments. After additional considerations, it was decided to generate new in-structure spectra by performing new seismic analysis of the reactor/auxiliary and the RHR buildings. This decision was made because the results of preliminary scaling studies showed high amplifications. Generation of new spectra would remove the conservatism associated with scaling available spectra and provide more realistic results for the SMA.

3.1.3.1 Seismic Margin Earthquake Selection

One of the first elements to be determined for performing a seismic margin assessment is the selection of the seismic margin earthquake (SME). As stated in EPRI NP-6041, the SME should be set sufficiently high so that some plant elements in the success path have capacity levels less than the SME level. On the other hand, it should not be set too high such that it will result in too many outliers and substantial increase in the workload. The seismic margins methodology was designed to demonstrate sufficient margin over the plant design basis earthquake to ensure plant safety and find any "weak links" that might limit the capability of the plant to safely withstand a seismic event bigger than the DBE. The objective is to select a level for the SME which would ultimately result in finding the actual plant seismic margin.

In NUREG-1407, Fermi 2, like most other plant sites in the Central and Eastern United States, was assigned a peak ground acceleration (PGA) review level earthquake of 0.3g. Therefore, since all Fermi 2 safety related structures are founded on bedrock, the seismic margin program for Fermi 2 utilized a NUREG/CR-0098 median rock spectrum anchored at 0.3g for the horizontal free field excitation. The vertical motion was considered equal to two-thirds of the horizontal motion.

3.1.3.2 Structure Seismic Models

Fermi 2 safety related structures are founded on bed rock; therefore, no soil structure interaction effects are considered. Because the effects of horizontal and vertical excitations are reasonably independent, design basis dynamic analysis was done separately for the two orthogonal horizontal directions and the vertical direction. The horizontal design basis dynamic analysis stick models accounted for torsional effects in the building by considering the torsional inertia of the lumped masses representing the floor slabs, and by using mathematical models which account for the eccentricity between the center of mass and the center of stiffness. Three dynamic degrees of freedom (DOF) were assigned to each mass point, two horizontal translational DOFs and one rotational DOF about a vertical axis in the plane of excitation. Figure 3-4 shows the horizontal dynamic model for the reactor-auxiliary building. Figure 3-7 shows the horizontal model for the RHR building.

The reactor pressure vessel model was coupled with the reactor/auxiliary building mathematical model to account for interactions between the RPV and the building. Figure 3-5 shows the mathematical model for the RPV and reactor internals. The vertical design basis dynamic analysis utilized two-dimensional models which incorporate the wall axial stiffness and the beam-slab system flexural stiffness. Figure 3-8 shows the reactor/auxiliary building vertical dynamic model and Figure 3-9 shows the RHR building vertical model. Report SL-2682 **[3.43]** documents the design basis seismic analysis of the reactor/auxiliary building and report SL-3147 **[3.44]** documents the analysis of the RHR building.

For the new dynamic analysis performed to develop the RLE in-structure response spectra, the design basis horizontal dynamic models were modified by adding four weightless corner nodes for each slab to evaluate the building torsional effects at slab locations away from the center of mass. The corner nodes were connected to the slab mass center node with rigid elements. The torsional inertia assigned to the mass center nodes represent the entire slab torsional inertia. Two independent horizontal orthogonal excitation analyses were performed.

The vertical dynamic analysis for the RLE in-structure response spectra utilized the same mathematical models used in the design basis analysis. Calculation DC-5546 **[3.45]** documents the RLE dynamic analysis.

3.1.3.3 Structural Damping

EPRI NP-6041 provides recommendations for the damping values to be used in the seismic margins assessment evaluation. For reinforced concrete structures three levels are possible: 3% for slightly cracked concrete stressed at about half yield stress, 5% for moderately cracked concrete with about half yield stress, and 10% for concrete stressed close to yield. For welded structures the recommended values are between 3% and 7% depending on the stress level. The EPRI report states that the Regulatory Guide 1.61 **[3.46]** damping values are considered excessively conservative for the SMA.

For the Fermi 2 SMA, 7% of critical structural damping was used to determine the dynamic response of all reinforced concrete structures in both the horizontal and vertical directions. For other structures in the reactor/auxiliary building horizontal model, the following damping values were used:

- 3% for the steel containment vessel
- 3% for the steel crane bridge
- 3% for the reactor pressure vessel, support skirt, shroud, shroud head separator and the CRD guide tubes
- 3.5% for CRD housing
- 7% for fuel elements

The most significant damping value for the analysis is the 7% damping used for the dynamic response of the reinforced concrete structures. This is an increase from the 5% damping used for the design basis analysis. This increase is considered justified based on a review of the recommendations in EPRI NP-6041 and Regulatory Guide 1.61 and the stress levels anticipated due to the 0.3g RLE versus the 0.15g SSE.

3.1.3.4 Artificial Time History

A synthetic time history consistent with NUREG/CR-0098 median rock spectrum was used in the dynamic analysis performed to develop the in-structure RLE response spectra. The NUREG/CR-0098 spectrum was anchored at 0.3g for the horizontal analysis and at 0.2g for the vertical analysis. The artificial time history was digitized at 0.0C5 second. A plot of the time history with a peak acceleration of 0.3g is shown in Figure 3-18. A comparison between a response spectrum generated from the time history at 5% damping and a NUREG/CR-0098 spectrum is shown in Figure 3-19. The comparison in Figure 3-19 shows that the spectrum developed from the time history envelopes the target spectrum at most frequencies.

EPRI NP-6041 indicates that it is acceptable to have the spectrum, developed by the time history, fall below the smoothed target spectrum by as much as 10% over 10% of the frequency in any octave bandwidth over the amplified regions of the spectrum. Therefore, it was realized that additional conservatism in the artificial time history may still be extracted as discussed below.

3.1.3.5 Floor Response Spectra

RLE in-structure response spectra were generated by performing time history analyses on the building mathematical models. The time history excitation was applied at the building foundations represented by the fixed base of the stick models. As stated in EPRI NP-6041, more realistic input into the foundation may be determined by convolution or de-convolution of the free field surface input and accounting for the soil kinematic interaction. Therefore, conservative results may be expected if the free ground surface motion is applied to the foundation and the soil kinematic effects are neglected.

The RLE dynamic analysis was performed separately for two horizontal orthogonal directions and for the vertical direction. The time history method was used to generate response time histories of different mass points of the structural models. Frequency domain response spectra were then generated from the response time histories.

Horizontal response spectra included the effect of multi-directional excitation by combining the response of excitation from both horizontal analyses for each horizontal response. The combination used the SRSS method; therefore:

$$R_x = \sqrt{R_{xx}^2 + R_{yx}^2}$$

where:

R_x: Response spectral value at a node in the X-direction
R_{xx}: Response spectral value at a node in the X-direction due to X-excitation
R_{yx}: Response spectral value at a node in the X-direction due to Y-excitation

The in-structure RLE response spectra were not smoothened or broadened because EPRI NP-6041 recommends the use of peak shifting instead of peak broadening. Two sets of horizontal spectra were generated, one at the center of mass of the slab and the other set as an envelope of the response at the center of mass and the four corners of the slab. Most in-structure spectra were generated at 3% and 5% equipment damping. These damping values were selected to match the recommended values in Table 4-3 of EPRI NP-6041.

In order to provide a correction factor for the conservatism introduced in the artificial time history, the average of the ratios between the spectral acceleration of a NUREG/CR-0098 median 5% damped rock spectrum and the corresponding acceleration of the spectrum generated from the artificial time history was calculated for the frequency range between 4 and 25 Hz. This factor, which was equal to 0.9278, was used as a reduction factor for the in-structure response of the buildings. Since the horizontal floor response spectra were generated without the use of this factor, this reduction factor was applied to the resulting floor spectra. However, for the vertical analyses, this reduction factor was directly incorporated in the analyses resulting in floor response spectra with this correction built in.

In the vertical dynamic analysis, the mathematical models are modified to study the effect of slab response by including a multi-mass system at that slab level. A modified vertical model for determining slab response spectrum at the second floor of the reactor/auxiliary building is shown in Figure 3-10. The total effective slab mass is divided by the number of masses and assigned to each individual mass. The first mass simulates the stiffness of the lowest natural frequency of the slab and the other masses represent higher frequency mass systems within the seismic response frequencies (i.e., under 33 Hz). The total response of the slab is an envelope of the responses of the multi-mass system.

To further evaluate the effect of this multi-mass system, some RLE spectra were generated separately for each mass point of the multi-mass system; however, the results showed very little variation in the response indicating the insignificance of varying the slab frequency parameter on the overall response.

3.1.3.6 Scaling In-Structure Response Spectra

New RLE in-structure response spectra were generated for all horizontal locations in the reactor/auxiliary building and the RHR building and for most vertical locations. However, for few locations such as the fourth and fifth floor slabs of the reactor/auxiliary building, vertical response spectra were generated by scaling the available design spectra.

To scale the design spectra, three factors were considered. The first is a reduction factor to account for the higher structural damping used with the RLE compared to the SSE.

The second factor accounts for the difference in shape between the NUREG/CR-0098 ground response spectrum and the ground design basis spectrum. The third factor accounts for the increase in excitation level of the RLE compared to the SSE.

The correction factor for damping accounts for the higher energy dissipation in the structure at 7% of critical damping for the RLE compared with 5% for the SSE. The factor was calculated as the ratio of spectral accelerations from the Site Specific Earthquake ground spectra at 7% damping to that at 5% damping taken at the building fundamental frequency. This factor was also verified from different floor response spectra Zero Period Acceleration (ZPA) ratios for 7% and 5% structural damping.

The second factor was calculated as the ratio of spectral accelerations from the NUREG/CR-0098 median 5% damped rock ground spectrum and the design ground spectrum, taken at the building fundamental frequency. Both ground spectra were anchored at the same ZPA value. This factor accounts for the different level of excitation between the two ground spectra shapes at the response frequency of interest.

The third factor was taken as 2.0 to account for the 0.3g RLE versus the 0.15g design maximum ground acceleration. The product of all three factors was used to scale the design spectra to obtain RLE spectra.

3.1.4. Seismic Margin Evaluation

The EPRI seismic margin methodology relies heavily on earthquake experience data, generic qualification and fragility test data, results of seismic PRAs, and extensive use of expert judgment and experience to try to concentrate efforts on the evaluation of potential weak links that may determine the plant's real seismic margin above the SSE. The process is dependent on the knowledge and expertise of the SRT members who use the screening guidelines provided by EPRI and observations from the plant walkdown to screen out structures and components from further seismic review.

The margin evaluation emphasis is on demonstrating operability and survivability of components required for the functioning of a subset of plant systems that will bring the plant to a stable hot or cold shutdown condition and maintain that condition for at least 72 hours after the earthquake. While a single path of safe shutdown systems and components is all that is ultimately required, it is prudent to select two separate paths with the least amount of overlap between the systems and components. However, certain support systems, such as the emergency diesel generators, are required for all shutdown paths regardless of the primary systems chosen.

3.1.4.1 Overall Approach

Implementation of the Fermi 2 seismic margin evaluation is based on the deterministic approach outlined in EPRI NP-6041 with a main objective of identifying the "weaker-link" components that may limit the plant seismic margin. The seismic evaluation at

Fermi 2 may be broken down into three different groups: structures, distribution systems, and equipment.

Program guidelines for implementing the seismic margin evaluation at Fermi 2 were established and published as an attachment to the seismic design basis document, revision A **[3.47]**. These guidelines outline the implementation steps required for completing the seismic IPEEE as they apply to Fermi 2. Since Fermi 2 is not subject to the requirements of USI A-46 **[3.48]**, the guidelines were based primarily on the provisions in EPRI NP-6041 with the enhancements outlined in NUREG-1407.

The Fermi 2 success path and alternate path selection and the compilation of the SSEL were primarily performed by the PRA group; therefore, a great level of experience regarding plant systems, their reliability, and their risk significance was built into the selection process. Also, due to the significant interface with plant Operations during the internal events IPE analysis, Operations input was factored in the system selection and equipment list to a great extent even before Operations reviewed the seismic margin SSEL. Preliminary plant walkdowns were conducted in the plant accessible areas early in the program using a first-draft equipment list. The purposes of the preliminary walkdown were to locate equipment in the plant, to perform a preliminary evaluation based only on judgment, and to get a feel for the main areas where larger efforts may be required in the evaluation process. As a result of the preliminary walkdown, certain special evaluations, such as masonry and shield wall analysis, were identified as areas that required special attention in the seismic evaluation process.

The intent of the seismic evaluation process is to address the effects of the increased seismic demand resulting from the RLE on three aspects of the existing design of structures, systems and components. First, the capability of the element's anchorage to withstand the RLE loads is evaluated by performing generic bounding calculations or specific analyses based on the guidelines in EPRI NP-6041 and NP-5228 **[3.49]**. The anchorage is further evaluated during plant walkdowns to verify compliance with the configuration documents used in the analyses and to look for any unusual installations that may be of seismic concern. Second, the element's capability to withstand the RLE and remain functional is assessed. This process uses the screening tables and screening checklists in EPRI NP-6041 to screen out the element from any further functionality review. Further functionality evaluations may be required if the screening process indicates the need for additional reviews. Finally, the potential seismic interaction effects of items located external to and in close proximity to each component are evaluated during the plant walkdown. All three aspects of the equipment seismic evaluation process are coordinated and documented in the screening and evaluation work sheets (SEWS) recommended in EPRI NP-6041 for the corresponding class of equipment.

The Fermi 2 seismic margin program utilized senior utility engineers to conduct the evaluation. The system engineers' function was fulfilled by individuals from the PRA group supplemented by others from the Instrumentation and Controls group. The seismic engineering work was performed by structural engineers cognizant of the plant seismic design basis and qualification. For a few seismic evaluations, such as anchorage calculations, structures and distribution systems evaluation, and evaluations of some of the outliers, resources from outside engineering and consulting firms were utilized. However, these efforts were thoroughly reviewed by and very closely coordinated and discussed with the Detroit Edison project engineer during and after completion of the work.

3.1.4.2 Seismic Review Team

As stated in EPRI NP-6041, the seismic review team (SRT) involved in the seismic margin evaluation has the main responsibility for conducting the various steps of the SMA, especially applying experience and judgment in the screening process and the plant walkdown. The Fermi 2 SRT was comprised of senior engineers with a wealth of knowledge and experience in the field of structural and seismic design and analysis. The system engineers involved in the SMA have extensive knowledge in nuclear plant system functions and probabilistic risk assessment (PRA) application.

For the most part, Fermi 2's SMA was implemented by in-house employees; however, contractor consultation and reviews were utilized throughout the implementation of the SMA in order to ensure the validity of the assumptions and decisions made and to reinforce the Fermi 2 evaluation with experiences from other nuclear plant seismic margin studies. Most members of the SRT attended the SQUG and EPRI training courses related to the SMA, had access to all the relevant reports and training material, and easily met the qualification requirements in EPRI NP-6041. Other seismic capability engineers from outside engineering firms and consulting companies were also utilized in performing certain analyses and evaluations associated with the SMA. The Fermi 2 SRT members are listed below with a brief summary of their experience and qualifications.

<u>A. I. Hassoun</u>

Mr. Hassoun has a very broad experience in the structural mechanics and seismic design of nuclear power plants. He holds a B.S. degree in civil engineering and an M.S. degree in structural engineering, both from the University of Michigan at Ann Arbor. He has about 20 years experience in the design and analysis of industrial facilities, most of which are related to nuclear plants. Mr. Hassoun has been involved with the seismic program at Fermi 2 for fifteen years and is responsible for the establishment and management of the seismic qualification program since the completion of the design phase and the start of the operation phase of the plant. He is well familiar with industry developments in seismic design and evaluations and has represented Detroit Edison in seismic utility groups such as SQUG and the seismicity owners group. Mr. Hassoun served as the project engineer for the implementation of the seismic margin program at Fermi 2 and was the primary SRT member involved in the program implementation. He is a registered professional engineer in the State of Michigan.

A. P. Burg

Mr. Burg is a member of the Mechanical and Civil group at the Fermi 2 engineering department. He holds a B.S. and an M.S. degree in civil engineering from Drexel University in Philadelphia. Mr. Burg has over 22 years of diversified experience in the structural and seismic design and analyses of nuclear power stations. He has been working in the engineering group at the Fermi 2 plant site for ten years. Mr. Burg served as an SRT member for most of the equipment evaluation work and was an essential part of the walkdown team at Fermi 2. He is a registered professional engineer in the State of Michigan and the Commonwealth of Pennsylvania.

D. D. Jondle

Mr. Jondle has been a member of the Risk Assessment Engineering group at Fermi 2 since 1987. He holds a B. S. degree in nuclear engineering from the University of Wisconsin at Madison. Mr. Jondle has over 20 years of experience in nuclear reactor physics and engineering, and was involved in the PRA work for the Individual Plant Examination (IPE) program. He was the primary system engineer involved in the selection of the safe shutdown paths and equipment list. Mr. Jondle is a registered professional engineer in the State of Michigan.

L. G. Ferguson

Mr. Ferguson is a member of the Instrumentation and Controls group at the Fermi 2 engineering department. He holds a B.S. degree in electrical engineering from Michigan Technological University in Houghton, Michigan. Mr. Ferguson has extensive I&C engineering experience with nuclear power plants. He has been associated with the design and technical evaluation of I&C activities at the Fermi 2 plant for about 23 years. Mr. Ferguson was involved in the review of the SSEL and in finalizing the instrumentation scope for the seismic IPEEE program.

K. C. Hsu

Mr. Hsu is a member of the Mechanical and Civil group at the Fermi 2 engineering department. He holds a B.S. and an M.S. degree in civil engineering from National Cheng Kung University in Taiwan and another M.S. degree in structural engineering from Oklahoma State University. Mr. Hsu has about 30 years experience in the structural and seismic qualification area. He has worked at the Fermi 2 site since 1991. Mr. Hsu was involved in the seismic margin program as SRT member for the containment components. He is a registered professional engineer in the State of Ohio and the Commonwealth of Pennsylvania.

G. P. Hietpas

Mr. Hietpas is a supervising engineer with VECTRA Technologies, Inc. in Fort Worth, Texas. He holds a B.S. degree in civil structural engineering from the University of Wisconsin at Madison. Mr. Hietpas has more than twelve years of experience in structural engineering analysis and design and equipment seismic qualification. He was involved as an SRT member in some of the walkdowns, and in addressing the seismic/fire interaction issues. Mr. Hietpas is a registered professional engineer in the State of Wisconsin.

M. Amin

Dr. Amin is an engineering supervisor with Sargent and Lundy Engineers in Chicago. He holds a B.S. degree in civil engineering from Worcester Polytechnic Institute and an M.S. and a Ph.D. degree in structural engineering from the University of Illinois at Urbana. Dr. Amin has many years of teaching and industry experience in the structural mechanics area. He was involved as an SRT member in the Fermi 2 seismic margin evaluation for structures and distribution systems.

A. M. Al-Dabbagh

Dr. Al-Dabbagh is a senior structural engineer with Sargent and Lundy Engineers in Chicago. He holds a B.S. degree in civil engineering from the University of Baghdad in Iraq and an M.S. and a Ph.D. degree in structural engineering from Colorado State University. Dr. Al-Dabbagh has extensive experience in plant structures design and seismic qualification. He was involved as an SRT member in the evaluation of structures and distribution systems. Dr. Al-Dabbagh is a registered structural engineer in the State of Illinois.

3.1.4.3 Walkdown Preparation

Prior to the seismic walkdowns, the SRT collected and reviewed relevant design documents in order to become familiar with component design bases and to assist in screening out SSEL components. Various types of documents were reviewed, as applicable to each component.

1. Drawings showing physical locations in the plant and mounting or anchorage configuration;
2. UFSAR seismic licensing basis sections, as required, to determine conformance of the plant design with some of the screening guidelines in EPRI NP-6041;
3. Other design basis documents, as required, such as vendor manuals, specifications, seismic qualification reports, and seismic anchorage calculations;
4. Design calculations prepared for the SMA which developed the RLE required floor response spectra and documented bounding anchorage evaluations for the different equipment types; and

5. Seismic IPEEE reference books to review the caveats and other seismic information associated with the different equipment types.

In addition to the document review, the SRT performed the following tasks prior to the walkdowns:

1. Conducted a preliminary walkdown with an IPEEE consultant to become familiar with equipment locations and types and to get an initial indication of the seismic susceptibility of the equipment;
2. Initiated screening and evaluation worksheets (SEWS) for the various equipment; and
3. Determined what additional field information was required so that it could be collected during the walkdowns.

The preliminary work prior to the walkdowns enabled the SRT to optimize its efficiency and to obtain maximum benefit during the walkdowns.

3.1.4.4 Screening Criteria

The purpose of screening in the seismic margin program is to eliminate certain elements from the detailed review and evaluation scope in order to be able to concentrate on the evaluation of those elements that are considered "weaker-link" items and may potentially determine the real seismic margin of the plant. The screening criterion used at Fermi 2 is the one presented in Tables 2-3 and 2-4 of EPRI NP-6041. These screening tables provide generic conservative estimates below which it is generally not necessary to perform a seismic margin review. However, this does not mean the element does not require a plant walkdown. Screening tables address functionality considerations but do not address anchorage or seismic interaction. Thus, all elements on the SSEL require a plant walkdown. For Fermi 2, the applicable column in the EPRI screening tables is the first one corresponding to a peak spectral acceleration, for a 5% damped ground spectrum, not exceeding 0.8g. This is based on the Fermi 2 RLE which is consistent with a NUREG/CR-0098 median rock spectrum with a peak ground acceleration of 0.3g and a peak spectral acceleration of 0.636g.

Tables 2-3 and 2-4 and Appendices A and F of EPRI NP-6041 constitute the main screening tools used at Fermi 2 in conjunction with observations noted during the plant walkdowns. For those items that were not considered screened out, more detailed evaluations were performed utilizing the design basis seismic qualification documentation as base line information. The design basis qualification was reviewed for validity and completeness. If it was found acceptable, simple extrapolation to account for the higher seismic demand loads was utilized to evaluate the available margin. When applicable, certain built-in conservatism was eliminated, in accordance with the EPRI methodology, to evaluate the item's actual seismic capability.

Table 2-4 of EPRI NP-6041 includes a footnote (y) which states that items mounted at elevations exceeding 40 feet above grade should be reviewed if realistic SME 5% damped horizontal floor spectra exceed 2g. However, this footnote only applies to seismic margin evaluations for a 5% damped peak ground spectral acceleration of more than 0.8g. As stated above, Fermi 2's corresponding ground acceleration is less than 0.8g; therefore, this footnote does not apply to the Fermi 2 evaluation. The EPRI document includes another general caution statement on the use of the screening tables for components mounted significantly more than 40 feet above grade or in other spots where large resonant buildup of input motion might occur.

The intent of the note in EPRI NP-6041 is to caution the user against blindly applying the screening criteria in Tables 2-3 and 2-4 for items subject to horizontal response spectra with significant amplifications or high peak accelerations. The intent of the caveat was clarified in discussions with Dr. R. P. Kennedy, co-author of the EPRI document. Dr. Kennedy stated that the screening tables are applicable without further evaluation if the SME 5% damped clipped floor spectra have peaks of about 2.0g or less. Appendix Q of EPRI NP-6041 provides guidelines for clipping floor response spectra. Clipped peaks closer to 2.5g would be more of a concern and possibly result in more detailed evaluation of the components.

As stated in Section 3.1.3.5, the new RLE in-structure floor response spectra for Fermi 2 were conservatively generated by applying the free field seismic input at the base mat elevation of 540'-0" for the reactor/auxiliary building and 555'-0" for the RHR complex. The free field seismic motion would have been more appropriately considered at the grade level of 583'.

At Fermi 2, there are three buildings where SSEL components are located. The highest elevation in the reactor building where SSEL components are found is the second floor at elevation 613'-6". The auxiliary building has SSEL components at all elevations including the control room HVAC equipment on the fifth floor, elevation 677'-6". The RHR complex has most of the SSEL components on the first and second floors with the highest elevation of 617'-6". Clipping factors were developed for the RLE horizontal floor spectra per Appendix Q of EPRI NP-6041. None of the clipped spectra peaks was above 2.5g, and only the higher floors of the auxiliary building had peak spectra over 2.0g.

The EPRI NP-6041 caution statement, regarding the use of the screening tables for components mounted on the higher floors of the buildings, was addressed by performing a bounding functional evaluation on the most vulnerable components on the SSEL. The bounding components were selected by reviewing the peak clipped horizontal response spectra for both the reactor/auxiliary and RHR buildings and considering the type of SSEL components located at higher elevations in the buildings. Motor Control Centers were selected because of their location in the plant and because of their low natural frequency which is close to the peak frequency of the horizontal response spectra (5 Hz). Additionally, MCCs contain sensitive components with potential susceptibility to chatter during seismic events.

Components on the SSEL subjected to the highest seismic input are those located on the fifth floor of the auxiliary building. For the bounding MCCs, clipped floor RLE spectra were compared **[3.50]** with the test response spectra (TRS) in the seismic test reports. The TRS bounded the clipped RLE floor spectra for all frequencies above 4 Hz. The frequency range below 4 Hz is not significant since there is no equipment natural frequency below 5 Hz. Additionally, an evaluation of the 5/16-inch diameter bolts connecting the MCC with the base channels was performed. This latter check **[3.51]** was considered prudent because these bolts have been identified as weak links in the earthquake experience data as referenced in EPRI report number NP-5223-SL **[3.52]**. Results of the evaluation demonstrated that the bolts have adequate capacity to accommodate the RLE.

Based on the results of the spectra clipping calculation and the bounding evaluation of the MCCs, it was determined that the screening criteria presented in Table 2-4 of EPRI NP-6041 could be used to establish the functional capability of components on the SSEL regardless of their location within the Fermi 2 plant structures.

3.1.4.5 Seismic Capability Walkdown

One of the most important steps in the SMA is the plant seismic capability walkdown. The walkdown of the various plant areas where SSEL components are located is the responsibility of the SRT. Prior to the walkdowns, the SRT reviewed component drawings, bounding anchorage calculations, and other necessary documentation to become familiar with the components' documented configuration and the bases for their seismic qualification.

The seismic capability walkdowns assist in addressing the effects of the increased seismic demand from the SME on three aspects of the existing equipment design and qualification. First, functional capability of the equipment is assessed to determine the ability of the equipment to withstand the higher demand of the SME and remain functional to perform its safety-related function. Second, an assessment of the adequacy of equipment anchorage is required because the most common failure mode for equipment in actual earthquakes is the failure of its anchorage. Finally, the potential effects of items located external to and in close proximity of the equipment are evaluated.

Various methods are available for determining the seismic functional capability levels of equipment. These methods include those recommended by EPRI and SQUG, in addition to seismic capability data found in the specific equipment qualification reports. A preferred sequence of consideration for the methods is:

- Screening criteria in Table 2-4 of EPRI NP-6041;
- Original equipment qualification reports; and
- Generic equipment ruggedness spectra.

Several approaches are available for assessment of equipment anchorage adequacy. These approaches include, but are not limited to:

- Margins in existing anchorage qualification for the design basis event may be adequate to accommodate the SME;
- Existing anchorage qualification may be reworked with appropriate refinements to show acceptance for the SME; or
- Generic bounding calculations to typical anchor details may be developed.

The identification of potential seismic interaction issues is a key element of the seismic capability walkdowns. Seismic interaction items and concerns are identified based on engineering review and judgment during the walkdowns. Generic bounding analyses, such as the evaluation of block and shield walls, were prepared prior to the walkdowns to assist in identifying or eliminating seismic interaction concerns.

Thus, the purpose of the seismic capability walkdown is three-fold: (1) to screen from the margin review all elements with probable HCLPF capacities greater than the specified RLE level based on experience and judgment; (2) to define the failure modes of structures or components which are not screened, and the types of review which should be conducted; and (3) to identify seismic (system) interactions with the potential to adversely affect equipment on the SSEL. The walkdowns are also conducted to look for outliers, lack of similarity between divisional components, differences in anchorage from what is shown on drawings, potential system interaction issues, and any other areas of seismic concern.

All components on the SSEL were walked down and inspected. Each component was adequately inspected to rule out any seismic concern or anchorage deficiency. For groups of similar energized components, at least one of the components was opened and inspected.

A very small number of components to be inspected were located in either inaccessible or high radioactive or contaminated areas. In such cases, the SRT inspected the components by using photographs or a quick "walk-by." Screening of these items relied more on as-built drawing configurations and anchorage seismic re-analysis than on detailed field inspection.

The SRT checked the mounting of instruments on their respective racks and panels. Mounting was reviewed for conformance with the manufacturer recommendations and sound seismic installation judgment.

The SRT also "walked by" a sampling of subsystems such as piping, tubing, cable trays, conduits, and HVAC ducts to determine their ability to withstand the RLE seismic loads. The walkdown concentrated on distribution systems in the areas containing essential equipment as recommended in EPRI NP-6041.

The walkdowns were facilitated by the use of screening and evaluation worksheets (SEWS) found in Appendix F of EPRI NP-6041. SEWS are written in an abbreviated format and contain keywords to remind the SRT of important screening criteria and guidelines for each equipment category. The SEWS forms are divided into categories including general descriptive information, component evaluation (functional capability), relay walkdown, anchorage evaluation, system interaction effects, and potential problem description. The SRT used the SEWS to record results of the walkdowns, reference outlier resolution, and document seismic evaluations.

3.1.4.6 Combination of Seismic and Hydrodynamic Loads

Since seismic events can induce hydrodynamic loads in a BWR, questions with regard to whether and how these loads (or the responses they induce) should be combined must be addressed. Seismic induced hydrodynamic loads can be the result of an SRV discharge or due to a LOCA.

In accordance with Appendix K of EPRI NP-6041, seismic induced intermediate or large LOCA are not considered credible events. Also, the hydrodynamic loads caused by a small LOCA (chugging) tend to occur after the earthquake. Hence the hydrodynamic loads induced by these events need not be considered in the seismic margin assessment.

Safety relief valves will actuate in response to a reactor system pressure transient. Since an SME is assumed to cause loss of offsite power and subsequent turbine trip, SRV actuation may be rapidly induced as a result of the transient. Thus, SRV loading should be combined with seismic loading in the seismic margin assessment.

Due to the relatively short duration of the peak SRV loading and the random nature of the time history, it is highly unlikely that the peak SRV and earthquake responses will occur simultaneously. Hence, EPRI NP-6041 recommends that the responses due to the RLE and SRV discharge be combined by SRSS. Therefore, the Fermi 2 seismic margin assessment considered SRV response in combination with RLE response using the following combination:

$$\text{Total Response} = \sqrt{SME^2 + SRV^2}$$

3.1.4.7 Evaluation of Structures

Table 2-3 of EPRI NP-6041 provides guidelines for screening of civil structures included in the seismic margin evaluation. The applicable structural items included in the Fermi 2 SMA are the steel containment, drywell internal structures, CRD housings and mechanisms, containment shield wall, other shear walls, diaphragms and footings. Category I steel frame structures, and non-Category I structures with potential to fail Category I structures. Fermi 2 structures are founded on bed-rock; therefore, no soil failure modes or soil liquefaction and slope stability evaluations are necessary. Also, no dams, levees or dikes are required for the SMA safe shutdown paths. The evaluation of the reactor vessel internals is discussed in Section 3.1.5.2

For certain other structures, such as masonry and shield walls and the control room ceiling, the screening tables indicate the need for more detailed evaluation. Fermi 2 masonry and shield walls were evaluated for the RLE in design calculation DC-5591 **[3.53]** and the control room ceiling was evaluated in detail by the SRT during the walkdown. Results of these evaluations are discussed in Section 3.1.5.1 and 3.1.5.3, respectively.

Drywell and Torus

The Fermi 2 drywell is a typical Mark I Boiling Water Reactor (BWR) bulb-shaped steel vessel with a spherical lower portion and a cylindrical upper part. Figure 3-20 is a general arrangement section through Fermi 2's reactor, auxiliary and turbine buildings showing the relative location of the drywell in the reactor building. The spherical section at the bottom of the drywell has a 68-foot diameter, while the upper cylindrical section is 38'-10" in diameter. The overall height of the drywell vessel is about 115 feet. The drywell is completely encased in the reinforced concrete drywell pedestal at the bottom. Special shear lugs are provided to connect the drywell shell with the concrete floor inside the drywell and the drywell pedestal. The drywell skirt was left in place during the erection process, thereby providing additional horizontal shear resistance. Figure 3-21 shows the attachment of the drywell to the drywell pedestal.

There is a small pocket of compacted sand surrounding the drywell above the drywell pedestal that provides for drainage and forms a transition area between the fully embedded portion of the drywell and the unrestrained upper part. Above elevation 572'-1", a reinforced concrete biological shield wall, monolithic with the floor slabs of the reactor building, surrounds the drywell. The drywell is separated from the concrete shield wall by a two-inch gap filled with compressible foam. The drywell is also restrained by the biological shield wall at elevation 647 feet through eight guided connections at 45-degree spacing. These connections permit some radial and vertical movement but inhibit any tangential movement of the drywell at this elevation.

According to Table 2-3 of EPRI NP-6041, the drywell may be screened out if the steel pressure boundary is keyed to the base mat to prevent slipping. Therefore, based on the construction details as discussed above, the drywell is considered to satisfy the screening requirements for the RLE.

For BWR Mark I tori, EPRI NP-6041 indicates that an evaluation is required for seismic input beyond the design basis. The Fermi 2 pressure suppression chamber is a torus-shaped vessel with a 112'-6" major diameter and a 30'-6" cross sectional diameter. Figures 3-22 and 3-23 show the plan and support details for the torus, respectively. Saddle supports are located at sixteen mitered joints around the perimeter, and additional seismic ties are provided at four locations with 90-degree spacing.

In NUREG/CR-5098 **[3.54]** the Fermi 2 containment integrity was analytically evaluated against seismic vulnerabilities. This report, which considered four plant containments, evaluated the seismic capability of the containments under different internal pressure scenarios. Seismic acceleration capacities were calculated for various elements of the containment systems, including the torus, using horizontal and vertical time history analysis of the relevant building models. For the Fermi 2 containment, it was concluded that yielding in the torus supports would initiate at a peak ground acceleration well beyond the 0.6g level. This conclusion was based on using a Regulatory Guide 1.60 spectral shape input. The critical stress point for the seismic loading was identified as point A as shown in Figure 3-23.

Based on the study in NUREG/CR-5098, it is concluded that the Fermi 2 torus has adequate seismic capacity for the RLE.

Drywell Internal Structures

Figure 3-20 shows a cross section of the drywell delineating the major internal structures. The drywell internal structures include the sacrificial shield, reactor pedestal, drywell floor, gallery floor levels, earthquake stabilizer truss, and the pipe break support truss system. All these structures are classified as Category I. The table below identifies the construction material for each structure and the section in the UFSAR which documents that SSE loads are considered in the design of the structure.

Internal Structure	Construction	SSE in Design?	UFSAR Section
Sacrificial Shield	Composite structural steel and plain concrete cylindrical shell	Yes	3.8.3.3.1
Reactor Pedestal	Reinforced concrete cylindrical shell	Yes	3.8.3.3.2
Drywell Floor	Reinforced concrete pad	Yes	3.8.3.3.3
Gallery Floor Levels	Steel beams	Yes	3.8.3.3.4
Earthquake Stabilizer Truss System	Structural steel truss	Yes	3.8.3.3.5
Pipe-Break-Support Truss System	Structural steel truss	Yes	3.8.3.3.6

According to Table 2-3 of EPRI NP-6041, Category I containment internal structures may be screened out, for a peak spectral acceleration less than 0.8g (or peak ground acceleration of 0.3g), if the design was based on an SSE of 0.1g peak ground acceleration or greater. Since the Fermi 2 design is based on an SSE of 0.15g, the drywell internal structures are considered to possess adequate seismic capacity to withstand the RLE.

Control Rod Drive Housings and Mechanisms

Per Table 2-4 of EPRI NP-6041, the control rod drive (CRD) housings and mechanisms do not require evaluation for seismic margin provided the housings are provided with a lateral seismic support.

As shown in Figure 3-24 **[3.55]**, the CRD housings at Fermi 2 are laterally supported at an elevation 124 inches below RPV invert. The housings are provided with adjustment bolts that are welded in their position after final adjustment to provide lateral contact between adjacent housings. The peripheral housing bolts are adjusted to provide contact with steel members anchored to the reactor pedestal before the bolts are welded in their final position. GE drawing numbers 197R603 **[3.56]** and 762E827 **[3.57]** show the CRD housing restraint details.

Based on the discussion above, the CRD housings and mechanisms satisfy the EPRI screening guidelines and are screened out for a 0.3g RLE.

Containment Shield Wall

The containment (biological) shield wall enclosing the steel drywell is a reinforced concrete shell structure, monolithically constructed with the drywell pedestal, with a thickness that varies from four to seven feet. It is separated from the drywell steel shell by a two-inch gap filled with compressible foam. The shield wall extends up to the refueling floor at elevation 684'-6" and it is integral with the intermediate floors. The shield wall is designed as a Category I structure using load combinations that include the effects of the SSE. In accordance with Table 2-3 of EPRI NP-6041, the containment shield wall can be screened out from further evaluation since it has been designed for an SSE of 0.15g, which is greater than the 0.1g screening threshold.

Additional information is available on the seismic capability of the Fermi 2 biological shield wall based on evaluations made in NUREG/CR-5098. The part of the wall which is most seismically vulnerable was identified to be the section between the first and second floors of the reactor building. Evaluation of this section using the ACI code yielded a seismic capability of 0.39g based on a Regulatory Guide 1.60 spectral shape.

Based on the above discussion, the biological shield wall is considered to have a HCLPF of 0.3g or greater.

Shear Walls, Diaphragms and Footings

SSEL components are located in the reactor/auxiliary building and the RHR complex; therefore, reinforced concrete elements in these two buildings must be addressed. The reactor/auxiliary building has one common 4'-0" thick foundation mat. This mat is thickened to 19'-0" over a circular area of 77'-0" in diameter under the drywell to form the drywell pedestal. The auxiliary building is a reinforced concrete building up to and including its roof. The reactor building is a reinforced concrete building up to the refueling floor at elevation 684'-6". Above the refueling floor, the reactor building is a steel framed structure with metal siding and metal roof deck. The RHR complex is a 280 ft. long, 127 ft. wide reinforced concrete structure with a 4'-0" thick mat. The RHR complex houses the emergency diesel generators and the RHR reservoir.

Both the reactor/auxiliary and RHR buildings are designed as Category I structures. The load combinations used in the design included the effects of the SSE on the walls, floors and foundations. In accordance with Table 2-3 of EPRI NP-6041, these elements may be screened out for an RLE with peak spectral acceleration less than 0.8g (or 0.3g peak ground acceleration) provided they were designed to an SSE level of at least 0.1g peak ground acceleration. Since the Fermi 2 design is based on an SSE of 0.15g, the shear walls, diaphragms and footings satisfy the EPRI screening guidelines and can be assigned a minimum HCLPF capacity of 0.3g PGA.

Category I Concrete and Steel Frames

Lateral load resistance in the reactor/auxiliary and RHR buildings is provided by the shear wall and diaphragm elements discussed above; therefore, no other reinforced concrete frames are used for lateral load transfer.

Steel framing is used above the reactor building refueling floor for the building crane and roof support. Structural steel bents, made from built-up girders and rolled columns, are used to support the roof. Columns supporting the reactor building crane girders are welded to the roof framing columns. Two braced bays are provided on each of the four sides to provide a lateral load support mechanism. Also, horizontal bracing is available on the roof framing steel to provide diaphragm action. Groups of four or six cast-in place 2¼ inch diameter bolts are used to connect the column base plates to the concrete floor at elevation 684'-0". The average bolt embedment length is about 20 inches.

Table 2-3 of EPRI NP-6041 states that Category I steel frames may be screened out for an RLE with peak spectral acceleration less than 0.8g provided they were designed to an SSE level of 0.1g or greater. Since the Fermi 2 design SSE is 0.15g and since the design load combinations for the reactor building roof steel framing considered the SSE loads, the framing elements satisfy the EPRI screening guidelines and can be assigned a minimum HCLPF capacity of 0.3g PGA.

In addition to the EPRI screening guidelines, the SRT reviewed the design calculation [3.58] and associated drawings for the reactor building roof framing structure. This review identified that the design basis shows little margin with respect to pullout capacity for some of the column base plates. Therefore, although the EPRI screening guidelines resulted in a satisfactory screening, it was considered prudent to perform additional evaluation on the column embedments to ascertain their seismic capacity relative to the RLE.

An evaluation was prepared to assess the capacity of the reactor building refueling floor steel framing embedment details relative to the larger seismic loads associated with the RLE. The evaluation **[3.59]** demonstrated that adequate margin exists to accommodate the RLE loads. The additional margin was realized by eliminating conservative assumptions and procedures employed in the design basis work.

Non-Category I Structures

The only non-Category I structure with potential to fail or affect seismic Category I structures is the turbine and radwaste building and the hyperbolic cooling towers. The turbine/radwaste building is located next to the auxiliary building with a four-inch gap between the two buildings. The two buildings are on separate foundations. The EPRI guidelines for screening non-Category I structures indicate that they can be screened out as long as the structures are capable of meeting the 1985 Uniform Building Code (UBC), zone 4 requirements.

The turbine/radwaste building was designed using the 1971 version of the UBC. Lateral seismic forces were calculated per the UBC formula based on the structure type, the seismic zone factor, and other pertinent factors. As is always the case for UBC building design in low seismic regions, the wind forces govern the lateral load resistance design over the seismic loads. However, a design basis concern regarding the survivability of the outboard main steam isolation valves (MSIVs), located in the steam tunnel portion of the turbine building, resulted in further seismic evaluation of the turbine building. Therefore, a dynamic modal analysis and a response spectrum analysis were performed on a model representing the turbine building. The analysis **[3.16]** was similar to the dynamic analysis performed for other Category I structures.

As a result of the seismic analysis, it was concluded that the turbine building would maintain structural integrity under seismic loading associated with the SSE. Furthermore, the maximum building lateral displacement was very small.

Although no documentation demonstrates compliance of the turbine/radwaste building design with the EPRI screening criteria, the SRT concluded that the later seismic analysis of the building provides a level of seismic capacity commensurate with the 1985 UBC Zone 4 requirements. Therefore, the building was screened out. In addition, lateral deflections under an earthquake magnitude similar to the RLE would not be large enough to close the four-inch gap between the two buildings. Therefore, impact between the two buildings is not anticipated. Even if contact between the two buildings was postulated, it is not expected that any significant damage would occur due to the massive reinforced concrete design of the two buildings. Also, if the turbine building steel superstructure framing was postulated to fail, the failure would be away from the auxiliary building due to the height of the adjacent auxiliary building as illustrated in Figure 3.20.

3.1.4.8 Evaluation of Distribution Systems

This section describes the evaluation of certain subsystems such as the NSSS primary coolant system and other distribution systems such as piping, HVAC ducting, cable trays, conduits, and instrument tubing.

NSSS Primary Coolant System (Piping, Vessel and Supports)

The EPRI SMA methodology and screening criteria indicate that the NSSS primary coolant system can be screened out from detailed margin evaluation provided that the NSSS piping welds susceptible to intergranular stress corrosion cracking (IGSCC) are properly evaluated. The NSSS supports can be screened out if their design considered dynamic SSE loading combined with pipe break loads.

As stated in the UFSAR Section 3.9.1.5.6, Fermi 2 reactor coolant pressure boundary piping systems, including supports, have been analyzed for an SRSS load combination of dynamic SSE loads and annulus pressurization (AP) load. AP refers to loading caused by a postulated guillotine pipe rupture in the area between the sacrificial shield wall and the RPV. As a result of the analysis, minor structural steel support weld modifications were implemented to ensure that allowable weld stress limits were not exceeded.

Based on the discussion above, it is concluded that the NSSS supports meet the EPRI NP-6041 screening criteria and can be assigned a minimum HCLPF capacity of 0.3g PGA.

IGSCC

Intergranular stress corrosion cracking is a form of cracking that occurs along the grain boundaries of certain stainless steel materials. IGSCC is normally found in the heat affected zone of butt welded joints along the piping. The current NRC position regarding IGSCC in BWR austenitic stainless steel is documented in NUREG/0313 **[3.60]**. Two generic letters document NRC guidance for IGSCC: GL 88-01 **[3.61]**, and Supplement 1 to GL 88-01 **[3.62]**.

GL 88-01 required assessment of austenitic stainless steel piping with 4-inch or larger diameter that contain reactor coolant, and is above 200°F during normal power operation, regardless of ASME code classification. For welds found within the scope of NUREG/0313, GL 88-01 requires compliance with certain in-service inspection programs and leak detection procedures. Supplement 1 to GL 88-01 provided acceptable alternative staff positions with regard to inspection of reactor water clean-up system piping and the leak detection requirements.

The major points related to Detroit Edison's program for IGSCC are contained in communications between Detroit Edison and the NRC **[3.63 through 3.66]**. These communications indicate the following:

- Detroit Edison took a number of steps prior to commercial operation of Fermi 2 to avoid IGSCC. These included replacement and heat treating of piping **[3.63]**. As a result, only 18% of the total population of welds within the scope of GL 88-01 were identified as susceptible to IGSCC and requiring inspection. Another 24 welds were added to this list in a subsequent assessment by Detroit Edison **[3.64]**;

- Fermi 2 currently has a list of welds requiring in-service inspection that has been accepted by the NRC **[3.65]**;

- Fermi 2 water chemistry specifications meet or exceed the intent of BWR Owners Group Guidelines for mitigation of IGSCC in the primary coolant **[3.63]**; and

- Fermi 2 plant technical specifications have been revised to implement the NRC guidance contained in Supplement 1 to GL 88-01 for Reactor Coolant System Operational Leakage Detection. This revision has been approved by the NRC as Amendment No. 89 to the Facility Operating License **[3.66]**.

Based on the above, it is concluded that Fermi 2 has an acceptable program for addressing IGSCC in the NSSS primary coolant system. Therefore, this system can be screened out and assigned a minimum HCLPF capacity of 0.3g PGA per the guidelines in EPRI NP-6041.

In addition to the steps mentioned above, a Hydrogen Water Chemistry (HWC) program has been recently implemented at Fermi 2 . The purpose of HWC is to slow the rate of crack formation in the RPV lower internal components and the NSSS primary coolant piping system. Control of the IGSCC phenomenon is achieved through hydrogen gas injection into the feedwater system and oxygen gas injection into the off-gas system. Although the main purpose of HWC is to control IGSCC in the RPV, the NSSS primary coolant system is an obvious beneficiary of the program.

Category I Piping

In accordance with Table 2-4 of EPRI NP-6041, Category I piping may be screened out from detailed margin review provided a representative sample walkdown is conducted to look for certain known concerns. The purpose of the visual inspection is to assess piping system vulnerability based on known failure modes from earthquake experience data. The two main issues known to cause seismic piping failure are inadequate piping system flexibility and excessive relative displacements. Valve seismic interaction concerns are addressed separately in the valve evaluation section.

The SRT selected the EECW Division 2 pump discharge piping and the service water return line to EECW Division 2 heat exchanger, on the second floor of the reactor building, as representative samples for visual inspection. The isometric piping diagrams for these two systems are shown on drawings 6M721-3368-1 **[3.67]** and 6M721-3353-1 **[3.68]**, respectively. In addition to the specific piping systems, the SRT reviewed various other safety-related piping runs throughout the plant while conducting the equipment walkdowns. The following observations were made during the walkdown of the sample piping systems and other plant area walkdowns:

- Safety-related piping has welded or bolted flange connections. The only threaded connections were observed on non-safety related fire protection overhead sprinkler lines; however, threaded connections in these lines were judged to be acceptable given the good support configuration of the piping. The only credible vulnerability was identified as sprinkler head failure due to spatial interaction with any adjacent items. However, discussions with fire protection engineers and field observation confirmed that the sprinkler heads are installed in configurations with at least one foot of clear space around them to permit their sprinkling action.

- No cast iron piping was found during the walkdowns.

- No credible failure that may result from stiff branch lines attached to larger flexible pipes were found. Branch lines possess adequate flexibility due to piping offsets and support configuration restraints.

- Piping systems are well supported near their connections into vessels and other equipment. The piping and equipment supports are generally compatible in stiffness; therefore, no excessive nozzle loading cases were found that may be of concern.

- No excessively flexible piping runs that may cause seismic interaction concerns were found. Fermi 2 initiated a rattlespace program to identify and evaluate seismic interaction problems in safety related components prior to plant operation.

The SRT also performed a drawing review of connections of buried piping at building penetrations shown on drawings 6M721N-2180-1 **[3.69]** and 6M721-3185-1 **[3.70]**. The piping penetrations are either flexible through a guard pipe or are well anchored at the wall. Since only small displacements are expected for a rock site like Fermi 2, these details are considered acceptable connections for buried piping.

Based on the sample walkdown and other evaluations discussed above, Category I piping at Fermi 2 is considered to have a HCLPF seismic capacity equal to or greater than 0.3g.

HVAC Ducting and Dampers

According to Table 2-4 of EPRI NP-6041, HVAC ducting and dampers may be screened out from detailed margin evaluation provided a walkdown of a representative sample shows no seismic concerns. The SRT performed a walkdown of representative ducting and supports, including appurtenances. The main areas reviewed were the CCHVAC system, the switchgear rooms, and the first and second floors of the RHR building. The CCHVAC system originates on the fifth floor of the auxiliary building and serves the standby gas treatment rooms, control room, computer room, relay room, and cable spreading room.

The following observations were made during the plant walkdowns:

- Horizontal and vertical duct spans are reasonable, typically less than about 8 feet.

- Ductwork has companion angle construction at transverse seams, with adequate bolting of flanges and welding to the duct skin.

- Ductwork is positively restrained by duct supports, including wall bracket supports. On lateral supports the duct is surrounded by support members to restrain duct movement. On longitudinal supports the duct is welded to the support.

- Ductwork has flexible joints at connections to fans.

- Adequate framing and support are provided near equipment attached eccentrically to ductwork.

- Supports are provided near heavy in-line equipment.

- Supports are fabricated from structural steel sections (primarily angle members). Rod hangers are not used to support safety-related ductwork. Supports are typically braced in-plane and out-of-plane, as appropriate.

Most of the ductwork appeared to be well constructed and adequately supported. In general, supports were judged to have adequate capacity for the Review Level Earthquake. Adequate support was provided in the longitudinal direction and at vertical runs of duct. No particular "hard-spots" were noted ;however, the SRT did note that some supports in the CCHVAC system were not as rugged as those in other areas. In particular, supports above the control room are long, slender trapeze-type supports with minimal lateral bracing. The SRT noted that the anchorage for these supports may be non-ductile.

Based on observations from the plant walkdown combined with a review of plant design drawings, the SRT selected several representative supports from the CCHVAC system for review. This review is documented in design calculation DC-5749 **[3.71]**. The selected supports were evaluated based on the recommendations of the GIP and EPRI NP-6041. Results of this review demonstrate that the duct supports have adequate capacity for the RLE.

Fire dampers in the reactor/auxiliary and RHR buildings were generically evaluated. Sample dampers of different sizes were reviewed and walked down to evaluate the potential of undesired damper closure due to a seismic event. No concerns were generated from the review and walkdowns. The fire dampers used at Fermi 2 were proof tested by the vendor at relatively high accelerations intended to envelope applications in west coast nuclear plants. Therefore, it was concluded from the review that fire dampers have adequate capacity to withstand the RLE and remain functional.

In addition to the general area walkdown, the SRT reviewed ductwork inside the drywell. The primary objective of this review was to determine whether any ductwork was attached to both the containment vessel and internal structures. Ductwork attached to both could be subjected to substantial loading due to relative displacements. This review was based on general observations during the drywell equipment walkdowns, as well as a review of plant design documentation. No instances were found where ductwork is attached to both the containment vessel and the internal structures.

Based on the walkdown observations and the analysis performed, the SRT concluded that the Fermi 2 HVAC ducting and dampers have adequate capacity to withstand the RLE without compromising their essential function.

Cable Trays and Conduits

Table 2-4 of EPRI NP-6041 indicates that cable trays and conduits and their supports may be screened out for a 0.3g seismic margin earthquake without any additional evaluation. However, the SRT performed a walkdown on sample runs of cable trays and conduits to look for possible weak links that may present a concern.

Fermi 2 safety-related cable trays and conduits, including their supports, are qualified to withstand SSE loads. Cable trays at Fermi 2 are generically qualified based on bounding cable loading and maximum spans. There is an on-going monitoring program to track cable weight in trays due to plant modifications and to reconcile cable tray hanger calculations due to these load changes, as necessary. The cable tray support system includes lateral load bracing in the longitudinal and transverse directions. Electrical conduits and supports are also generically qualified for maximum conduit span lengths as specified in installation specifications.

The SRT found the cable trays and conduits to be well supported for seismic loading, with rugged support and connection details. There were no seismic concerns identified as a result of the walkdown; therefore, both cable tray and conduit systems at Fermi 2 are considered to have sufficient seismic capacity to withstand the RLE and are assigned a minimum HCLPF capacity of 0.3g PGA.

Instrument Tubing

EPRI NP-6041 does not provide specific guidelines for screening instrument tubing; however, the rules for piping systems can be applied. Due to their low mass, inertia failure of tubing systems is typically not a concern if the tubing is reasonably well supported. Of primary concern are spatial interaction effects and relative displacements.

Fermi 2's control air system design does not use local accumulators in the various service areas of the plant; therefore, a failure of the tubing at one location would remove a whole division of the system. However, instrument tubing and supports were seismically designed for SSE loading. Conservative calculations for bounding design basis load combinations resulted in generic installation and support specifications for safety-related instrument tubing.

Based on observations from preliminary plant walkdowns, the SRT concluded that the only credible failure mode for control instrument tubing would be spatial interactions (items in close proximity to instrument tubing, or potential failure and falling of items onto the tubing). Subsequently, the SRT observed instrument tubing installations during equipment walkdowns, with special attention to spatial interaction concerns at components serviced by control air and in the general routing areas of the plant. No potentially damaging interactions were found. The tubing supports were found to be well designed, yet the routing of the tubing provided adequate flexibility for relative displacement.

Since no specific seismic concerns were identified, it is concluded that the instrument tubing and its supports are rugged enough to withstand the RLE without compromise of equipment serviced by control air. Therefore, instrument tubing is assigned a minimum HCLPF capacity of 0.3g PGA.

3.1.4.9 Evaluation of Equipment

The SSEL equipment in the 22 categories discussed in Section 3.1.4.9.2 were evaluated by the SRT. The results of the evaluations are summarized in this section. The equipment categories correspond to the ones listed in EPRI NP-6041, Table A-1. The screening criteria of EPRI NP-6041, Table 2-4, were used for guidance in the evaluation of the equipment. The SRT reviewed pertinent drawings and other documents for the equipment prior to the walkdowns in each category. Walkdowns were performed by the SRT for virtually all of the equipment listed in the sections below with exceptions being noted. During the walkdowns, the SRT observed the following items for each component, as applicable: overall appearance, strength and stiffness, instrument and internal device mountings, component mounting, anchorage and concrete condition, proximity to adjacent structures and system interaction effects, conformance with design drawings, and unique attributes associated with a particular category. Any non-conformances were noted during the walkdowns and on the SEWS and were later evaluated for their effects on the components' ability to withstand the RLE, as described below.

Numerous techniques were used by the SRT in the evaluation of concerns noted during the walkdowns. The methods allowed by EPRI NP-6041 were employed, as necessary, in the evaluations. Examples of such methods include the use of the "100-40-40" rule, increased allowable stresses, peak clipping of response spectra, system characteristics which justify non-concurrence of peak operating and seismic loads, a 1.0 instead of 1.5

multi-frequency response and multi-mode excitation factors used in the equivalent static analysis method, and dynamic response spectrum analysis.

In a number of cases, minor field modifications, in lieu of detailed evaluations, were made to address non-conforming conditions. A list of these modifications is presented in Table 3-7. One EDP resulted from the equipment evaluations; its purpose was to attach adjacent relay panels to each other to prevent their knocking during seismic events and thus adversely affecting sensitive internal equipment. Plant improvements resulting from the seismic review are discussed in Section 7.1.

3.1.4.9.1 Evaluation of Equipment Anchorage

Based on past performance of industry-wide plant equipment subjected to seismic events, the existence of properly engineered anchorage is one of the most important items which affects the seismic performance of the equipment. Equipment failure due to sliding or overturning has resulted from lack of positive anchorage and improperly engineered anchorage. Examples of poor anchorage include expansion bolts with short embedments, friction clips, and base anchorage with large eccentricities which allow equipment base to bend or tear or which generate large prying forces. Guidelines for the inspection of equipment anchorage have been documented in numerous references **[3.4, 3.49, 3.72]**.

The importance of good, sound anchorage of equipment to their support structures is to assure that the equipment remains in place during a seismic event. Adequate anchorage provides a sound load path for the seismic loads from the equipment to the supporting structure. Equipment is anchored to concrete structures by one of three general methods – expansion anchor bolts, cast or grouted-in-place anchors, and welds to embedded steel. The expansion anchors are generally three types – Phillips self-drilling and wedge anchors and "Hilti" wedge anchors.

The SRT examined the anchorage of accessible mounted equipment, in detail, to assure its soundness and rigidity. For components such as switchgears and transformers where the anchorage was inaccessible without removing front and back panels of energized electrical equipment, the SRT made special arrangement with plant operation personnel to have one typical unit of each type opened for SRT inspection and examination. Other units of the same type were considered adequate based on the anchorage inspection of the typical units. The SRT looked for the following items, as appropriate for each component, in accordance with the guidelines of EPRI NP-6041:

1. Verification of the number, type, condition and size of anchor bolts, plug welds, and fillet welds.
2. The anchor spacing, free-edge distance, and concrete condition.
3. Anchorage conformance to design drawings.
4. The presence and tightness of nuts on anchors and the tightness of bolts in expansion anchors based on visual inspection.

5. Relative stiffness of anchorage and the possibility of excessive prying action on the anchors.
6. Gaps between the equipment base and the concrete floor in excess of 1/4".
7. Equipment base strength and structural load path.

If one or more of the above attributes was deficient in an equipment anchorage, further evaluation was performed to justify the as-found anchorage condition. Seismic demand was compared with the anchorage capacity. As necessary and appropriate, layers of conservatism were removed in the evaluations until the capacity exceeded the demand.

Design calculation DC-5634 **[3.73]** was prepared as a generic bounding anchorage evaluation for the SSEL components to determine if they are able to withstand the RLE. The DC was a tool used by the SRT prior to the equipment walkdowns to determine what anchorage might need further detailed inspection and to provide information to assist in making field judgments. The SRT also verified assumptions which were made in the DC and confirmed that field configurations matched the DC-evaluated configurations. In the DC, equipment was grouped to simplify the analyses by allowing bounding calculations to be developed based on similarity and common configurations. An assumption was made in the DC that hairline concrete cracks were unlikely. This assumption allowed the use of a factor of safety of 3.0 for single bolts and 2.8 for two or more bolts as recommended in EPRI NP-6041. The SRT observed the concrete in the vicinity of the component anchorage and confirmed this assumption.

3.1.4.9.2 Equipment Category Evaluation

The equipment categories described in this section are identified in EPRI NP-6041 as typically required for a safe plant shutdown. The categories consist of active electrical and mechanical equipment and passive electrical equipment. Table 3.6 provides a listing of the different equipment categories, the number of SSEL items in each, and the number of outlier items. Each subsection is arranged in a fashion similar to the order of the SEWS, namely, category, location, equipment evaluation, anchorage evaluation, system interaction effects, and outlier and outlier resolution. Relay evaluation is described elsewhere in this report (Section 3.1.2.3).

Component mounting and seismic restraints inside electrical equipment were observed by the SRT on a sampling basis. For personnel safety and continued assurance of equipment operability, only a limited number of the electrical switchgear, MCCs, and panels were opened for internal device inspection as well as observation of the equipment mounting and restraints. Since the similar, unopened electrical equipment was built, in general, by the same manufacturer, it was felt that observation of only a sample was sufficient to assure satisfactory mounting and restraint of the unobserved equipment.

An evaluation of four, bounding condition block walls and shield walls was performed to determine their ability to withstand the RLE loading **[3.53]**. The walls were selected based on their plant locations on second and fifth floors of the reactor and auxiliary buildings. The block walls which were evaluated lack internal reinforcement ; therefore, their capacity to resist out-of-plane loading is significantly reduced. It was determined that the worst-case walls were qualified to withstand the RLE loads. A more complete discussion of block walls is found in Section 3.1.5.1. Since the masonry block walls were qualified generically, there will be no mention of the walls in the "System Interaction Effects" sections of the individual equipment categories below.

The abbreviations used for the equipment location are as follows: RB is Reactor Building, AB is Auxiliary Building, RHR is RHR Building, DW is drywell, SB is sub-basement, B is basement, and the number represents the floor in the corresponding building. PIS is plant identification system.

The equipment categories below correspond to the ones in Table A-1 of EPRI NP-6041.

3.1.4.9.2.1 Motor Control Centers

Equipment Category: 1

Number of items on SSEL: 15

Location	PIS No.			
RB1	R1600S002B	R1600S004B		
RB2	R1600S003B	R1600S003D	R1600S005C	
AB2	R1600S002A			
AB3	R1600S005A	R3200S015	R3200S016	
AB5	R1600S003A	R1600S005D		
RHR2	R1600S016A	R1600S017A	R1600S018A	R1600S019A

Equipment Description

The main purpose of a motor control center (MCC) is to house the controls which turn motors on and off. They also may contain over-current relays to prevent system overheating and small transformers and distribution panels for lighting and 120V utility service. Motor control cubicles typically include the following types of components: molded case circuit breakers or disconnect switches, magnetic contactors, relays, control transformers, distribution panels, transfer switches, fuses, push buttons, and pilot lights. In addition, a horizontal bus bar runs near the top and through each section of an assembly.

The Fermi 2 MCCs are manufactured by ITE Gould Corporation as Model 5600 Series. Each MCC assembly consists of a series of vertical sections, each section being approximately 20 inches wide, 20 inches deep, and 90 inches high. The typical weight of each section is less than 650 pounds. The sections consist of an angle framework covered by 12-gage sheet metal. Adjacent sections are connected by bolting in the front and back and from bottom to top of the units. The assembly is bolted to an inverted mounting channel at the base which is, in turn, welded to an embedded leveling channel. Series 5600 MCCs conform to NEMA Type 12 specifications.

Each section of an assembly has a stack of individual control units, up to six NEMA size 1 or 2 combination starter units, each with its own door. The control units slide into the vertical sections on snap-in channel brackets and connect to the vertical bus bar. Doors are secured with slotted, quarter-turn, knurled fasteners.

Equipment Evaluation

All MCCs were walked down and visually inspected. A few cubicles were internally inspected. The EPRI NP-6041 screening criteria was used in the inspection. The inspection showed that the internal equipment was securely mounted and that the vertical sections were bolted to the mounting channels.

Based on the walkdown and review of MCC mounting drawings, the SRT concluded the following:

1. The MCCs are solidly constructed with a steel angle frame and sheet metal skin plates. The seismic load path appears adequate to transmit the loads through the units to the foundation. The cabinets do not appear excessively flexible.
2. All the doors and panel covers are secured by latches or fasteners.
3. There are no apparent excessively large cutouts in the lower half of the cabinets.
4. External enclosure weight is judged to be less than 100 pounds per cabinet.
5. Internal device mountings appear rugged, are not excessively flexible, and are attached properly to the cabinet.

In accordance with the guidelines in EPRI NP-6041, screening Tables 2-3 and 2-4 are primarily intended for components mounted less than 40 feet above grade in stiff nuclear power plant type structures. It is recommended in EPRI NP-6041 that care be exercised in the use of the tables. For this reason, MCCs were not considered screened out just by meeting the EPRI screening criteria. A bounding calculation and evaluation **[3.50]** were performed to show that the MCCs located on the AB fifth floor, as well as those located on lower floors, remain functional even at an elevation greater than 40 feet above grade when subjected to the RLE. This evaluation confirmed a HCLPF value of 0.3g or greater for all MCCs at Fermi 2. MCCs were selected as the most seismically vulnerable of the components on the SSEL. The evaluation is based on comparison of MCC test response spectra (TRS) and RLE required response spectra (RRS), clipped in accordance with Appendix Q of EPRI NP-6041. The results of the evaluation show that the MCCs are qualified for the RLE since the TRS enveloped the RRS at all but the lower frequencies below the MCCs' first natural frequency.

A concern has been raised in the industry about the connection detail for the MCC base mounting channel to the MCC assembly. Some damage to the connection bolts during testing has been reported **[3.52]**. SQUG documentation **[3.53]** recommends that if the MCC frame is connected to the external base mounting channel with internal bolts, there should be at least four, 3/8-inch diameter internal mounting bolts per section. An evaluation was performed for the Fermi 2 design basis earthquake which recommended the use of high-strength bolts, with ultimate tensile strength (UTS) greater than 100 ksi if the bolts were less than 3/8-inch diameter. The four, 5/16-inch bolts at the MCC corners are high-strength, ASTM A449 steel with an UTS of 105 ksi. The SRT evaluated the bolts for the RLE acceleration values and concluded that they are adequate to withstand the RLE loads **[3.51]**.

Anchorage Evaluation

The MCCs are constructed with mounting channels at their base. The mounting channels are either continuously or intermittently welded to leveling channels which are embedded in concrete. Anchorage Design Calculation DC-5634 **[3.73]** was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis anchorage evaluation. The calculation concluded that the anchorage is able to withstand the RLE loads.

System Interaction Effects

The area around the MCCs was examined by the SRT and with the exception of the outliers listed below, was free from interaction concerns. External cables and conduits connected to the MCCs were adequately flexible to accommodate relative movement. Generally the MCCs are sufficiently far from adjacent components and structures to preclude interaction with the MCCs. No potential sources of spraying or flooding were discovered that could affect sensitive equipment in the MCCs. The MCC units are bolted to adjacent units to preclude "banging" during a seismic event.

A channel-shaped, fire barrier partition approximately 6'-10" wide by 16'-8" long separates MCCs R3200S015 and R3200S016 on the third floor auxiliary building. A sufficient rattlespace, about 5 1/2", exists between the MCC and the partition to preclude interaction between the two components during an RLE seismic event.

A monorail trolley and hoist are mounted near the ceiling slab in each of the four RHR building switchgear rooms. Chains for hoisting and moving the trolley extend down to the floor where they rest in a bucket. The location of the bucket has been chosen to be away from the MCC and two switchgear assemblies in the room. Therefore, the bucket is about five feet away from any electrical assembly. The SRT reviewed a design basis calculation **[3.74]** which calculated a maximum chain displacement of eight inches under a design basis earthquake. Based on the expected maximum displacement of the chain and the available distance to adjacent components, the SRT concluded that this potential seismic interaction is acceptable as is.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Outlier resolutions are also provided.

1. About a one-inch rattlespace exists between stair structural steel and the top of MCC R1600S003B. The rattlespace was considered acceptable as-is by the SRT based on the stair support rigidity, resulting in small seismic displacement, and the estimated MCC displacement at its top **[3.51]**.
2. MCCs R1600S002A and R1600S005A are adjacent to modular power units (MPUs) but are not bolted to the MPUs. Provisions for bolting the MCCs to the MPUs will be included in Engineering Design Package (EDP) 27108 **[3.75]** which is scheduled for implementation by the end of the fifth refueling outage (RFO5).
3. Rattlespace violations exist between lifting angles atop MCC R1600S003D and a steel water shield. The gaps are 1/4" and 3/8". A work request (WR No. 000Z953621) was prepared to trim the lifting angles to produce an acceptable gap of about one inch. Work was completed in October 1995.
4. Two adjacent vertical sections in MCC R3200S016 were not bolted together. A work request (WR No. 000Z951314) was prepared to bolt the sections together. Work was completed in July 1995.
5. A wire-mesh cage, which rests on the RHR building second floor and protects seismic instrumentation, is in close proximity to MCC R1600S016A. An evaluation **[3.76]** was performed which showed that the cage would not slide during the RLE seismic event and, hence, would not impact the adjacent MCC.
6. A large air dryer tank, located several feet away from MCC R1600S005C on RB2, is supported on four small angles anchored to the floor slab with four 3/8-inch wedge anchors. A search for seismic mounting calculations was unsuccessful. Although the SRT did not have any MCC operability concerns, it was considered prudent to provide better seismic restraint for the tank. Therefore, Technical Service Request number 28,195 **[3.99]** was initiated recommending the addition of a top tank lateral support or other appropriate restraints.

3.1.4.9.2.2 Low Voltage Switchgear (LVS)

Equipment Category: 2

Number of items on SSEL: 8

Location	PIS No.			
AB2	R1400S022	R1400S023		
AB3	R1400S020	R1400S021		
RHR2	R1400S036	R1400S037	R1400S038	R1400S039

Equipment Description

A low voltage switchgear assembly consists of individual vertical sections bolted together through adjoining walls. The term "low voltage" refers to circuits of 600 volts or less, in this case 480 volts. The Fermi 2 LVSs are manufactured by ITE Gould Corporation. Each LVS assembly consists of switchgear sections, a transformer section, a voltage regulator section (four out of the eight units), transition sections, and bus terminal sections. Each of the switchgear sections is about 7'-6" high, 5'-8" deep, and 1'-6" to 2'-0" wide. A typical section weighs about 1300 pounds. The assemblies vary in length from about 12 feet to about 28 feet depending on their function and whether they include regulators. Each vertical section is a 14-gage or heavier steel sheet metal enclosure welded to a frame of steel angles or channels. Adjacent sections are connected by bolting in the front and back from bottom to top of the units. The section doors are secured in place by knurled bolting mechanisms at numerous locations along the height and by a locking mechanism near mid-height. The assembly is plug welded to embedded leveling channels at the base.

Each switchgear section contains a stack of two to four circuit breaker cubicles. The circuit breaker and other control devices are in a front compartment, and bus connections for the primary circuits are in the rear compartment. The vertical sections include ammeters, voltmeters, relays, transformers, disconnect switches, and distribution buses. The circuit breakers include electric contacts, closing solenoids, tripping devices, fuses and auxiliary switches.

The low voltage circuit breakers are the horizontally-racked, draw-out type in which they are mounted on a roller/rail support system that allows them to be disconnected from the primary contacts at the rear and rolled forward out of the compartment for maintenance. During operation, the circuit breaker clamps to the bus bars at the rear of the assembly.

On top of the vertical switchgear sections is a hoist and trolley structure which is used to assist in the removal and reinstallation of the individual circuit breaker cubicles.

Equipment Evaluation

All of the low voltage switchgear assemblies were observed and evaluated by the SRT. One LVS was selected to have a breaker door opened and the breaker rolled out. Based on the walkdown and review of LVS mounting drawings, the SRT concluded the following:

1. The LVSs are solidly constructed with a steel angle frame and sheet metal skin plates. The seismic load path appears adequate to transmit the loads through the units to the foundation. The cabinets do not appear excessively flexible.
2. All the doors and panel covers are secured by latches or fasteners.
3. There are no excessively large cutouts in the lower half of the cabinets.
4. External enclosure weight is judged to be less than 100 pounds per cabinet.

5. Internal device mountings appear rugged, are not excessively flexible, and are securely fastened to the cabinet. The drawers are mechanically fastened in the side-to-side and the front-to-back directions for restraint during seismic motion. The breaker lateral restraint was verified when the breaker door was opened and the breaker was rolled out.

Anchorage Evaluation

The low voltage switchgears are plug welded at the base to leveling channels which are embedded in concrete. The SRT confirmed that the plug welds are per design drawings in the compartments which were made accessible for the walkdown. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concluded that the anchorage is able to withstand the RLE loads.

System Interaction Effects

The area around each LVS was examined by the SRT and with the exceptions of the outliers listed below, was free from system interaction concerns. External cables and conduits connected to the LVS were adequately flexible to accommodate relative movement. Generally, the LVSs are sufficiently far from adjacent components and structures to preclude spatial interactions. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment in the LVSs. The LVS sections are bolted to adjacent sections to preclude their knocking into each other during a seismic event. A two-inch gap between the trolley rail atop the R1400S021 LVS and a duct stiffener was considered adequate by the SRT to prevent any interaction between the switchgear and the duct.

The overhead trolley and hoist chains adjacent to the switchgears on RHR2 have been evaluated for their potential impact on nearby equipment. As stated in Section 3.1.4.9.2.1, the chains were found to have no impact on the electrical equipment assemblies in the area.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each concern has been resolved as indicated.

1. Switchgear R1400S022 was inspected internally. The plug welds to the leveling channels were verified. In some cases, it appeared that Nelson studs were used to anchor the switchgear instead of plug welds. The Nelson studs were attached to the embedded channel and welded to the switchgear frame with all around fillet welds. The studs were cut off above the fillet welds. The SRT concluded that it was an acceptable alternate to the plug weld.

2. The trolleys atop each switchgear were not prevented from rolling in the side-to-side and front-to-back directions during a seismic event. Such movement could cause impact loading on the LVS which, in turn, could affect the behavior of any sensitive equipment inside the switchgear. An evaluation, documented in Detroit Edison File C1-4498 **[3.77]**, was prepared to design seismic restraints for the switchgear trolleys. Work requests (WR Nos. 000Z957668 through 000Z957675) were prepared to install the seismic restraints on the trolleys for the eight switchgear units.
3. For switchgears R1400S038 and S039, the adjacent switchgear section and regulator cabinet are not bolted together. The two components will be fastened together as part of EDP-27108 **[3.75]**, scheduled for completion by the next (fifth) refueling outage (RFO5).
4. Some of the bolts and bolting mechanisms for the switchgear doors closed were either stripped or not properly engaged. This condition was evident in switchgears R1400S038 and R1400S039. This could result in excessive vibration to the door-mounted relays in the switchgear. Work requests (WR Nos. 000Z952649 and 000Z952650) were prepared to repair the door fastener bolts. This work is complete.

3.1.4.9.2.3 Medium Voltage Switchgear (MVS)

Equipment Category: 3

Number of items on SSEL: 8

Location	PIS No.			
AB2	R1400S001B	R1400S001C		
AB3	R1400S001E	R1400S001F		
RHR2	R1400S002A	R1400S002B	R1400S002C	R1400S002D

Equipment Description

A medium voltage switchgear assembly consists of individual vertical sections bolted together through adjoining walls. The term "medium voltage" refers to circuits from 2400 volts to 4160 volts, in this case 4160 volts. The Fermi 2 MVSs are manufactured by ITE Gould Corporation. Each MVS consists of metal-clad sections bolted together to form an assembly. Each of the switchgear sections is about 7'-6" high, 5'-6" deep, and 2'-1" wide. A typical section weighs about 1300 to 1600 pounds. The assembly length is 8'-4" in the RHR building and 15'-2" or 19'-6" in the auxiliary building. Each vertical section is a 14-gage or heavier steel sheet metal enclosure welded to a frame of steel angles or channels. Adjacent sections are connected by bolting in the front and back from bottom to top of the units. The section doors are secured in place by knurled bolting mechanisms at numerous locations along the height and by a locking mechanism near mid-height. The assembly is plug welded to embedded leveling channels at the base.

The vertical sections house electrical switching and fault protection circuit breakers, control relays, internal transformers, junction boxes, and attached conduit and cables. The rear of the section contains a separate compartment for electrical connections. The circuit breakers are mounted on rollers which allow them to be wheeled in and out of the enclosures. The circuit breaker has clamping bus connections at the rear.

Equipment Evaluation

All of the medium voltage switchgear assemblies were observed and evaluated by the SRT. Based on the walkdown and review of MVS mounting drawings, the SRT concluded the following:

1. The MVSs are solidly constructed with a steel angle frame and sheet metal skin plates. The seismic load path appears adequate to transmit the loads through the units to the foundation. The assemblies do not appear excessively flexible.
2. All the doors and panel covers are secured by latches or fasteners.
3. There are no excessively large cutouts in the lower half of the cabinets.
4. Internal device mountings appear rugged, are not excessively flexible, and are securely fastened to the cabinet. The drawers are mechanically fastened in the side-to-side and the front-to-back directions for restraint during seismic motion.

A sheet metal box, about 2 feet by 2 feet by 2 feet and weighing approximately 170 pounds, rests on top of each switchgear in the RHR building. The box houses bus potential transformers. An evaluation **[3.51]** was performed to determine any effect the box could have on the qualification of the switchgear. Results of the evaluation demonstrate that the box has negligible effect on the response of the switchgear and the switchgear mounting. The connection to the switchgear was also found to be adequate. Based on its rugged construction, secure mounting to the MVS, and relative light weight, the SRT concluded that it will withstand the RLE.

Anchorage Evaluation

The medium voltage switchgears are plug welded at the base to leveling channels which are embedded in concrete. The SRT confirmed that the plug welds are per design drawings. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concluded that the anchorage is able to withstand the RLE loads.

System Interaction Effects

The area around each MVS was examined by the SRT and with the exceptions of the outliers listed below, was free from system interaction concerns. External cables and conduits connected to the MVS were adequately flexible to accommodate relative movement. Generally the MVSs are sufficiently far from adjacent components and structures to preclude interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment in the MVSs. The MVS sections are bolted to adjacent sections to preclude their knocking into each other during a seismic event.

The overhead trolley and hoist chains adjacent to the switchgears on RHR2 have been evaluated for their potential impact on nearby equipment. As stated in Section 3.1.4.9.2.1, the chains were found to have no impact on the electrical equipment assemblies in the area.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each concern has been resolved as indicated.

1. The four RHR building switchgear assemblies, PIS Nos. R1400S002A to D, are in close proximity to adjacent relay panels housing sensitive relays. An evaluation **[3.51]** was performed which showed that the rattlespace is sufficient to preclude interaction between the adjacent components.
2. Some of the bolts and bolting mechanisms for the switchgear doors were either stripped or not properly engaged. This condition was evident in switchgears R1400S001B, C and E and R1400S002A through R1400S002D. This could result in excessive vibration to the door-mounted relays in the switchgear. Work requests (WR Nos. 000Z954328 through 000Z954331 and 000Z952645 through 000Z952648) were prepared to repair the door fastener bolts. This work is complete.
3. A "Calvert bus" box in close proximity to switchgear R1400S001B was not supported in accordance with design documents. Two missing mounting bolts from the Unistrut support were replaced per WR No. 000Z951305.

3.1.4.9.2.4 Transformers and Regulators

Equipment Category: 4

Number of items on SSEL: 12

Location	PIS No.				
AB2	R1400S022A	R1400S023A			
AB3	R1400S020A	R1400S020B	R1400S021A	R1400S021B	
RHR2	R1400S036A	R1400S037A	R1400S038A	R1400S038B	R1400S039A
	R1400S039B				

Equipment Description

The 480V switchgear assemblies have associated bus transformers which step down the 4160V distribution voltage to 480V levels for component power. The four 1500KVA transformers feed essential equipment in the reactor/auxiliary building and the four, 750KVA transformers feed equipment in the RHR building. The indoor ventilated-dry transformers are Type VU-9 manufactured by ITE Gould. The transformer and its enclosure weigh about 6000 pounds. The enclosure dimensions are about 7'-6" wide, 5'-8" deep, and 8'-0" tall. The enclosure consists of sheet metal attached to steel angle framing. Each transformer is bolted to a channel mounting frame (skid) which, in turn, is plug-welded to embedded leveling channels. With the exceptions of some units in the RHR building, each enclosure is bolted to adjacent switchgear sections.

A regulator is an electrical device used to maintain current or voltage. It includes a rotor, stator, and an operating mechanism. Regulators are components of the Division 2 reactor building and RHR building switchgear assemblies. The regulators, manufactured by GE, are dry-type, three-phase, and motor-operated. The regulator assemblies in the auxiliary building weigh about 6300 pounds, while the ones in the RHR building weigh about 1900 pounds. The regulator enclosures in the AB are about 5'-10" wide, 3'-10" deep, and 4'-11" tall; those in the RHR building are about 2'-4" wide, 4'-4" deep, and 5'-9" tall. The regulator is housed in an enclosure consisting of sheet metal siding attached to steel angle framing. Each regulator is bolted to a channel mounting frame which, in turn, is welded to embedded leveling channels. With the exceptions of some units in the RHR building, each enclosure is bolted to adjacent switchgear sections.

Equipment Evaluation

All of the transformers and regulators were walked down and visually inspected, to the extent possible, by the SRT. Because the equipment is energized, the front enclosure panels were not removed for personal safety reasons. The SRT was able to look through louvered or expanded metal sections of the enclosure. One panel on transformer R1400S037A (RHR building) was removed for inspection. The limited inspection showed that the equipment is securely attached to the floor leveling channels.

An evaluation **[3.78]** was performed to determine whether the top of the transformer coils are adequately restrained to limit relative displacement between the coil assembly and the surrounding cabinet and to determine the stress levels in the structural components in various load paths. The evaluation showed that there is adequate clearance between the transformers and their enclosures to preclude interaction between the components and that stress levels in the structural components are within allowable values.

Anchorage Evaluation

The transformers and regulators are constructed with a structural mounting assembly at their base. The SRT ascertained that the welds between the mounting assembly and the leveling channels are in accordance with design drawings, except for outliers discussed below. Anchorage Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concluded that the anchorage is able to withstand the RLE loads.

Welds for the RHR building regulators were evaluated **[3.51]** by the SRT. Results of the evaluation showed that the existing welded attachments to the bases are adequate to withstand the RLE.

System Interaction Effects

The general areas around the transformers and regulators were reviewed by the SRT for any seismic interaction effects. With the exceptions of the outliers listed below, the SRT concluded that the areas are free from interaction concerns. External cables and conduits connected to the equipment are adequately flexible to accommodate relative movement. Generally the equipment is sufficiently far from adjacent components and structures to preclude any interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment. Some of the equipment enclosures are bolted to adjacent sections to preclude their knocking into each other during a seismic event.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each outlier has been resolved as indicated.

1. The RHR regulators are not fastened to the adjacent switchgear and transformer sections on both sides. The adjacent enclosures will be fastened together as part of EDP-27108, scheduled for completion during RFO5.
2. The design drawings for RHR building transformers R1400S036A to R1400S039A, show connection details consisting of fillet welds as well as four plug welds, one near each corner. The SRT noticed during a field walkdown that these four plug welds are not installed. Review of the connection calculation shows a large margin between the design and allowable stress values due to the fillet welds. Therefore, it is concluded that the attachment of the transformers to their bases has adequate capacity to withstand the RLE **[3.51]**.
3. Some door latching bolts on the front and back of units R1400S020A and R1400S020B required tightening or replacement. This work was completed as part of Work Request No. 000Z954332.

3.1.4.9.2.5 Horizontal Pumps

Equipment Category: 5

Number of items on SSEL: 21

Location	PIS No.				
RBSB	E5101C001	E5101C004			
RB2	P4400C001A	P4400C001B			
ABSB	E4101C001A	E4101C001B	E4101C001C	E4101C003	E4101C005
AB5	T4100B008A	T4100B009A	T4100C040	T4100C041	

Location	PIS No.				
RHR1	R3000C001	R3000C002	R3000C003	R3000C004	R3000C009
	R3000C010	R3000C011	R3000C012		

Equipment Description

The pumps in this category are in the HPCI, RCIC, EECW, Control Center HVAC, and EDG systems. They are used for pumping water, oil, and fuel oil. Their motor sizes range from one to 100 horsepower and capacities from seven to 5000 GPM. Several of the pumps are driven by steam turbines in the HPCI and RCIC systems. The pump manufacturers include Bingham, Nash, Crane Deming, Byron Jackson, Delaval Turbine, Tuthill Pump, Goulds Pump, and Viking.

Equipment Evaluation

The SRT walked down and visually inspected all of the pumps in this category, except as described in the Outlier section below. The screening criteria of EPRI NP-6041 was used to evaluate the pumps.

Based on the walkdown and review of pump mounting drawings, the SRT concluded the following:

1. The driver and pump are attached to a common, stiff skid (except as discussed in the Outlier section below).
2. The support structure is adequate to resist lateral loads.
3. There is no concern for excessive nozzle loading resulting from gross pipe motion or differential displacement and prying about a rigid pipe support.
4. There are no adjacent unsupported or lightly supported in-line components or long unsupported pipe spans.

The Control Center HVAC chiller oil pumps were considered seismically acceptable by the SRT based on review of vendor seismic test reports and comparison of test response spectra with RLE required response spectra.

Anchorage Evaluation

In general, the pumps rest atop concrete pedestals or pads and are securely anchored with embedded anchor bolts. In most installations, there is sufficient anchor edge distance and adequate embedment into the base concrete. Pedestals and pads are doweled into the concrete floor slabs. Anchor spacing meets minimum requirements. The anchorage conforms to the design drawings. The nuts are present and apparently tight. The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concludes that the anchorage is able to withstand the RLE loads.

The HPCI and RCIC barometric condenser vacuum pumps are rigidly bolted directly to the condensers. The attachments are adequate to withstand the RLE loads.

System Interaction Effects

The general areas around the pumps were reviewed by the SRT for any seismic interaction effects. With the exceptions of the outliers listed below, the SRT concluded that the areas are free from system interaction concerns. Attached lines are adequately flexible to accommodate relative movement. Generally the pumps are sufficiently far from adjacent components and structures to preclude any interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each concern has been resolved as indicated.

1. The SRT discovered two missing bolts which attach the HPCI turbine-driven oil pump foot to its support springs. The bolts were replaced as part of Work Request No. 000Z951289.
2. A small edge distance (2 1/8") exists between one of the hold-down bolts for fuel oil transfer pump R3000C004 and the edge of the concrete pedestal. Review of DC-5634 shows an edge distance evaluation for 2 1/4"; however, sufficient margin exists between allowable and design values to conclude that the 2 1/8" edge distance is sufficient to preclude side bursting. Therefore, the pump anchorage is sufficient to withstand the RLE loads.
3. The Control Center HVAC chiller oil pumps are located inside the oil sump. The pump mounting was considered acceptable based on a review of drawings and the seismic test report.

3.1.4.9.2.6 Vertical Pumps

Equipment Category: 6

Number of items on SSEL: 16

Location	PIS No.				
RBSB	E1102C002A	E1102C002B	E1102C002C	E1102C002D	E5101C003
ABSB	E4101C004				
RHR1	E1151C001A	E1151C001B	E1151C001C	E1151C001D	P4500C002A
	P4500C002B	R3001C005	R3001C006	R3001C007	R3001C008

Equipment Description

The pumps in this category are in the RHR, RHRSW, EESW, DGSW, HPCI, and RCIC systems. The motors are manufactured by GE and Allis-Chalmers and range in size from 3 to 2250 horsepower. Pump capacities range from 35 to 10,000 GPM. The four, 10,000-GPM RHR pumps are single stage, vertically mounted, centrifugal pumps manufactured by Byron Jackson. The RHR pumps draw their suction from the water in the torus. The RHRSW, EESW, and DGSW pumps are deep well pumps in which the pump impeller is attached to the end of a long vertical shaft extending below the pump base plate and submerged in water. These pumps draw their suction from the RHR reservoir. They are manufactured by Gould Pumps. The HPCI and RCIC pumps, manufactured by Nash, are condensate pumps mounted to the barometric condensers. Their function is to remove water from the condenser vacuum tanks.

Equipment Evaluation

The SRT walked down and visually inspected all of the pumps in this category. The screening criteria of EPRI NP-6041 was used to evaluate the pumps.

Based on the walkdown and review of pump mounting drawings, the SRT concluded the following:

1. The base plates are not excessively flexible. The pumps are free of intermediate flexible bases.
2. The impeller drive shafts are supported within the casing.
3. There is no concern for excessive nozzle loading resulting from gross pipe motion or differential displacement and prying about a rigid pipe support.
4. There are no adjacent unsupported or lightly supported in-line components or long unsupported pipe spans.

Anchorage Evaluation

Each RHR pump is welded to a rigid base plate which is securely anchored with embedded anchor bolts. The other pumps in this category are welded to retaining rings which are, in turn, securely anchored to concrete pedestals. There is generally sufficient anchor edge distance on pedestals and anchor spacing meets minimum requirements. Pedestals are doweled into the concrete floor slabs. The anchorage conforms to the design drawings. The nuts are present and apparently tight. The anchorage appears to be relatively stiff and free of gaps under the base. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. Anchor edge distance violations were evaluated and accepted in this calculation. The calculation concludes that the anchorage is able to withstand the RLE loads.

The HPCI and RCIC condenser pumps are rigidly bolted directly to the barometric condensers. The attachments are adequate to withstand the RLE loads.

System Interaction Effects

The general areas around the pumps were reviewed by the SRT for seismic interaction effects. With the exceptions of the outliers listed below, the SRT concluded that the areas are free from interaction concerns. Attached lines are adequately flexible to accommodate relative movement. Generally, the pumps are sufficiently far from adjacent components and structures to preclude any interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment.

Outlier and Outlier Resolution

The following concern was identified during the plant walkdowns. Each concern has been resolved as indicated.

The RHR building deep-well pump impellers and casings extend about 40 feet into the RHR reservoir. The pump PIS numbers are shown above. The shafts are well supported within the casings; however, there are no additional supports for the impellers and casings below the baseplates. A finite element analysis **[3.79]** was performed to evaluate shaft and casing stresses and impeller drive shaft deflection. The evaluation showed that the casings and other pump internals are not overstressed as a result of the seismic margin earthquake and that the deflection at the ends of the casings would not affect the operability of the pumps.

3.1.4.9.2.7 Fluid-Operated Valves

Equipment Category: 7

Number of items on SSEL: 281

Location	PIS No.				
DWB	B3100F014A	B3100F014B	T4800F455		
DW1	B2100F010A	B2100F010B	B2103F022A	B2103F022B	B2103F022C
	B2103F022D	B2104F013E	B2104F013H	B2104F013J	B2104F013P
	B2104F013R	B21F022A	B21F022B	B21F022C	B21F022D
	E1100F050A	E1100F050B	E11F610A	E11F610B	P4400F245A
	P4400F245B	T2300F400A	T2300F400B	T2300F400C	T2300F400D
	T2300F400E	T2300F400F	T2300F400G	T2300F400H	
DW2	T4901F021	T4901F024	T4901F027	T4901F030	T4901F033
RBSB	E1100F030A	E1100F030B	E1100F030C	E1100F030D	E5100F017
	E5100F018	E5150F025	E51F004	E51F015	E51F025
RBB	C1100F011	C1100F181	E1100F025A	E1100F025B	E1100F029
	T2300F409	T2300F410	T23F409	T23F410	T4800F453

Location	PIS No.				
RBB	T4800F454	T4800F456	T4800F457	T4800F458	
RB1	B2100F076A	B2100F076B	B2103F028A	B2103F028B	B2103F028C
RB!	B2103F028D	B21F028A	B21F028B	B21F028C	B21F028D
	B3100F016A	B3100F016B	C1100F010	C1100F180	C11F160A
	C11F160B	C11F162A	C11F162B	C11F162C	C11F162D
	C11F163A	C11F163B	C11F182A	C11F182B	C11F409A
	C11F409B	E1100F056A	E1100F056B	E1100F078	P34F401B
	T4901F465	P50F519A	P50F519B		
RB2	E1100F001A	E1100F001B	E11F412	E11F413	E11F414
	E11F415	P4400F125A	P4400F125B	P4400F126A	P4400F126B
	P4400F142A	P4400F142B	P44F400A	P44F400B	P44F402A
	P44F402B	P44F403A	P44F403B	P4500F141A	P4500F141B
	T4800F451	T4901F468	T5000F455	T5000F456	T50F450
	T50F451				
ABSB	E4100F020	E4100F026	E4100F028	E4100F050	E4100F053
	E41F035	E41F200	P5000F207A	P5000F207B	P5000F223A
	P5000F223B				
ABB	P5000F440	P5000F441	P5000F541A	P5000F541B	P5000F542A
	P5000F542B	P5002D029A	P5002D029B		
AB4	T4100F031A	T4100F031B	T4100F033A	T4100F033B	T4100F035
	T4100F038	T4100F041	T4100F042	T41F084A	T41F084B
	T41F085A	T41F132	T41F134	T41F164	T41F183
	T41F185	T41F189	T41F191		
AB5	T4100F039B	T4100F040B	T4100F068A	T4100F068B	T4100F069A
	T4100F069B	T41F071A	T41F071B	T41F074A	T41F074B
	T41F083B	T41F086A	T41F086B	T41F088A	T41F089A
	T41F103A	T41F103B	T41F104A	T41F104B	T41F111A
	T41F111B	T41F114A	T41F114B	T41F142	T41F143
	T41F144	T41F145	T4100F157A	T4100F157B	T4100F158A
	T4100F158B	T4100F159A	T4100F159B	T4100F160A	T4100F160B
	T41F160	T4100F161A	T4100F161B	T41F161	T4100F162A
	T4100F162B	T41F162	T4100F163A	T4100F163B	T41F181
	T41F182	T41F187	T41F188	T41F382A	T41F382B
	T41F384A	T41F384B			
RHR1	R3000F035A	R3000F035B	R3000F035C	R3000F035D	R3000F036A
	R3000F036B	R3000F036C	R3000F036D	R3000F096A	R3000F096B
	R3000F096C	R3000F096D	R3000F111A	R3000F111B	R3000F111C
	R3000F111D	R30FA04A	R30FA04B	R30FA04C	R30FA04D
	R30FA05A	R30FA05B	R30FA05C	R30FA05D	X4103F157
	X4103F159	X4103F161	X4103F162	X4103F164	X4103F166
	X4103F168	X4103F169			
RHR2	X4103F103	X4103F104	X4103F106	X4103F108	X4103F109
	X4103F115	X4103F116	X4103F118	X4103F120	X4103F121
	X4103F127	X4103F128	X4103F130	X4103F132	X4103F133

Location	PIS No.				
RHR2	X4103F139	X4103F140	X4103F142	X4103F144	X4103F145
	X4103F149A	X4103F149B	X4103F149C	X4103F149D	X4103F150
	X4103F151A	X4103F151B	X4103F151C	X4103F151D	X4103F152
	X4103F153A	X4103F153B	X4103F153C	X4103F153D	X4103F154
	X4103F155A	X4103F155B	X4103F155C	X4103F155D	X4103F156

Equipment Description

This equipment class includes a wide variety of valve sizes, types, and applications. Most of the valves are actuated by air and only a few valves are actuated by water. The three main types of fluid-operated valves are diaphragm-operated, piston-operated, and pressure relief valves. In general, a valve in this category functions by means of a pressure differential across an internal diaphragm. A return spring controls the actuated rod or valve stem. The actuated rod position, in turn, controls the valve position. A solenoid valve controls the air pressure across the diaphragm. A piston-operated valve is similar to a diaphragm-operated valve except that a piston replaces the diaphragm as the valve actuator. A pressure relief valve balances confined fluid pressure against a spring force. Valve assemblies in this category generally include air lines, pneumatic relays, control solenoids, and conduit.

The Fermi 2 fluid-operated valves in this category include the following types and manufacturers:

1. Solenoid valves by ASCO, Valcor, Target Rock, Skinner, and Automatic Valve
2. Relief valves by Crosby, Fisher, Kunkle, Aquatrol and Pall Pneumatics
3. Testable check valves by Anchor Valve and Anchor-Darling
4. MSIVs by Atwood & Morrell
5. Torus-to-drywell vacuum breakers by GPE Controls and Neles-Jamesbury
6. Damper actuators by Shan Rod, Powers Regulator, Centerline, and ITT General Controls
7. Air-operated valves by Rockwell, Target Rock, Copes-Vulcan, Fisher, Jamesbury, Poweli and ASCO
8. Regulating valves by Sterling and Johnson Controls and Marotta Scientific Controls

The Fermi 2 valves range in size from less than one-inch relief valves to the 26-inch main steam isolation valves.

Equipment Evaluation

The SRT walked down all of the fluid-operated valves in this category with the exception of those few which were inaccessible due to interferences or ALARA or operability considerations. For those valves, extensive drawing and other documentation reviews were performed to address configuration caveats and potential spatial interaction issues. The screening criteria of EPRI NP-6041 was used to evaluate the valves. In a number of cases, the applicable valve stress reports were reviewed to evaluate the valves for margin to accommodate the RLE.

Based on the walkdown and documentation review, the SRT concluded the following:

1. In general, the valves are mounted to pipe sizes one-inch or greater. When a valve is mounted to a small pipe, the pipe in the vicinity of the valve and, in most cases, the valve actuator are well supported to a common structure to prevent excessive displacement.
2. In the cases where the valve operator and the pipe are both supported, supports are braced to a common structure. In other cases, the operator and yoke are supported by the pipe without any other bracing.
3. The valve body, bonnet, yoke, and operator supports are not made of cast iron.
4. The distance from the pipe centerline to the top of the valve operator generally meets the screening criteria of EPRI NP-6041, Figure F-25. In cases where the distance was greater than the screening values, the valves were individually evaluated to assure that they were not overstressed.

Rockwell air-operated valves with PIS Numbers B3100F014A and B and B3100F016A and B are attached to less-than-one-inch diameter piping. For these cases, the valves and actuators were screened out by the SRT based on their light weights, the fact that the pipe and actuator are well supported, and a comparison of applicable RLE seismic accelerations with seismic qualification accelerations.

Solenoid valves B21F022A to D manifolds are mounted to MSIVs B2103F022A to D, respectively. The manifolds weigh about 100 pounds each and are mounted about 45 inches above the MSIV bonnets with four 1/2-inch mounting bolts. The valves and manifolds, forming the pneumatic control assemblies, were considered seismically acceptable by the SRT based on a review of the seismic qualification test report which shows large margins between the test response spectra and the required response spectra.

System Interaction Effects

The general areas around the valves were reviewed by the SRT for seismic interaction effects. For the few inaccessible valves, the Fermi 2 rattlespace program documentation and photographs from recent outages were used for seismic interaction screening. If no documented rattlespace violations were found, the SRT concluded that the valves were free from interaction concerns. Attached tubing, piping, and conduits were reviewed and found to have adequate flexibility to accommodate relative movement. Generally the valves were sufficiently far from adjacent structures, walls, overhead equipment and distribution systems to preclude any interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each concern has been resolved as indicated.

1. On temperature control valve P44F400A on RB2, the handwheel is about 1/4" below an insulated pipe. Since any impact between the two components would be absorbed by the pipe insulation, the SRT judged that such impact would not affect the valve's function.
2. The flange on relief valve P4400F125B on RB2 is about 1/2" away from an insulated vertical pipe. Since any impact between the two components would be absorbed by the pipe insulation, the SRT judged that such impact would not affect the valve's function.
3. The top of valve E1100F078 in the south RHR heat exchanger room is about 1/2" from a platform steel beam. Based on an evaluation **[3.51]** by the SRT, the sum of the displacements of the interacting components is less than the available rattlespace; therefore, the interaction is acceptable as-is.
4. The actuator of valve P5000F441 in the AB basement is about 1/2" from an adjacent pipe. The existing rattlespace was found acceptable by the SRT based on the low anticipated seismic movement of the two components **[3.51]**.
5. Four damper actuators (PIS Nos. X4103F150, F152, F154, and F156) on the second floor of the RHR building are located near concrete wall cutouts which were made to provide clearance for the actuator housing and tubing. The resulting clearances vary from 1/4" to 1/2". Based on the high natural frequency and relatively rigid mounting of the actuator, the SRT concluded that impact with the wall is not likely. In the event that there is some impact, the SRT concluded that the actuators will remain functional based on their rugged construction, relatively flexible linkage, and no apparent sensitive parts inside the actuators **[3.51]**.
6. Eight damper actuators (PIS Nos. X4103F106, F108, F118, F120, F130, F132, F142, and F144) on the second floor of the RHR building are located near duct supports. The clearances vary from "none" to about one inch. The relative movement of the cantilevered portion of the actuator housing will be very small due to the light mass of the actuator. The SRT concluded that the interaction between the components is acceptable based on the high seismic accelerations at which the actuator was proof-tested and remained functional. The high test accelerations would bound loading resulting from the interaction**[3.51]**.
7. Solenoid valves B21F028A to D, located in the main steam tunnel, and B21F022A to D, located in the drywell, were added to the SSEL after the walkdowns for the MSIVs in these areas. They were inaccessible due to ALARA and plant operating conditions at the time of their addition to the SSEL. Seismic interaction screening was based on the prior walkdowns of the inboard and outboard MSIVs and review of the Fermi 2 rattlespace program; no rattlespace violations were identified for the MSIVs on which they are mounted.
8. Two seismic interaction concerns were identified for valve E41[illegible] in the HPCI room: (1) the yoke support steel is about 3/8" from the handwhe[illegible] another valve,

and (2) the valve actuator cover is about 1/2" from the yoke support steel for another valve. The SRT considered these gaps acceptable since the valve and piping are well-supported and the seismic levels in the AB sub-basement are low **[3.51]**.

9. Numerous damper switching, regulating and control valves, on the fourth and fifth floors of the auxiliary building, were not closely examined by the SRT since they were inaccessible. They were considered acceptable by the SRT based on examination of similar valves on an accessible damper.
10. Solenoid valves T41F071A and B on the CCHVAC chiller skids have small electrical boxes attached to their operators. Based on review of the seismic test report for the chiller skid mounted components and comparison of fragility test results with the RLE required response spectra, the SRT considered these configurations acceptable **[3.51]**.
11. A loose mounting bolt was found on pressure control valve T41F114B on AB5. Work request (WR No. 000Z957680) was prepared to tighten the mounting bolt.
12. A 3/4-inch rattlespace exists between 3/8-inch tubing emanating from the actuator for torus vacuum breaker valve T2300F409 and adjacent 5/8-inch tubing. An evaluation [3.33] was performed by the SRT which showed that, although the two components may interact with each other, the interaction will not result in overstressing the tubing.
13. A 3/4-inch rattlespace exists between solenoid valve T50F450 and adjacent 3/8-inch tubing. An evaluation **[3.51]** was performed by the SRT to show that the two components will not interact with each other during the RLE.
14. A search through the Fermi 2 rattlespace program documentation identified a rattlespace concern for valve P4400F245B. An evaluation was performed to show that the existing rattlespace is sufficient to prevent interaction between this relief valve and an adjacent pipe **[3.51]**.

3.1.4.9.2.8 Motor-Operated Valves

Equipment Category: 8

Number of items on SSEL: 89

<u>Location</u>	<u>PIS No.</u>				
DWB	G1154F018	G1154F600	P4400F608	P4400F614	P4400F616
DW1	E1150F009	E1150F608	E4150F002	E5150F007	G3352F001
	T4901F601				
DW2	T4901F602				
RBSB	E1150F004A	E1150F004B	E1150F004C	E1150F004D	E1150F006A
	E1150F006B	E1150F006C	E1150F006D	E4150F012	E4150F042
	E5150F029	E5150F031	E5150F045	E5150F046	E5150F059
	G5100F600	G5100F601	G5100F602	G5100F603	P4400F605A
	P4400F605B				
RBB	E1150F007A	E1150F007B	E1150F017A	E1150F017B	E1150F024A
	E1150F024B	E1150F028A	E1150F028B	E1150F611A	E1150F611B
	E2150F031A	E2150F031B	E4150F075	E5150F019	E5150F062

Location	PIS No.				
RBB	G5100F604	G5100F605	G5100F606	G5100F607	P4400F606A
	P4400F606B	P4400F607A	P4400F607B		
RB1	E1150F008	E1150F015A	E1150F015B	E1150F048A	E1150F048B
	E4150F003	E4150F006	E5150F008	E5150F013	G3352F220
	P4400F601A	P4400F601B	P4400F603A	P4400F603B	
RB2	E1150F068A	E1150F068B	G3352F004	P4400F602A	P4400F602B
ABSB	E4150F001	E4150F004	E4150F041	E4150F59	
ABB	P4400F604				
AB3	P4400F613				
AB5	T41F072A	T41F072B	T41F073A	T41F073B	
RHR1	R3000F601	R3000F603	R3000F605	R3000F607	

Equipment Description

A motor-operated valve (MOV) is a valve actuated by an electric motor. Components of the MOVs include a motor operator with a control box, gear box, and drive motor. The gear box includes the gears which link the valve actuator to the drive motor shaft. Local controls include a relay for primary circuit actuation and torque and limit switches for coordinating the drive motor with the valve position. The valve which is actuated by the motor operator could be of any type, size, or orientation. The motor operators could be cantilevered above, below, or at the side of the valve.

Most of the MOVs in this category are manufactured by Powell Valve Company with others by Rockwell-Edwards Valve Company and Anchor-Darling Valve Company. They are the gate and globe valve types and range in size from 1 1/2" to 24". The only valves smaller in size are two 3/4-inch valves, manufactured by Trane, which are mounted on the AB fifth floor CCHVAC chiller skid. The valve actuators are manufactured by Limitorque and their motors by either Reliance Electric or H. K. Porter. They primarily serve as isolation valves in various plant systems and locations.

Equipment Evaluation

The SRT walked down all of the motor-operated valves in this category with the exception of those few which were inaccessible due to interferences or ALARA considerations. The screening criteria of EPRI NP-6041 was used to evaluate the valves.

Based on the walkdown and documentation review, the SRT concluded the following:

1. With the exception of the two chiller skid 3/4-inch valves, the valves are mounted to 1 1/2-inch or greater pipe. When a valve is mounted to a small pipe, the pipe in the vicinity of the valve and, in most cases, the valve actuator are well supported to prevent excessive displacement. (The SRT looked for cases where large extended operators were attached to two-inch or smaller piping per the screening criteria of

EPRI NP-6041, Table 2-4. In all cases, the SRT judged the actuators and piping to be adequately supported.)

2. In some cases the valve operator and the pipe are both supported by supports braced to a common structure. In other cases, the operator and yoke are supported by the pipe without any other bracing.
3. The valve body, bonnet, yoke, and operator support are not made of cast iron.
4. In most cases, the valve weight and eccentricity (i.e., the distance from the pipe centerline to the top of the motor actuator) conform to the requirements of EPRI NP-6041, Figure F-26. Cases where the distance and/or the weight were greater than the screening values were evaluated to assure that the valve and piping were not overstressed. The evaluations were based on a comparison of the allowable acceleration values (i.e., qualified accelerations) with the required RLE accelerations. If necessary, other techniques permitted in EPRI NP-6041 (e.g., "100-40-40" rule, higher allowable stresses) were employed to screen out the valves.

System Interaction Effects

The general areas around the valves were reviewed by the SRT for seismic interaction effects. In a few cases where valves were inaccessible due to congestion, the Fermi 2 rattlespace program documentation was used for seismic interaction screening. If no documented rattlespace violations were found, the SRT concluded that the valves are free from interaction concerns. Attached tubing, piping, and conduits were reviewed and found to have adequate flexibility to accommodate relative movement. Generally the valves are sufficiently far from adjacent structures, walls, overhead equipment and distribution systems to preclude any interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each concern has been resolved as indicated.

1. A block shield wall is in close proximity to valve G3352F220 in the steam tunnel on the RB first floor. Based on the wall configuration, a block wall bounding calculation **[3.53]**, the SRT concluded that the block wall will not fall and encroach upon the valve.
2. The handwheel on valve E1150F024B is about 1" from a cable tray. The interacting components are not in the same plane horizontally. The SRT judged that the components would not interact **[3.51]**.
3. The operator on valve G5100F600 near the floor of the torus room is separated by about 1" from the torus. An evaluation **[3.51]** by the SRT showed that the sum of the displacements of the interacting components is less than the available rattlespace. Therefore, the interaction is acceptable as-is.
4. The actuator de-clutch handle on valve E5150F045 in the RB sub-basement RCIC quad room is about 3/4" from an electrical box. Based on low seismic accelerations in

the RB sub-basement and the fact that only the actuator would experience any significant movement during a seismic event, the SRT judged that the existing rattlespace is acceptable as-is **[3.51]**.

5. The insulation on valve P4400F602B touches the nameplate on the adjacent EECW make-up tank P4400A002 on RB2. The SRT concluded that any interaction between the two components would be absorbed by the insulation and the valve's function would not be affected **[3.51]**.
6. A small gap of 3/16" exists between the flange on valve E4150F041 in the AB sub-basement HPCI room and another valve in a branch line. Because the piping is well supported next to each valve, the SRT concluded that the existing rattlespace between the two components is acceptable as-is **[3.51]**.
7. The operator weights for valves E1150F048A and E1150F017A and B are greater than the allowable weight for MOV screening per EPRI NP-6041, Table F-26. To prescreen the valves, the SRT investigated (1) the system operating logic to determine when the valves would be required for operability and (2) the duration of the seismic event. It was determined that the valve would only be required to stroke after, and not during, the relatively short-duration RLE. Therefore, the seismic and thrust loads are not coincident and the SRT concluded that the valves could withstand the RLE without a loss of function **[3.51]**.
8. A 5/8-inch gap exists between valve E1150F048A position indicator and an adjacent structural steel platform beam in the south RHR heat exchanger room. An evaluation **[3.51]** by the SRT showed that the position indicator rod could impact the steel beam during a seismic event resulting in a possible bent rod. However, because of the very large thrust capacity of the valve and the low required thrust, it was concluded that a small bend in the rod would not have any adverse effect on the valve's ability to open or close during or after the seismic event.
9. Walkdown of valve E4150F002, located in the drywell, was not possible since it was added to the SSEL after the drywell was closed for operation. The Fermi 2 rattlespace program documentation was reviewed to determine whether there are existing rattlespace violations associated with this valve. None was found. The valve was screened by the SRT based on a documentation review.
10. Valve E5150F059 in the ABSB RCIC quad room has a 3-inch inlet and a 4-inch outlet. The average weight and height limits per EPRI NP-6041 were used for valve screening. Since the valve met the intent of the screening criteria, it was screened by the SRT **[3.57]**. The throttle mechanism attached to the valve appeared to be light in weight, rugged and well supported.
11. Control Center HVAC chiller guide vane actuator valves T41F072A and B do not have the typical MOV configuration. However, the actuator was well supported to the main chiller unit and the rods and linkages from the actuator to the valves appeared rugged. In addition, review of the seismic test report showed that the fragility test response spectra fully envelope the RLE required response spectra. Therefore, the SRT considered the valves and actuators adequate to withstand the RLE loading **[3.51]**.
12. Control Center HVAC chiller compressor flow control valves T41F073A and B do not have the typical MOV configuration. The actuator is positioned perpendicular to

the valve stem and controls the valve with short linkages. The actuator weight and eccentricity are within the earthquake experience database limitations. In addition, review of the seismic test report showed that the fragility test response spectra fully envelope the RLE required response spectra. Therefore, the SRT considered the valves and actuators adequate to withstand the RLE loading **[3.51]**.

3.1.4.9.2.9 Fans

Equipment Category: 9

Number of items on SSEL: 26

Location	PIS No.				
AB5	T4100C030	T4100C031			
RHR1	X4103C017	X4103C018	X4103C019	X4103C020	
RHR2	E1156C001A	E1156C001B	E1156C001C	E1156C001D	X4103C001
	X4103C002	X4103C003	X4103C004	X4103C005	X4103C006
	X4103C007	X4103C008	X4103C009	X4103C010	X4103C011
	X4103C012	X4103C013	X4103C014	X4103C015	X4103C016

Equipment Description

Three types of fans are considered in this category; Control Room HVAC return air fans, RHR mechanical draft cooling tower fans, and RHR building ventilation fans. The return air fans are centrifugal fans manufactured by Trane, about 85 inches in diameter and driven by a 25-HP motor. The cooling tower fans are manufactured by the Marley Company and are 240 inches in diameter, four-bladed and driven by a 150-HP, two-speed motor. The ventilation fans are three different models manufactured by Buffalo Forge and ranging in capacity from about 3900 SCFM to 34,000 SCFM. The fans draw mixed air (i.e., fresh, outside air and recirculated air) into the diesel generator rooms, switchgear rooms, and pump rooms.

Equipment Evaluation

The SRT walked down all of the fans in this equipment category. The screening criteria of EPRI NP-6041 was used to evaluate the fans. Since the ventilation fans were enclosed in fan housings, they were not directly observed by the SRT. Also, the SRT was unable to perform a close visual inspection of the cooling tower fans and their gearboxes and shafts due to personal safety considerations.

Based on the walkdown and a review of fan mounting drawings, the SRT concluded the following:

1. The drive motor and fan are typically connected to a stiff common base. The RHR cooling tower fans and motors have separate bases and are connected by a drive shaft. The X41 fans have internal motors.
2. The housings surrounding the fans are stiff and well supported to preclude excessive distortion of the housings which could cause damage to fan blades.
3. The ventilation fans are mounted on vibration isolators which have seismic stops to limit lateral movement.
4. Access doors are secured by latches or fasteners.

Anchorage Evaluation

The RHR building ventilation fan units and cooling tower fan units are securely anchored to concrete pedestals with embedded anchor bolts. The return air fans are mounted to a rigid steel frame which is anchored to the concrete slab. There is generally sufficient anchor edge distance on pedestals. Pedestals are doweled into the concrete floor slabs. Anchor spacing meets minimum requirements. The anchorage conforms to the design drawings. The nuts are present and apparently tight. The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. DC-5634 also evaluates the cooling tower fan gearbox and motor anchorage. The calculation concludes that the anchorage has adequate capacity to withstand the RLE loads.

System Interaction Effects

The general areas around the fans were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas were free from interaction concerns. Attached lines were adequately flexible to accommodate relative movement. The fans were typically located away from adjacent components and structures to preclude any interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment.

Outliers and Outlier Resolution

No concerns were identified for the equipment in this category.

3.1.4.9.2.10 Air Handlers

Equipment Category: 10

Number of items on SSEL: 16

Location	PIS No.		
RBSB	T4100B018	T4100B019	T4100B021
RB2	T4100B034	T4100B035	
ABSB	T4100B022		

Location	PIS No.			
ABB	T4100B029	T4100B030		
AB2	T4100B002	T4100B003		
AB3	T4100B004	T4100B005	T4100B043	T4100B044
AB5	T4100B006	T4100B007		

Equipment Description

The air handlers (i.e., room coolers) in this category consist of a sheet metal enclosure on a steel member frame containing, as a minimum, a fan and heat exchanger or coil. Centrifugal fans blow air across the coil for heat transfer. The assemblies also include piping for cooling water, electrical conduits, and instrument lines. The room coolers are manufactured by Porter, Cryenco, CTI Nuclear, Trane, and Philips. They are driven by motors which vary in size from five to 40 HP. The Control Center HVAC supply air fan units are supported by a steel frame and have additional internal components, including filters, air mixing boxes, and dampers.

Equipment Evaluation

The SRT walked down all of the air handlers in this category. The screening criteria of EPRI NP-6041 was used to evaluate the units. The SRT was unable to perform a close visual inspection of the Control Room air handlers and related internal components due to their inaccessibility and for personal safety considerations.

Based on the walkdown and a review of air handler mounting drawings, the SRT concluded the following:

1. The drive motors are firmly connected to the outside of the fan housings.
2. Enclosures surrounding the units are stiff and well supported to preclude excessive distortion of the enclosures which could cause damage to fan blades.
3. Access doors are secured by latches or fasteners.
4. Internal devices are securely mounted and appear seismically adequate.
5. The gravity air dampers included with the dampers are securely mounted either in ducts or concrete openings. The dampers appear to be seismically rugged units.

Anchorage Evaluation

The air handlers, with the exception of the supply air fans, are anchored to the concrete floor slabs or pads with wedge anchors. Three of the air handler units are suspended from floor slabs. The supply air fans are mounted to a rigid steel frame which is anchored to the concrete slab with wedge anchors. Anchor spacing meets minimum requirements. The anchorage conforms to the design drawings. The nuts are present and apparently tight. The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concludes that the anchorage is able to withstand the RLE loads.

System Interaction Effects

The general areas around the air handlers were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns. Attached lines are adequately flexible to accommodate relative movement. The air handlers are typically located away from adjacent components and structures to preclude any interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each concern has been resolved as indicated.

1. Four room coolers (PIS Nos. T4100B002 through T4100B005) are mounted on C6 channels which are flexible in the direction perpendicular to the web. An evaluation **[3.51]** showed that the channels are not overstressed and are able to withstand the RLE loading.
2. The C6 channels from item 1 above are anchored to the concrete pads with wedge anchors. However, shim packs as high as 1 1/2 inches are underneath the channels. Such a configuration would tend to cause bending stresses in the anchors in addition to the normal tension and shear stresses. An evaluation **[3.51]** showed that the anchors are not overstressed as a result of the RLE loading and are adequate to support the air handler units.
3. On air handler T4100B018, the top of some wedge anchors were obscured by caulking material. Work Request 000Z951277 was prepared to remove the caulking material so that the anchors could be observed. The wedge anchors were found to be properly installed.

3.1.4.9.2.11 Chillers

Equipment Category: 11

Number of items on SSEL: 2

Location	PIS No.	
AB5	T4100B008	T4100B009

Equipment Description

These chiller units condense and evaporate refrigerants in order to provide cold air for the Control Center HVAC air handling units. The 100-ton capacity units are manufactured by

Trane. The skid-mounted units include a cylindrical condenser and evaporator stacked on top of each other, a compressor, a 130-HP compressor motor, valve actuators, a purge compressor and motor, and purge and oil heaters. The components are bolted to a steel skid which, in turn, is anchored to a concrete pad. Attachments to the chiller include piping for routing the refrigerant and chilled water, electrical conduits, and instrumentation and control lines. The units are supported vertically on vibration isolators, which eliminate vibrations from being transferred to the floor slab, and horizontally by snubbers.

Equipment Evaluation

The SRT walked down the two chillers and related equipment in this category. The screening criteria of EPRI NP-6041 was used to evaluate the units. Based on the walkdown and review of chiller mounting drawings, the SRT concluded the following:

1. The motor and compressor are attached to concrete pads which are doweled into the common concrete floor slab.
2. Lateral load is adequately resisted by snubbers which are mounted to the concrete pads and are connected to the metal frame.
3. The units are mounted on vibration isolators; however, lateral load is resisted by the snubbers, not the isolators.
4. There is no concern for excessive nozzle loading resulting from gross pipe motion or differential displacement and prying about a rigid pipe support.
5. There are no adjacent, massive, unsupported or lightly supported in-line components or long unsupported pipe spans.
6. For the horizontal condenser mounted above the evaporator, the saddle supports are adequate for horizontal loads. Saddles are welded all around to the evaporator and condenser and are stiffened in the direction of weak axis bending.
7. Pipe connections to the evaporator and condenser are welded, not threaded.

An evaluation **[3.51]** was performed for numerous equipment attached to and supplied with the CCHVAC chiller skids to demonstrate that they can be qualified for the RLE. The evaluation included a comparison of the fragility test response spectra to the RLE required response spectra. The comparison showed that the TRS fully envelope the RLE at all frequency ranges. Therefore, the CCHVAC components are qualified for the RLE seismic loads.

Anchorage Evaluation

The chillers and related equipment are anchored to the concrete pads with embedded anchors. Anchor spacing meets minimum requirements. The anchorage conforms to the design drawings. The nuts are present and apparently tight. There are no apparent gaps under the equipment bases. The anchorage appears to be relatively stiff. Anchorage Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis anchorage evaluation. The

Equipment Evaluation

The SRT walked down all of the air compressors in this category. The screening criteria of EPRI NP-6041 was used to evaluate the units. Based on the walkdown and review of air compressor mounting drawings, the SRT concluded the following:

1. The motor and compressor are rigidly connected to a common concrete mounting pad.
2. There is no concern for excessive nozzle loading resulting from gross pipe motion or differential displacement and prying about a rigid pipe support.
3. There are no adjacent, massive, unsupported or lightly supported in-line components or long unsupported pipe spans.
4. The units are not mounted on vibration isolators.

Anchorage Evaluation

Each air compressor is securely anchored with embedded (cast-in-place) bolts. There is sufficient anchor edge distance on pads. Pads are doweled into the concrete floor slabs. Anchor spacing meets minimum requirements. The anchorage conforms to the design drawings. The nuts are present and apparently tight. The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concludes that the anchorage is able to withstand the RLE loads.

System Interaction Effects

The general areas around the air compressors were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns. Attached lines are adequately flexible to accommodate relative movement. Soft targets are free from impact from nearby equipment or structures. The air compressors are sufficiently far from adjacent components and structures to preclude any interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment.

Outliers and Outlier Resolution

There were no concerns associated with the equipment in this category.

3.1.4.9.2.13 Motor Generators

Equipment Category: 13

Number of items on SSEL: None

calculation concludes that the anchorage has adequate capacity to withstand the RLE loads.

System Interaction Effects

The general areas around the chillers were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns. Attached lines are adequately flexible to accommodate relative movement. The chillers are typically located away from adjacent components and structures to preclude any interaction. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment.

Outlier and Outlier Resolution

The following concern was identified during the plant walkdown. This concern has been resolved as indicated.

The T41 current transformers (CTs) are located inside chiller motor junction boxes. The components were not observed by the SRT because opening one of the boxes would render that Control Center HVAC division inoperable. Four mounting bolts (apparently for the CTs) were observed underneath and extending through each box. One of the nuts was not tightened on one of the bolts. A work request (WR No. 000Z957682) was prepared to tighten the nut.

3.1.4.9.2.12 Air Compressors

Equipment Category: 12

Number of items on SSEL: 6

Location	PIS No.			
ABB	P5002D001	P5002D002		
RHR1	R3000D001	R3000D002	R3000D003	R3000D004

Equipment Description

The function of each air compressor is to maintain the proper air pressure in the air accumulator tanks for their respective systems (NIAS and EDG Air Start). The NIAS air compressor is a single cylinder, single stage, reciprocating piston type manufactured by Joy, Model WGOL-9. It is powered by a 30-HP motor. The EDG air compressor is the same type, manufactured by Quincy, Model 325-103, and powered by a 5-HP motor. The compressors are mounted on concrete pads and are anchored to the concrete with embedded anchors.

3.1.4.9.2.14 Distribution Panels

Equipment Category: 14

Number of items on SSEL: 20

Location	PIS No.				
RBB	H21P561	H21P562			
RB1	H21P560				
RB2	H21P559				
AB2	H11P900	H11P901	H11P902	H11P903	R3200S061A
	R3200S061B	R3200S062	R3200S064A	R3200S064B	
AB3	R3200S026	R3200S027	R3200S065		
AB5	H21P557	H21P558			
RHR2	R3200S063	R3200S066			

Equipment Description

The distribution panels in this category are the panelboard type, which include buses, switches, and automatic protective devices. Their primary function is to distribute low voltage electrical power from a main circuit to a branch circuit and to provide overcurrent protection. The panels, Model QMB manufactured by Square D, are wall-mounted, steel cabinets accessible only from the front. The overall dimensions and weights of three typical panel sizes are: 2'-7" wide by 4'-8" high by 11" deep; 430 pounds; 2'-7" by 3'-8" by 1'-3"; 385 pounds; and 3'-2" by 5'-8" by 1'-7", 850 pounds.

Equipment Evaluation

The SRT walked down all of the distribution panels in this category. The screening criteria of EPRI NP-6041 was used to evaluate the units. Based on the walkdown and a review of distribution panel documentation, the SRT concluded the following:

1. Internal device mountings, when visible, were not excessively flexible, appeared seismically rugged, and were properly attached to the cabinet. However, most of the internal equipment was not visible because it was behind an internal wall or panel which prevented observation of the equipment.
2. The panels were in accordance with NEMA Type 4 specifications.
3. The doors were secured with latches or fasteners.
4. The panels were mounted on concrete walls.

Anchorage Evaluation

The panels are mounted to walls by one of two methods: (1) attached directly to the wall with wedge or self-drilling anchors, or (2) bolted to Unistrut members which, in turn, are wedge anchored to the wall. Anchor spacing meets minimum requirements. The anchorage conforms to the design drawings. The nuts are present and apparently tight. The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concludes that the anchorage has adequate capacity to withstand the RLE loads. (See Outlier section below.)

System Interaction Effects

The general areas around the distribution panels were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns. Attached lines are adequately flexible to accommodate relative movement. The distribution panels are sufficiently far from adjacent components and structures to preclude any interaction. There is no potential collapse of adjacent structures or equipment. No potential sources of flooding were discovered that could spray or cascade onto sensitive equipment.

Outliers and Outlier Resolution

The following concerns were identified during field walkdowns and review of DC-5634. Each outlier has been resolved as indicated.

1. The pre-screening anchorage evaluation for panel R3200S062 assumed that the anchorage consisted of six 1/2" diameter Phillips wedge anchors. However, based on field observations six 3/8" diameter Phillips self-drilling anchors were used. Further evaluation **[3.51]** by the SRT showed that the self-drilling anchors have adequate capacity for the RLE.
2. The attachment brackets for some of the panels are Z-shaped instead of angle shaped as shown on the design drawings. Technical Service Request (TSR)-27874 was initiated to revise the appropriate drawings to reflect the as-built condition. The Z-brackets were judged adequate by the SRT.
3. The nut on one wedge anchor was missing for panel H21P561. A work request (WR No. 000Z955432) was prepared to replace the nut.

3.1.4.9.2.15 Batteries and Racks

Equipment Category: 15

Number of items on SSEL: 2

Location	PIS No.	
AB3	R3200S003	R3200S004

Equipment Description (See Figure 3-25)

The batteries in this category are the lead-calcium type, Model KC-17 manufactured by C & D Battery Company. Each 250 VDC battery consists of 120 individual nominal two-volt cells arranged in series and connected with bolted, lead-plated copper connectors. The individual cells, weighing about 140 pounds each, are arranged on multi-rowed racks, all at the same level. There are two battery rooms (one per division), two parallel rack assemblies per division, four individual racks per assembly, and 15 batteries per rack. Each rack assembly consists of four racks arranged in a two by two matrix configuration. Each rack is about 11'-6" long, 1'-6" wide, and 1'-10" high. The steel racks are made from welded angle and channel sections, with diagonal bracing bars. The batteries are supported at mid-height by bars running the length of the racks to prevent overturning.

Equipment Evaluation

The SRT walked down all of the batteries and racks in this category. The screening criteria of EPRI NP-6041 was used to evaluate the units. Based on the walkdown and a review of battery rack mounting drawings, the SRT concluded the following:

1. Battery plates are of the lead-calcium type.
2. Batteries are restrained in all horizontal directions by side rails and shims at the ends of the racks.
3. Spacers between the batteries are made from soft, open-cell foam and fill most of the space between the batteries. This type of foam spacer is listed as a caveat in EPRI NP-6041 and is discussed further in the Outlier section below.
4. Side rails are adequately strong and stiff.
5. The lateral load resistance system of the racks is adequate to support the batteries during a RLE.
6. There is adequate space between adjacent racks to preclude impact.

Anchorage Evaluation

The battery racks are attached to the floor slab with threaded studs which anchor a small sole plate at the end of each leg to an embedded Gateway insert. Each rack has ten such attachment points. Each sole plate rests atop about a one-inch thick grout pad at each leg location. The anchorage conforms to the design drawings. The nuts are present and appear tight. The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concludes that the anchorage has adequate capacity to withstand the RLE loads.

System Interaction Effects

The general areas around the batteries and racks were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns. Attached cables and conduits have adequate flexibility to accommodate relative movement. The batteries and racks are sufficiently far from adjacent components and structures to preclude any interaction. There is no potential for collapse of adjacent structures or equipment. No potential sources of flooding were discovered that could spray or cascade onto the batteries.

Outlier and Outlier Resolution

The use of soft, open-cell foam spacers between the individual batteries is listed as a caveat in EPRI NP-6041 based on the concern that the batteries may impact each other during a seismic event if the spacers do not provide adequate protection. Based on the satisfactory use of the same type foam spacers in vendor seismic testing and the fact that the test response spectra envelope the Fermi 2 RLE required response spectra at pertinent frequencies, the SRT concluded that the spacers would not adversely affect the batteries' function and that the batteries could withstand RLE loads **[3.51]**.

3.1.4.9.2.16 Battery Chargers

Equipment Category: 16

Number of items on SSEL: 6

Location	PIS No.				
AB3	R3200S020A	R3200S020B	R3200S020C	R3200S021A	R3200S021B
	R3200S021C				

Equipment Description

The battery chargers in this category are manufactured by C & D Battery company and consist of a floor-mounted cabinet made from sheet metal over a steel angle frame. The cabinets contain an instrument panel with push-button switch assembly, voltmeter, circuit breaker, timer, and ammeter; as well as relays, transformers, rectifiers, resistors, fuses, timer, and bus bar. The purpose of the battery charger is to convert AC power to DC power to keep the batteries in the fully charged condition. Each cabinet measures 2'-4" wide, 4'-10" high, and 1'-8" deep and weighs about 600 pounds.

Equipment Evaluation

The SRT walked down all of the battery chargers in this category. The screening criteria of EPRI NP-6041 was used to evaluate the units. Based on the walkdown and a review of battery charger mounting drawings, the SRT concluded the following:

1. The battery chargers are solid state devices.
2. The transformers are near the base and securely attached to the cabinet.
3. The load path is adequate to transmit loads from the cabinet to the foundation.
4. The base mounting channels are adequate to resist lateral forces.
5. The doors are secured by latches or fasteners.

Anchorage Evaluation

The battery chargers are attached to the floor slab with a combination of anchor bolts, attached to embedded Gateway inserts, and concrete wedge anchors. Each charger has four such attachment points, one near each corner. There are no edge distance or spacing violations for the anchors. The anchorage conforms to the design drawings. The nuts are present and appear tight. There are some gaps under the bolted base channel; however, all gaps are less than 1/4". The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concludes that the anchorage has adequate capacity to withstand the RLE loads.

System Interaction Effects

The general areas around the battery chargers were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns, except as noted in the outlier section below. Attached cables and conduits have adequate flexibility to accommodate relative movement. The battery chargers are sufficiently far from adjacent components and structures to preclude any interaction. There is no potential for collapse of adjacent structures or equipment. No potential sources of flooding were discovered that could spray or cascade onto the battery chargers.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each concern has been resolved as indicated.

1. There is a gap of approximately 1 3/4" between a card reader and unit R3200C021C. The card reader is very flexible but is light in weight. The SRT judged that the potential interaction between the two components is not a concern because of the relatively large gap and the light weight of the card reader **[3.51]**.
2. Small gaps exist under portions of mounting channels for battery chargers R3200S021A and R3200S021B. The gaps extend from near the anchor to the edge of the channel. The SRT concluded that the small gaps would not have a significant effect on the load transfer path into the foundation **[3.51]**.

3.1.4.9.2.17 Engine Generators

Equipment Category: 17

Number of items on SSEL: 4

Location	PIS No.			
RHR1	R3001S001	R3001S002	R3001S003	R3001S004

Equipment Description

There are four emergency diesel generators at Fermi 2. The EDGs are housed in the reinforced concrete RHR building to afford protection from floods, tornado winds, and tornado-generated missiles. Each unit is housed in a separate room for fire protection purposes. The EDGs provide emergency AC power in the event of loss of off-site power. The EDGs are Fairbanks-Morse turbo-charged diesel generator units, two per division. Each engine has 12 cylinders, 24 opposed pistons, dual crankshafts, and is rated 3967 HP at 900 RPM. Each EDG is provided with a 4160 VAC, three-phase, 60 Hz generator. The generator's rotor is driven by the engine. The engine and generator are mounted to a common skid.

Each EDG is provided with the following subsystems to provide independence from the other units: starting air, fuel oil, governor, lube oil, air intake and exhaust, air cooler coolant, jacket coolant, and diesel generator service water.

Numerous peripheral equipment, as follows, was also evaluated during the EDG walkdowns: temperature control valves R3000F012A to D and R3000F123A to D; float lube oil regulators R3000F109A to D; signal generators R30NA17A to D; lube oil crankcase pressure switches R30NA08A to D, R30NA09A to D, R30NA10A to D, R30NA11A to D, R30NA12A to D, and R30NA13A to D.

Equipment Evaluation

The SRT walked down all of the EDGs in this category. The screening criteria of EPRI NP-6041 was used to evaluate the units. Based on the walkdown and a review of EDG mounting drawings, the SRT concluded the following:

1. The engine and generator are mounted to a common stiff skid.
2. The lateral load resistance system is adequate.
3. There is little or no potential for relative motion between the EDGs and non-flexible interconnecting fuel, lube oil, and water cooling lines.
4. Appurtenances are supported with stiff members.
5. There are no weak seismic links for attachments to the EDGs or skids.

6. The engine gauge panels are mounted at the side of the EDG skids on vibration isolators. See Item 12 in Section 3.1.4.9.2.21 for a discussion of a panel anchorage concern.

Anchorage Evaluation

Each EDG is attached to a steel skid which, in turn, is secured to the concrete slab with 16 embedded anchor bolts. There are no edge distance or spacing violations for the anchors. The anchorage conforms to the design drawings. The nuts are present and appear tight. The anchorage appears to be relatively stiff without excessive prying action. The SRT evaluated the anchor bolts for the RLE. Review of the design basis qualification report for the bolts showed that a large margin exists between the design and allowable stresses. Based on the large margin, the SRT concluded that the anchor bolts have adequate capacity to withstand the RLE.

System Interaction Effects

The general areas around the EDGs were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns. Attached tubing and conduits have adequate flexibility to accommodate relative movement. Soft targets on the EDGs are free from impact from nearby equipment or structures. The EDGs are sufficiently far from adjacent components and structures to preclude any interaction. There is no potential for collapse of adjacent structures or equipment. No potential sources of flooding were discovered that could spray or cascade onto the EDGs.

Outliers and Outlier Resolution

There are no concerns associated with the equipment in this category. Concerns associated with appurtenances mounted on the diesel generator skids are described in their pertinent sections.

3.1.4.9.2.18 Automatic Transfer Switches

Equipment Category: 18

Number of items on SSEL: None

3.1.4.9.2.19 Instrument Racks

Equipment Category: 19

Number of items on SSEL: 50

Location	PIS No.				
DW1	B21P402E	B21P402H	B21P402J	B21P402P	B21P402R

Location	PIS No.				
RBSB	H21P017	H21P485	H21P614A	H21P614B	
RBB	H21P006	H21P016	H21P018	H21P021	H21P022
	H21P034	H21P036	H21P037	H21P038	H21P548
RB1	C11P401	H21P009	H21P010	H21P035	H21P423A
	H21P423B	H21P474	H21P475	H21P631A	H21P631B
	T49P400A				
RB2	H21P004	H21P005	H21P595A	H21P595B	T49P400B
ABSB	H21P014	H21P420	H21P428		
ABB	H21P501A	H21P501B	P50P401A	P50P401B	
AB5	H21P572	H21P573	LM-24*	LM-25*	
RHR1	R30P405A	R30P405B	R30P405C	R30P405D	

* These tube stands support instruments T41N059A and B and do not have a unique PIS number.

Equipment Description

The instrument racks in this category typically consist of steel frames which provide mounting structures for local controls and instrumentation. The instrument racks consist of steel members, usually angles, channels, pipe, or Unistrut, bolted or welded together to form a frame. The floor-mounted instrument racks range in height from about four feet for a pipe or tube stand to eight feet for a regular frame rack. Width varies from three to ten feet depending on the number of instruments mounted on the rack. Some frame racks are supported in the front-to-back direction by inclined structural members. Wall-mounted instrument racks are smaller than the floor-mounted racks and generally consist of a steel plate mounted to embedded Unistrut members. Most of the racks at Fermi 2 are manufactured or supplied by GE and York, or fabricated by Detroit Edison. Instruments are either attached directly to the frame or to mounting plates which, in turn, are attached to the frame. Typical instruments mounted to instrument racks include pressure switches, transmitters, gauges, recorders, switches, manifold valves, and solenoid valves. Other attachments include tubing, conduits, wiring, and junction boxes.

Equipment Evaluation

The SRT walked down all of the instrument racks in this category. The screening criteria of EPRI NP-6041 was used to evaluate the units. Based on the walkdown and review of instrument rack mounting drawings, the SRT concluded the following:

1. The instruments and mounting plates attached to the racks are not excessively flexible and appear seismically rugged and properly attached to the rack.
2. The instrument racks are able to adequately resist lateral loads. The frame members are properly connected to each other and the rack is properly connected to the floor or wall.

Anchorage Evaluation

The instrument racks are attached to the floor slabs and walls with a combination of concrete expansion anchors and/or welds to embedded plates. There are no edge distance or spacing violations for the anchors. The anchorage conforms to the design drawings. The nuts are present and appear tight. Some gaps were noted under the bolted base channels but in all cases are less than 1/4". The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concludes that the anchorage has adequate capacity to withstand the RLE loads.

System Interaction Effects

The general areas around the instrument racks were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns, except as described in the outlier section below. Attached tubing and conduits have adequate flexibility to accommodate relative movement. Soft targets on the instrument racks are free from impact from nearby equipment or structures. The instrument racks are sufficiently far from adjacent components and structures to preclude any interaction. There is no potential for collapse of adjacent structures or equipment. No potential sources of flooding were discovered that could spray or cascade onto the instrument racks.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each concern has been resolved as indicated.

1. For drywell instrument racks B21P402E, H, J, P, and R, certain welded connections could not be visually verified during the walkdown due to physical obstructions. The SRT performed a tug test on the racks and they appeared very sturdy. Based on the field walkdown and the analytical evaluation **[3.51]**, the SRT concluded that the instrument rack connections have adequate capacity to withstand the RLE.
2. Some mounting hardware on rack H21P017 was either missing or loose. A work request (WR No. 000Z951276) was prepared and the hardware was tightened or replaced. Work was completed in June 1995.
3. Some mounting hardware on rack H21P021 was either missing or loose. A work request (WR No. 000Z948669) was prepared and the hardware was tightened or replaced. Work was completed in December 1994.
4. Wedge anchor spacing violations were noted between anchors for racks H21P021 and H21P036. The violations were addressed in DC-5634 and found acceptable.
5. For racks H21P034 and H21P038, the mounting anchors are 3/8". However, the mounting was qualified for the RLE in DC-5634 assuming 1/2" anchors. An evaluation **[3.51]** was performed by the SRT based on a comparison of these racks

with similar racks. The SRT concluded that the anchors have adequate capacity to withstand the RLE. A drawing update has also been initiated to reflect the as-built conditions.

6. Fluorescent light fixtures above racks H21P474 and T49P400A were not supported with redundant safety cables or chains, which are required by specification. The cables prevent the fixtures from falling if the main support should fail during a seismic event. A work request (WR No. 000Z947544) was prepared and redundant cables were installed.
7. A 1/4-inch rattlespace exists between rack H21P501B and a conduit support Unistrut member in the rack side-to-side direction. No sensitive relays are mounted on the rack. Based on the rack rigidity and the low seismic acceleration values which the rack may experience due to its basement level location, the SRT concluded that the existing rattlespace between the two components was acceptable **[3.51]**.
8. A 1/8-inch vertical gap exists between a tube above rack H21P501B and a pipe support member. Based on the low seismic accelerations in the auxiliary building basement and the rigidity of the support, the SRT concluded that the existing rattlespace between the two components was acceptable **[3.51]**.

3.1.4.9.2.20 Local Instruments/Temperature Sensors

Equipment Category: 20

Number of items on SSEL: 38

Location	PIS No.				
RBSB	E41N062B	E41N062D	E51N023A	E51N023B	T23N010A
	T23N010B				
RB2	P44N401A	P4N401B			
ABSB	E41K400	E41N030A	E41N030B	E41N203	
AB2	T41N061A	T41N061B	T41N062A	T41N062B	
AB3	T41N063A	T41N063B	T41N065A	T41N065B	T41N066A
	T41N066B				
AB4	T41N067A	T41N067B			
AB5	T41N068A	T41N068B	T41N117A	T41N117B	
RHR1	X41N058A	X41N058B			
RHR2	X41N056A	X41N056B	X41N056C	X41N056D	X41N057A
	X41N057B	X41N057C	X41N057D		

<u>Equipment Description</u>

The equipment in this category includes both thermocouples and resistance temperature detectors (RTDs) that measure fluid and air temperatures. They are mounted within piping by means of a thermowell, which is a permanently mounted tube within the pipe, as well as on concrete surfaces by means of mounting brackets. The thermocouple probes consist of two dissimilar metal wires in a protective sleeve which produce a voltage output proportional to the temperature difference. The RTD operation is based on variation in electrical resistance with temperature. The Fermi 2 temperature sensors are manufactured by Pyco, United Electric Controls, and Powers Regulator Company. The T41 sensors are wall-mounted thermostats. The X41 temperature detectors are welded to Unistrut members which are anchored to concrete walls.

The local instruments include indicator switches by Barton, temperature transmitters by Powers Regulator, level transmitters by Gould Pumps, and pneumatic pressure controllers by Fisher Controls. They are mounted to Unistrut members which are anchored to concrete walls.

Equipment Evaluation

The SRT walked down all of the local instruments/temperature sensors in this category. The screening criteria of EPRI NP-6041 was used to evaluate the units. Based on the walkdown and review of instrument rack mounting details, the SRT concluded the following:

1. The sensors are rigidly and securely mounted either in the "hotwells", with mounting brackets, or with concrete anchors.
2. There is no concern for excessive differential displacement between connection head and temperature sensor mountings.
3. The local instruments are securely mounted to Unistrut members which are, in turn, mounted to concrete walls with concrete anchors.

Anchorage Evaluation

The local instruments and air temperature thermocouples are attached to concrete surfaces with expansion anchors. There are no edge distance or spacing violations for the anchors. The nuts are present and appear tight. The anchorage appears to be relatively stiff. In many cases, the component was tug tested. The SRT concluded that the local instrument and temperature sensor mounting and anchorage have adequate capacity to withstand the RLE.

System Interaction Effects

The general areas around the local instruments and temperature sensors were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns. Attached wire and cabling have adequate flexibility to accommodate relative movement. Soft targets on the sensors are free from impact from nearby equipment or structures. The instruments and sensors are sufficiently far from adjacent components and structures to preclude any interaction except as noted in the outliers section. There is no potential for collapse of adjacent structures or equipment. No potential sources of flooding were discovered that could spray or cascade onto the instruments or sensors.

Outliers and Outlier Resolution

Only one concern was identified during plant walkdowns. Level transmitter E41N062D is about 1/16-inch from a wall-mounted Unistrut member. The SRT judged that there is not an interaction concern because of the secure and stiff mounting arrangement of the transmitter and mounting bracket.

3.1.4.9.2.21 Control and Instrumentation Cabinets

Equipment Category: 21

Number of items on SSEL: 83

Location	PIS No.				
RB1	G51P400A	G51P400B			
ABB	P50P402A	P50P402B			
AB2	H11P609	H11P611	H11P612	H11P613	H11P614
	H11P617	H11P618	H11P620	H11P621	H11P622
	H11P623	H11P626	H11P627	H11P628	H11P857
	H11P870	H11P898A	H11P898B	H11P914	H11P15
	H11P917A	H11P917B	H11P923	H11P929	H21P100
	R3101S001				
AB3	B21P400	B21P401	H11P601	H11P602	H11P809
	H11P810	H21P090-1	H21P090-2	R3101S002	
AB4	H21P080	H21P081	H21P082	H21P083	H21P084
	H21P085	H21P086	H21P087	H21P528	H21P529
AB5	H21P285A	H21P285B	H21P296A	H21P296B	H21P296C
	H21P296D	H21P296E	H21P296F	H21P527	H21P527A
RHR1	E11P400A	E11P400B	H21P517	H21P518	R30P310
	R30P320	R30P330	R30P340		
RHR2	H21P350	H21P351	H21P352	H21P353	R3000S005
	R3000S006	R3000S007	R3000S008	R3000S009A	R3000S009B
	R3000S009C	R3000S009D	R30P311	R30P321	R30P331
	R30P341				

Equipment Description

The panels in this category include the Control Room benchboard panels, the Relay Room vertical switchboard panels, and numerous local panels throughout the plant. This equipment class includes all types of electrical panels which support instrumentation and controls. The panels have a wide diversity of sizes, types, functions, and components. The instruments are mounted both on and inside the enclosures. The panels are both wall mounted and floor mounted. A cabinet generally consists of a steel frame of angles, channels, or tube steel welded together supporting sheet metal panels attached by welding. Large panels are made of individual sections bolted together through adjoining framing; access is through doors on one or both side panels. The Fermi 2 panels are manufactured or supplied by the following companies: Electro-Mechanics, GE, Automatic Industries, Citation Tool, Foxboro, Hoffman, Powers Regulator, Colt, Beloit, and Reliance Electric.

The chiller local control panels are also included in this category. The panels, manufactured by Hoffman and supplied by Trane, are 12-gage enclosures with internal stiffeners. They measure about 4'-0" wide, 2'-0" deep, and 6'-0" tall and weigh about 500 pounds.

Equipment Evaluation

The SRT walked down all of the cabinets in this category and examined most of the cabinets internally. A few locked panels were not examined internally by the SRT. The screening criteria of EPRI NP-6041 was used to evaluate the units. Based on the walkdown and a review of cabinet mounting details, the SRT concluded the following:

1. The panel mounting tabs and rolled flanges are not excessively flexible.
2. The internal device mountings, with the exceptions listed in the outlier section below, are not excessively flexible and appear seismically rugged and properly attached to the cabinet.
3. The seismic load path appears adequate to transmit the loads from the panel to the foundation.
4. There are no excessively large cutouts in the lower half of the cabinets.
5. The cabinets do not appear excessively flexible.
6. The panels and cabinets are in accordance with NEMA Type 12 specifications.
7. The doors are secured with latches and/or fasteners.

Anchorage Evaluation

The cabinets are attached to the floor slabs or walls with either concrete expansion anchors or plug welds to embedded plates. There are no edge distance or spacing violations for the anchors. The anchorage conforms to the design drawings. The nuts are present and appear tight. There does not appear to be any excessive flexibility between the tiedown anchorage and the cabinet walls. The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concludes that the anchorage has adequate capacity to withstand the RLE loads. Anchorage for panels not included in DC-5634 are discussed in the outlier section below.

System Interaction Effects

The general areas around the panels were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns, except as described in the outlier section below. Attached tubing and conduits have adequate flexibility to accommodate relative movement. Soft targets on the cabinets are free from impact from nearby equipment or structures. The cabinets are sufficiently far from adjacent components and structures to preclude any interaction. There is no potential for collapse of adjacent structures or equipment. No potential sources of flooding were discovered that could spray or cascade onto the cabinets.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each concern has been resolved as indicated.

1. Numerous panels with sensitive relays in the relay room, switchgear rooms, and RHR building are located adjacent to other panels but not bolted together. EDP-27108 was prepared to fasten these panels together to prevent interaction. Work is scheduled to be completed by the end of the upcoming RFO5 refueling outage.
2. At the base of numerous relay room panels (PIS Nos. H11P612 to H11P614, H11P617, H11P618, H11P620, H11P623, H11P626 to H11P628), fillet welds connecting the panels to the embedded leveling channels are on the sheet metal instead of the base angle as shown on design drawings. However, based on the sheet metal thickness, weld size, and small gap under the panels, the SRT judged that the existing fillet welds have adequate capacity to withstand the RLE **[3.51]**. Technical Service Request (TSR)-27548 was initiated to revise the drawings to conform to the as-built configuration.
3. Some GEMAC modules are not securely installed in panels H11P612 and H11P613. Work requests (WR Nos. 000Z951324 and 000Z952656, respectively) were initiated to secure the modules. This work is scheduled for completion during the upcoming RFO5 refueling outage.
4. Some screws were missing from a large insert in panel H11P614. A work request (WR No. 000Z951311) was initiated to replace the missing screws.
5. For panels H11P857 and H11P870, the design drawing base detail shows double angles at the bottom. The as-built condition has one angle and rolled out sheet metal. There is no flexibility concern due to the thickness of the sheet metal. The drawing will be revised as part of TSR-27548.
6. A 1 3/8-inch gap exists between the top of panel H11P857 and the surface of a fire-wrapped cable tray hanger vertical tube steel member. Based on an evaluation **[3.51]** by the SRT, the existing rattlespace is sufficient to preclude interaction between the two components during a RLE.
7. The anchorage for panels H11P609 and H11P611 could not be verified due to fire wrap at the base of the panels. However, the SRT judged that the anchorage was acceptable based on visual confirmation of similar details in other panels.
8. Some recorders on panel H11P602 had bent hinges on the doors; however, the SRT concluded that the doors would not break off and form missiles during the RLE because of the attachment of the hinge at more than one place.

9. Fluorescent light fixtures above numerous racks on AB2, AB3, and AB5 were not supported with redundant safety cables or chains, which are required by specification. The cables prevent the fixtures from falling if the main support should fail during a seismic event. Work requests (WR Nos. 000Z951312, 000Z951325, and 000Z953608) were initiated to install redundant cables. The work has not currently been scheduled.
10. A mounting screw for a Dwyer pressure switch in panel H21P296B was discovered missing. A work request (WR No. 000Z953697) was initiated to replace the missing screw. Work was completed in July 1995.
11. The SRT had an anchorage strength and stiffness concern regarding the EDG engine gauge panels, R30P310, R30P320, R30P330, and R30P340. Each panel has four brackets which are welded to the panel and bolted to vibration isolators. The brackets have 1 1/2" long, front-to-back slots which may allow the panel to slide during a seismic event. Such motion could affect sensitive components inside the panel. Based on discussions with the equipment vendor, it was decided to torque a jam nut against the attaching nut, thus producing a friction connection to prevent sliding within the slot. This fix also resolved the an as-found condition on panel R30P320 where a jam nut was not tightened against the bracket. Work requests (WR Nos. 000Z953622 through 000Z953625) were initiated to torque the jam nuts on the respective panels. Work was completed in August 1995. The pertinent vendor manual was revised to document the required jam nut torque value.
12. A small gap, varying from 1/8-inch near the base to 1-inch at the top, exists between panel H21P350 and the adjacent concrete wall. An evaluation **[3.51]** was performed to show that the existing rattlespace is adequate to preclude interaction between the two components during the RLE.
13. In panels H21P350 and H21P351, some relays had missing mounting screws. Work requests (WR Nos. 000Z955153 and 000Z955154) were initiated to replace the missing screws.
14. A separate anchorage evaluation **[3.51]** was performed for panel H21P100 which was not included in DC-5634. The panel is welded to embedded plates. The evaluation showed that the panel connections have adequate capacity to withstand the RLE loads.
15. A separate anchorage evaluation **[3.51]** was performed for panels H21P285A and B which were not included in DC-5634. The panels are anchored to the concrete floor with self-drilling anchor bolts. The evaluation, which included reductions for spacing violations, showed that the panel connections have adequate capacity to withstand the RLE loads.
16. During the walkdown of temperature controllers mounted in panels H21P350 through H21P353, it was discovered that one controller in each panel is mounted differently than documented in the original seismic qualification report. The controllers (PIS Nos. X4103K002E, F, G, and H) are bolted to small electrical boxes which are, in turn, bolted to an internal sub-panel. The mounting detail is a standard detail provided by the manufacturer but is not the detail tested. An evaluation **[3.51]** was performed which showed that the mounting bolts and overall arrangement are adequate to withstand the RLE loads.

17. The vibration isolators for the EDG engine gauge panels, PIS numbers R30P310, R30P320, R30P330, and R30P340, are made from cast iron. An evaluation **[3.51]** was performed to show that the isolators would not fail if the panels were to bottom out during the RLE, since the cast iron elements will primarily be subject to compression load.
18. Panels H21P296A and B are installed with one sided bevel welds between the panel base and inverted channels anchored into the floor. Since the welds are six inches from the outer edge of the panel, this detail provides a weak load path for lateral load resistance. Deviation Event Report (DER) 96-0289 **[3.81]** was initiated to address this issue and follow up with any necessary improvements.

3.1.4.9.2.22 Other Categories

Equipment Category: 22

The equipment in this category is comprised of those components on the SSEL that do not fit directly into the other 21 categories. The 21 categories correspond to the classes of safe shutdown equipment identified in the SQUG program and EPRI NP-6041. Some of the other equipment, such as tanks, heat exchangers, and valves in the hydraulic control units (HCUs), were evaluated using SEWS forms which are customized for such equipment and available in EPRI NP-6041, Appendix F. Other equipment, such as dryers, barometric condensers, temperature control valves and steam-driven turbines, were evaluated using SEWS forms for similar equipment.

<u>Equipment Description</u>

A. Other Valves – 1119 total

The valves in this sub-category are different from the fluid-operated and motor-operated valves in Sections 3.1.4.9.2.7 and 3.1.4.9.2.8, respectively.

<u>Location</u>	<u>PIS No.</u>			
RBSB	E5150F044			
RB1	C1103D001 to C1103D185 (Subcomponents 4 to 9, 1110 total)			
ABSB	E4100F067	E4100F068		
ABB	P5002D029A	P5002D029B		
RHR1	R3000F023A	R3000F023B	R3000F023C	R3000F023D

The E4100F067 HPCI stop valve is a ten-inch, hydraulic oil-operated valve manufactured by Schutte & Koerting. It allows the flow of steam to the turbine. The E4100F068 HPCI and E5150F044 RCIC control valves are ten- and three-inch hydraulic oil-operated valves manufactured by Terry Steam Turbine. They act as throttle valves to control the flow of steam to their respective turbines. The P50 drain traps, manufactured by Hankison Corporation, collect and discharge condensate from the NIAS air dryers. The R30 EDG temperature control valves are four-inch, three-way diaphragm control valves manufactured by Robertshaw Controls Company. The valves control the flow of the air coolant system water through or around the heat exchanger. The C11 valves consist of six valves on each HCU (inlet and outlet scram valves by Hammel-Dahl and four directional control solenoid valves by ASCO). They act to control the insertion and withdrawal of the control rod drives.

B. Tanks – 413 total

Location	PIS No.				
DW2	B2104A003A	B2104A003B	B2104A003C	B2104A003D	B2104A003E
RBSB	E5100B001				
RB1	C1103D001 through C1103D185 (Subcomponents 1 and 3, 370 total)				
	P5002A004A	P5002A004B			
RB2	P4400A001	P4400A002			
ABSB	E4100B001				
ABB	P5002A001	P5002A002	P5002D012	P5002D013	P5002D014
	P5002D015	P5002D016	P5002D017		
RHR1	R3000A001	R3000A002	R3000A003	R3000A004	R3000A005
	R3000A006	R3000A007	R3000A008	R3000A009	R3000A010
	R3000A011	R3000A012	R3000A013	R3000A014	R3000A015
	R3000A016	R3000A017	R3000A018	R3000A019	R3000A020
	R3000A021	R3000A022	R3000A023	R3000A024	

The B21 SRV accumulator tanks are approximately 5'-0" long and 2'-6" in diameter, manufactured by Richmond Engineering Company. Their function is to provide reserve air for the Automatic Depressurization System (ADS) activation of the SRVs. The E41 and E51 barometric condensers, manufactured by Nash Engineering Company, are essentially steel tubes approximately six inches in diameter and 72 inches long. Their function is to condense steam leakage from the turbine and control valves during HPCI and RCIC turbine operation. The barometric condenser also includes a vacuum tank into which the condensate drains. Each C11 hydraulic control unit, manufactured by GE, includes two accumulator tanks, one containing water and the other containing nitrogen. The function of the accumulator assembly is to provide a local source of kinetic energy to insert the CRD in case of low reactor pressure or CRD hydraulic system failure. The P50 air accumulators for the railroad access doors are fabricated by Detroit Edison and are approximately 5'-3" long and 10" in diameter. They provide a source of stored, pressurized air to inflate the door seals to assure secondary containment in the case of station compressed air failure. The P44 EECW make-up tanks, manufactured by National Annealing Box, are horizontal tanks measuring approximately 8'-0" long and 4'-0" in diameter. Their functions are to provide make-up water to the system and to provide an expansion volume to accommodate system pressure fluctuations. The P50 control air receiver tanks, manufactured by Buffalo Tank, are vertical tanks measuring approximately 14'-0" high and 6'-0' in diameter. Their functions are to dampen system pressure fluctuations and to provide a surge volume to meet sudden demands for control air. The P50 control air dryers with pre-filter and after-filter, manufactured by Pall Pneumatic Products, are arranged as vertical tanks. Each dryer unit contains two dryers measuring approximately 5'-10" long and 6 5/8" in diameter. Moisture is removed from air flowing through the dryer by adsorption to the desiccant. In addition, the pre-filter and after-filter, measuring approximately 2'-6" tall and 8" in diameter, act to clean the air after it is discharged by the compressor and again after it leaves the dryer. The R30 fuel oil tanks, manufactured by Graver Tank, are horizontal tanks approximately 53'-10" long and 12'-0" in diameter. The tanks' 42,000 gallon capacity is sufficient for a seven-day EDG run at full load. The R30 550-gallon fuel oil day tanks, manufactured by Colt Industries, are horizontal tanks approximately 8'-6" long and 3'-6" in diameter. They are sized to support a two-hour EDG run at full load. The R30 275-gallon lube oil tanks, manufactured by Colt Industries, are horizontal tanks approximately 5'-9" long and 3'-0" in diameter which store lube oil for engine lubrication. The R30 EDG jacket coolant expansion tanks, manufactured by Colt Industries, are vertical tanks approximately 2'-4" high and 2'-1" in diameter. The tanks accommodate the volume changes in the jacket coolant and provide the required head for the pumps. The R30 EDG starting air receiver tanks, manufactured by Lasker Boiling and Engineering, are vertical tanks approximately 9'-7" high and 2'-6" in diameter. The tanks store pressurized air to start an EDG without recharging.

There are no flat bottom metal fluid storage tanks in this category.

C. Heat Exchangers – 20 total

Location	PIS No.				
RBSB	E5100B002				
RB2	E1101B001A	E1101B001B	P4400B001	P4400B002	
ABSB	E4100B002				
ABB	P5002B004	P5002B005			
RHR1	R3001B001	R3001B002	R3001B003	R3001B004	R3001B017
	R3001B018	R3001B019	R3001B020	R3001B025	R3001B026
	R3001B027	R3001B028			

The E41 and E51 turbine lube oil coolers are tube and shell heat exchangers, manufactured by Whitlock. The E11 RHR shell and tube heat exchangers, manufactured by Fromson Heat Transfer, are vertically oriented and supported at mid-height and at the top. They measure approximately 25'-2" long and 4'-6" in diameter. Their function is to remove heat from the water used in various plant areas such as primary reactor coolant, torus water, and fuel pool. The P44 EECW heat exchangers, manufactured by Yuba Industries, are horizontal shell and tube type which measure approximately 43'-0" long and 2'-5" in diameter. The P50 control air compressor after-coolers, manufactured by R. P. Adams Company, are a combination horizontal heat exchanger and cyclone separator. The heat exchanger is approximately 11'-2" long and 3 1/2" in diameter. Their function is to lower the temperature and remove the oil and water from compressed air to prevent later condensation in piping and equipment. The R30 heat exchangers, manufactured by American Standards, are arranged in stacked assemblies of three (air coolant, lube oil, and jacket coolant, from bottom to top) on each EDG skid. Their function is to remove heat from air, lube oil, and water used in the operation of the EDGs.

D. Steam Driven Turbines – 2 total

Location	PIS No.
RBSB	E5101C002
ABSB	E4101C002

The HPCI and RCIC steam turbines, manufactured by Terry Turbine, are non-condensing turbines which operate on steam supplied from the reactor vessel. The normal speed range for the turbine is between 2100 rpm and 4000 rpm (HPCI) or 4500 rpm (RCIC).

Equipment Evaluation

The SRT walked down the equipment in this category with the exception of those few described in the outlier section below. Although the equipment in this "other" category does not fall directly into any of the above 21 categories, the screening criteria of EPRI NP-6041 were used as guidelines to evaluate the equipment.

A. Valves

Based on the walkdown and a review of valve drawings, the SRT concluded the following:

1. The HPCI and RCIC control valves consist of hydraulically operated components with an assemblage of linked structural members. The items are of sturdy, rigid construction and are adequately connected and mounted.
2. The HPCI stop valve meets the screening criteria of EPRI NP-6041, Figure F-26.
3. The P50 condensate drain traps are sturdy and well supported.
4. The R30 temperature control valves are sturdy and well supported.

B, C. Tanks and Heat Exchangers

Based on the walkdown and a review of equipment drawings, the SRT concluded the following:

1. The equipment was adequately welded to the support frames or skirts.
2. The overall appearances of the support systems were examined with no apparent weak links.

3. The tall, vertical, RHR heat exchanger had adequate lateral support at mid-height and the top.
4. Piping is attached to the equipment by welding.
5. Saddles or cradles for horizontal equipment are stiffened in weak axis bending.
6. There are no flat bottom metal fluid storage tanks in this category.

D. Steam Turbines

Based on the walkdown and a review of equipment drawings, the SRT concluded the following:

1. The turbines are attached to stiff concrete pedestals.
2. The lateral load resistance system is adequate for each turbine.
3. Relative motion between the turbines and connected piping is not a concern since the piping is either well supported to a common structure or has adequate flexibility to accommodate any relative motion.
4. There are no attachments to the turbines with apparent weak seismic links.

Anchorage Evaluation

The equipment (other than the valves) is attached to the floor slabs or pedestals with either concrete expansion anchors or embedded anchors. Some of the tanks are attached to steel frames which are, in turn, anchored to concrete walls. There are generally no edge distance or spacing violations for the anchors; any violations were evaluated and found acceptable. The anchorage conforms to the design drawings. The nuts are present and appear tight. The anchorage appears to be relatively stiff. Design Calculation DC-5634 was prepared to evaluate the adequacy of the anchorage based on the weakest link identified from the design basis evaluation. The calculation concludes that the anchorage has adequate capacity to withstand the RLE loads.

System Interaction Effects

The general areas around the equipment were reviewed by the SRT for seismic interaction effects. The SRT concluded that the areas are free from interaction concerns, except as described in the outlier section below. Attached tubing and conduits have adequate flexibility to accommodate relative movement. Soft targets on the equipment are free from impact from nearby equipment or structures. The equipment is sufficiently far from adjacent components and structures to preclude any interaction. There is no potential for collapse of adjacent structures or equipment. No potential sources of flooding were discovered that could spray or cascade onto the equipment.

Outliers and Outlier Resolution

The following concerns were identified during the plant walkdowns. Each outlier has been evaluated as indicated.

1. The actuator lid was loose on EDG air coolant system temperature control valve R3000F023D. A work request (WR No. 000Z953615) was initiated to replace the lid.
2. The yokes on the EDG air coolant system temperature control valves (PIS Nos. R3000F023A to R3000F023D) are made from cast iron. However, an evaluation **[3.51]** was performed by the SRT to show that the strength of the yokes is adequate to withstand the RLE loading.
3. Three of the C11 water and nitrogen tanks on the hydraulic control units had either a loose strap or bent or missing hardware. Some of the HCU inlet and outlet scram valve actuators had missing nuts, bolts and/or washers at the support bracket to the frame. Work requests (WR Nos. 000Z947541 to 000Z957543) were prepared to restore the tank support configurations to their original conditions. Work was completed in December 1994.
4. A 1/4-inch rattlespace exists between the insulation on EECW make-up tank P4400A001 and a relief valve discharge piping. Insulation on the other division's EECW make-up tank P4400A002 and a valve in the tank's outlet piping are touching. The SRT concluded that the interactions were acceptable because the interaction would be with the insulation around the tanks and would not adversely affect the tanks' function **[3.51]**.
5. The EDG fuel oil tanks (PIS Nos. R3000A001 to R3000A004) sole plates at one of their support pedestals have slotted holes to allow thermal growth of the tanks. In addition, embedded anchor bolts have edge distance violations which may affect the load carrying capability of the anchorage system. An evaluation **[3.51]** was performed that shows the tank anchorage has adequate capacity to withstand the RLE loading.
6. The HPCI and RCIC lube oil skid piping and lube oil cooler connections are threaded; however, the SRT concluded that this condition is acceptable based on the piping support system and the low seismic accelerations at the sub-basement level.
7. Threaded connections were observed on the HCU assemblies; however, due to the low seismic levels on RB first floor and the well supported piping near the connections, the SRT concluded that the connections are acceptable.
8. A 1/2-inch rattlespace exists between the insulation on EECW heat exchanger P4400B001 and a conduit and its support. The SRT concluded that the rattlespace was acceptable because the interaction would be with the insulation around the heat exchanger and would not adversely affect the heat exchanger's function **[3.51]**.
9. Some small rattlespaces exist between the insulation on north RHR heat exchanger E1101B001A and structural steel members. The SRT concluded that the rattlespace was acceptable because the interaction would be with the insulation around the heat exchanger and would not adversely affect the heat exchanger's function **[3.51]**.

3.1.5 Outliers and Special Evaluations

As a result of the seismic screening evaluation and walkdown of the structures and components, several field conditions and concerns resulting in the need for plant maintenance were identified. Most of these items consisted of loose, missing, or damaged hardware and were handled by initiating plant maintenance work requests. Three conditions requiring design modification were handled by initiating Engineering Design Packages, Technical Service Requests, or Deviation Event Reports. EDP-27,108 **[3.75]** was originated to connect adjacent relay panels together to prevent any panel contact during a seismic event that may result in unanalyzed relay chatter. TSR-27,566 **[3.41]** was approved to replace four low-ruggedness relays in the diesel generator control panels with suitable replacement relays. TSR-28,195 **[3.99]** was originated to address anchorage weaknesses in a non-safety related air dryer tank located on the second floor of the reactor building in the vicinity of SSEL components. DER 96-0289 **[3.81]** was initiated to document an identified weakness in the seismic load path for two large instrumentation panels on the fifth floor of the auxiliary building. The DER will track the resolution of this issue and the implementation of any necessary improvements.

Table 3-7 provides a listing of work requests generated as a result of the seismic walkdown, with a brief description of the anomalies and the resolutions. As a result of the insights gained from the walkdowns, the nuclear training department incorporated additional training for maintenance personnel in their periodic continuing training program. The training concentrates on emphasizing the need for proper installation and restoration of mounting hardware **[3.82]**.

Special evaluations were performed for several other items because of certain concerns or requirements in the EPRI seismic margin approach. These include the evaluation of masonry and shield walls **[3.53]**, reactor internals **[3.83]**, refueling floor superstructure frame embedment **[3.59]**, motor control center maximum seismic capability **[3.50]**, dry transformers lateral load resistance **[3.78]**, deep well vertical pump unsupported casing **[3.79]**, HPCI and RCIC lube oil coolers **[3.80]**, and unrestrained trolleys on 480V switchgear assemblies **[3.77]**. Descriptions of some of these special evaluations are included in the pertinent sections for the equipment category evaluation (Section 3.1.4.9.2). Other evaluations not related to any particular equipment category are described below.

3.1.5.1 Masonry and Shield Walls

Table 2-3 of EPRI NP-6041 indicates that unreinforced or lightly reinforced masonry walls require a margin review. Appendix A of EPRI NP-6041 states that masonry walls which were qualified to the plant SSE, in response to IE Bulletin 80-11 **[3.84]**, using arching or rigid body rocking methods, may have limited capacity beyond the SSE. Therefore, masonry walls qualified using these methods and located near safety related equipment should be investigated for seismic capacity. However, externally reinforced walls using rolled steel sections anchored to floor and ceiling, with bolts going through the walls, do not require investigation for earthquakes less than 0.3g.

At Fermi 2, masonry block walls in Category I structures are mainly used as non-load-bearing partitions. Only minor items such as junction boxes and key card readers are mounted on the walls. In the few cases where the attachments' weight is significant compared to the weight of the wall (more than about 2%), the attachment weight was considered in the design basis analysis performed to comply with IE Bulletin 80-11.

For the seismic margin program, representative masonry and shield walls in Category I structures were selected for bounding analysis. Four walls of different types were considered. The evaluation is documented in design calculation DC-5591 **[3.53]**. A brief description of the margin evaluation is given below.

Wall number 212 is a hollow, 12-inch thick, block wall located on the second floor of the auxiliary building, in the Division I switchgear room. The only reinforcement this wall has is some structural clip angles and plates running along the vertical interface lines between the block wall and the building concrete walls and the interface of the block wall with the ceiling slab. The wall was modeled using plate elements and was analyzed using the ultimate strength procedure and finite element computer method. The maximum stresses in the mortar joints resulting from RLE loading were calculated and compared to the 84% exceedance ultimate capacity of the mortar material. Based on the analysis, the HCLPF value of the wall was calculated as 0.62g.

Wall number 297 is located on the fifth floor of the auxiliary building in the control center HVAC equipment area. This 8-inch thick hollow masonry block wall is externally reinforced with vertical structural steel wide flange members and sandwich steel plates anchored to the floor and ceiling slabs. The most critical section of the wall, which was a section spanning 11'-10" horizontally between external steel columns and 15'-4" vertically between the floor and ceiling slabs, was analyzed as a two-way plate. The plate was assumed as simply supported at the steel columns, fixed at the base and free at the top. The ultimate strength method was used to estimate the maximum mortar joint stress under RLE loading. The maximum joint stress was compared to the 84% exceedance ultimate capacity of the mortar material. The comparison resulted in a HCLPF estimate for this wall of 0.32g.

In addition to masonry block walls, there are several radiation shield walls at Fermi 2. These walls may be categorized in two types: interlocked shield plank and shield block walls. Both types of walls consist of stacked concrete planks or solid blocks without the use of any mortar or other bonding material between the wall elements. Many of these shield walls have been analyzed and externally reinforced to address design basis seismic II/I concerns. A bounding case wall of each type was evaluated for the seismic margin program.

Radiation shield wall number 21 is a 5'-4" thick wall located on the fifth floor of the auxiliary building. It consists of stacked 11'-0" wide concrete planks and a pattern of 8-inch thick solid concrete blocks on top of the p[illegible]ks. The wall is externally reinforced with two clip angles at both sides of the wall [illegible]red directly to the floor to prevent sliding of the bottom plank out of the wall ope[illegible]olid blocks on top of the wall are restrained with a 1/8 inch steel plate which [illegible]red to the planks below and the concrete building walls on the sides of the wall oper[illegible]g. This wall was analyzed using the reserve energy method to study the stability of the wall during an RLE event. Based on the analysis it was determined that this shield wall has a HCLPF value of 2.4g.

The last wall evaluated for the margin program was radiation shield wall number 23. This wall is a 4'-6" thick structure comprised of interlocked 8-inch thick solid concrete blocks. The wall is located on the fifth floor of the auxiliary building between the Standby Gas Treatment System (SGTS) rooms and the control center HVAC equipment room. The wall is covered with a 1/8-inch thick steel plate membrane on one side of the wall. Since the blocks are only restrained laterally by friction forces, the likelihood of blocks ejecting from the wall and falling on vital equipment was investigated. A time-history record for the RLE auxiliary building fifth floor was used to perform a dynamic time-history analysis of one block. The analysis accounted for the vertical seismic component which reduces the friction forces when applied in the upward direction. The analysis also conservatively assumed that the coefficient of friction for sliding of the block back into the wall is 80% of the coefficient of friction of sliding out of the wall. The analysis demonstrated that the block would not slide enough out of the wall to cause it to be unstable and fall out of the wall. Based on the analysis, it was concluded that this wall has a HCLPF value greater than 0.3g.

3.1.5.2 Reactor Internals

Supplement 5 to NRC Generic Letter 88-20 states that, for focused-scope plants, the seismic capacity for reactor internals need not be evaluated for the seismic IPEEE. However, the assessment of the reactor internals at Fermi 2 was completed in the early stages of the seismic margin program before the issuance of Supplement 5. A brief description of the evaluation is given in the following paragraphs.

The evaluation of the Fermi 2 reactor pressure vessel internals is documented in Detroit Edison File number P1-15402 **[3.83]**. The conservative deterministic failure margin (CDFM) method was used to calculate a minimum HCLPF value for the weakest reactor internal component.

Based on a previous seismic evaluation of the reactor internals for the Fermi 2 site specific earthquake, General Electric performed an assessment of the major components in the vessel **[3.85]** for higher seismic loading associated with the site specific earthquake. The GE evaluation was based on the original design basis analysis as summarized in the UFSAR Table 3.7-14. In the GE evaluation the following internal components were reviewed:

- Top Guide
- Core Plate
- Stabilizer
- RPV support
- Shroud support
- CRD housing

- CRD housing restraint beam
- Fuel Assembly

From the review of the GE assessment, it was evident that the weakest component among the internals is the shroud support; therefore, the HCLPF capacity of the shroud support represents a HCLPF of the reactor vessel and its internals.

To estimate the seismic forces on the shroud support resulting from the RLE, the seismic nodal accelerations from the RLE analysis were obtained at the nodes representing the shroud elements in the seismic mathematical model. These accelerations were compared with the pertinent nodal accelerations from the 0.15g PGA analysis used by GE. The ratio of the RLE to SSE nodal accelerations was used to scale up the shear forces and bending moments on the shroud support element calculated in the SSE analysis. This scaling was possible because the same RPV seismic mathematical model was used for both the SSE and the RLE analyses. By evaluating the critical section of the shroud support element to the scaled up forces and moments, a HCLPF value was calculated.

The bounding HCLPF capacity of the RPV shroud support and the reactor internals was calculated to be 0.38g which exceeds the RLE seismic demand of 0.3g; therefore, it is concluded that the reactor vessel and internals at Fermi 2 possess adequate capacity to withstand an earthquake equal to and somewhat greater in magnitude than the RLE.

3.1.5.3 Control Room Ceiling

Table 2-3 of EPRI NP-6041 indicates that the control room ceiling requires a margin review and that it should be inspected for the adequacy of its bracing and/or safety wiring.

The SRT walked down the area above the control room (CR) and computer room to evaluate the ceiling bracing and other equipment anchorage regarding their response to seismic events, i.e., whether the ceiling and/or other equipment could possibly fall down into the control room and affect safe operation of the plant.

Description

The area above the control room consists of the following major items: structural steel beams and hangers which form the main support structures, a grating walkway to allow access to various equipment, lighting assemblies and their support structures, acoustic tile ceiling and its support structure, HVAC ducts and duct supports, cable trays, and electrical conduit and pull boxes. The heaviest pieces of equipment are the fan/coil units above the computer room. However, these are not a concern to control room operations because of their location.

The main support steel above the control room consists of W12 x 14 beams (girders) spanning east-west. Each beam is supported by vertical, double-angle hangers attached to the underside of the fifth floor slab and a beam seat on the west end. The hangers are, in turn, attached to angles which are anchored to the underside of the fifth floor slab by either self-drilling anchors or at "Gateway" inserts. Diaphragm restraint is provided by C4 x 5.4 channels (purlins) running perpendicular to the main support steel and P5000 Unistrut lighting support channels welded to the bottom flange of the main support beams.

The control room ceiling consists of a high ceiling above the operating area and a low ceiling behind the vertical facade which separates the operating area from the back section of the combination operating panels (COPs). The high ceiling, at approximate elevation 662'-8", consists of acoustical tiles, generally two feet by four feet, supported by a steel grid of main runners and cross tees. The tiles are held in place in the grid with hold down clips and nails above the tiles to prevent uplift. The grid is suspended about 32 inches below the support channels by 12-gauge wire hangers at two-foot spacing, each way. The support channels span either six, eight, or eleven feet between the structural steel beams in the north-south direction and are spaced at approximate two foot intervals in the east-west direction. The support channels are welded to the top flange of the steel beams at each end. Where the tiles abut the walls, they are supported by wall molding strips which are attached to the walls.

Each light fixture in the high ceiling has four steel straps attached to the outside of the shade. The straps in turn, are bolted to two P5000 Unistrut channels approximately eight inches apart. The Unistrut channels span either six, eight, or eleven feet between the structural steel beams in the north-south direction and are spaced at approximate four foot intervals in the east-west direction. The Unistrut channels are welded to the bottom flange of the steel beams at each end. The bottom of each shade fits through a hole in the ceiling tile.

The low ceiling, at approximate elevation 653'-4", is similar in construction to the high ceiling. The light fixtures are supported from double Unistrut members similar to the upper fixtures. The acoustic tiles are similar in size to the tiles in the high ceiling but their supporting framework is suspended from the lighting Unistrut members instead of separate channels. The tiles are above north and south portions of the control room behind the COPs. There are no ceiling tiles directly above the back portions of the COPs and the west end of the control room.

Behind the COPs at the west end, fluorescent lights are suspended from 3/16-inch rod hangers which are, in turn, attached to Unistrut members spanning between support beams. The Unistrut members are welded on each end to the low ceiling support steel beams.

Electrical pull boxes, conduits and cable trays are mechanically attached to the supporting steel beams in accordance with Specification 3071-128 **[3.86, 3.87]** standards.

Evaluation

In accordance with EPRI NP-6041, there is a concern that control room hung ceilings, which are typical in nuclear power plants, could fail during seismic events since similar ceilings in commercial buildings and fossil fuel plants have failed at moderate accelerations in past earthquakes. It is the consensus of the SRT that the ceiling tiles would remain essentially intact and not fall onto the control room floor and operating panels during an RLE based on the following reasons:

1. The relatively light weight tiles are supported on all four sides by the grid structure. Uplift is also prevented by the hold down clips and nails through the cross tees and above the tiles.
2. Many of the tiles have a light fixture penetrating their planes. Since the fixture is laterally restrained, it would tend also to restrain lateral movement of the tiles.
3. The grid structure is supported at close intervals (two feet each way) by relatively strong wire (12-gauge).
4. The wires are attached at the top to steel channels welded on each end to the structural support steel.
5. The structural steel framework is well supported by the hangers to the slab above, the west end beam seats, and diagonal bracing.

In addition, the SRT concluded that items located in the area above the control room ceiling will not fail of fall following an RLE as described below:

1. The electrical conduit, pull boxes, cable trays, and HVAC ducts are rigidly attached to their supporting structure and are installed in accordance with seismic specifications.
2. The fluorescent light assemblies in the back of the control room are suspended with threaded rods with relatively rigid connections on each end. The rods are attached to Unistrut members spanning between support steel beams.
3. The effective span of the P5000 Unistrut lighting support members is shortened by the attachment of the light fixture. The fixture tends to cause the two Unistrut members to act together which reduces the chance of buckling.
4. Miscellaneous items such as emergency, four-battery pack lights, other emergency lighting, camera support vertical tube steel, and the digital display panel above the westernmost COP H11P603 are all rigidly attached to their supporting members.

Based on the SRT's evaluation of the control room ceiling area, it is concluded that the Fermi 2 configuration offers good seismic design and connection details and meets all the known caveats for such structures. Therefore, it is judged that the Fermi 2 control room ceiling has a HCLPF of 0.3 or greater.

3.1.5.4 Refuel Floor Superstructure Embedment

As stated in Section 3.1.4.7, a review of the final design calculation **[3.59]** for the analysis of the steel framing of the reactor building crane support and roof above the refueling floor identified a small stress margin in some of the base plate embedment details. Therefore, despite the fact that the EPRI screening did not identify a condition requiring further evaluation, the SRT initiated a more detailed evaluation to check the embedment design against RLE demand loading.

The margin evaluation of the refueling floor embedment is documented in Detroit Edison File number P1-15396 **[3.59]**. The loads on various bounding column base plates were re-calculated using the RLE seismic accelerations and combined using the EPRI methodology. Other conservative steps in the design basis calculation were also revised or eliminated as permitted for the margin evaluation. The total load was then compared with the SSE design basis allowable stresses. It was concluded that the refueling floor superstructure column base plate embedment has adequate capacity to withstand loads from the RLE without compromise to the structural integrity of the support framing.

3.1.6 Analysis of Containment Performance

This section describes the evaluation of the Fermi 2 primary containment seismic capability in accordance with the guidelines provided in NUREG-1407. This evaluation requirement is one of the enhancements requested by the NRC to supplement the seismic margin assessment program described in EPRI NP-6041.

3.1.6.1 Containment Description

The Fermi 2 primary containment (designated by General Electric as a Mark I containment design) houses the reactor vessel, the reactor recirculation loops, and other branch connections of the reactor coolant system. It forms a fission product barrier which, in conjunction with the secondary containment system, contains the radioactive fission products generated during all modes of plant operation and any postulated design basis accident so that off-site doses will not exceed the requirements of 10CFR100. Primary containment is a pressure suppression system. It consists of two major structural components: (1) the drywell and (2) the suppression chamber or wetwell. The inverted light bulb shaped drywell is a steel plate pressure vessel that surrounds the reactor pressure vessel. The drywell is connected by eight vent pipes, each six feet in diameter, to the torus shaped suppression chamber. The suppression chamber, also called the torus or wetwell, contains a large volume of water affording an effective means of pressure suppression if steam is released from the reactor coolant pressure boundary into the drywell. It performs a similar pressure suppression function if steam is released through the safety relief valves on the main steam lines.

The main functions of the primary containment system are:

- To withstand the pressures and temperatures resulting from a loss-of-coolant accident;
- To provide an essentially leak tight barrier against uncontrolled release of radioactivity; and
- To house and support reactor vessel and support equipment.

In addition to those functions specified above, the containment also provides:

- A source of water for the Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), Core Spray (CS), and Low Pressure Coolant Injection (LPCI) systems;
- A heat sink using the suppression pool which, in turn, is cooled by the Residual Heat Removal (RHR) system in the suppression pool cooling mode; and
- A potential scrubbing mechanism in the radionuclide path in the event of a core damage accident using the suppression pool and the drywell sprays.

The drywell is enclosed in a reinforced concrete biological shield that also provides resistance to deformation and buckling. This reinforced concrete structure is an integral part of the reactor building. The internal design pressure for the drywell is 56 psig, which is the saturation pressure for the maximum design temperature of 340°F. The ASME Code, Section III allows a maximum overpressurization of 10 per cent; therefore, a maximum internal drywell pressure of 62 psig has been demonstrated. An ultimate containment pressure capability assessment performed by Chicago Bridge and Iron resulted in a best estimate pressure limit of 140 psig at 340°F with the initial failure point located in the wetwell air space **[3.31, 3.88]**. The maximum external design differential pressure is two psid.

The suppression pool is a torus-shaped, leak-tight steel pressure vessel with a major diameter of 112 ft. 6 in. The pool contains approximately 122,000 cubic feet (one million gallons) of demineralized water. Water volume is controlled over a narrow measured range.

3.1.6.2 Early Failures With High Consequences

The containment performance figure of merit used in the investigation for seismic vulnerabilities is the frequency of early failures of containment with high radionuclide release consequence potential.

The seismic IPEEE containment performance assessment uses this figure of merit as derived from the NRC objectives given in the following:

- Generic Letter 88-20, Supplement 4:
    - Appendix 2, "Containment Performance,"

      "*The evaluation of the containment performance for external events should be directed toward a systematic examination of whether there are sequences that involve containment failure modes distinctly different from those found in the IPE internal events evaluation or contribute significantly to the likelihood of functional failure of the* *containment (i.e., loss of containment barrier independent of core melt).*

The most efficient way to accomplish this is to use the information developed for the IPEEE to:

1. *Identify mechanisms that could lead to containment bypass,*

2. *Identify mechanisms that could cause failure of the containment to isolate, and*

3. *Determine the availability and performance of the containment systems under the external hazard to see if they are different from those evaluated under the internal event evaluation."*

- Section 4.2.2 of Appendix 4, "Documentation,"

  "Any seismically induced containment failures and other containment performance insights. Particularly, vulnerabilities found in the systems/functions which will lead to early containment failure and high consequences. This includes: isolation, bypass, containment integrity and systems (e.g., igniters) required to prevent early failure."

- IPEEE Guidance Document NUREG-1407:

  -Section 3.1.1.5, "Containment Performance,"

  "The purpose of the containment performance evaluation is to identify sequences and vulnerabilities that involve containment, containment functions, and containment systems (e.g., igniters and suppression pools) seismic failure modes or timing that are significantly different from those found in the IPE internal events evaluation."

Based on this guidance, the seismic containment evaluation includes the assessment of containment performance by examining failures that could cause early radionuclide releases of high consequence. Based on the internal events IPE, this includes such failures as containment bypass, containment isolation failure, and containment structural failure. Each of these is discussed in Subsections 3.1.6.3 through 3.1.6.5.

It is noted that the clarification of "large" release cited in the Severe Accident Policy Statement and GL 88-20 to include radionuclide releases that are both early in time and of high consequence is consistent with the Fermi 2 IPE approach **[3.31]**. The definition has been checked with the recently published PSA Applications Guide **[3.89]** criteria which uses Large Early Release Frequency (LERF). It is consistent with, but more conservative than, that definition.

Finally, the most recent Fermi 2 evacuation plan has also determined that the evacuation times associated with an accident are conservatively encompassed by the definition of "early".

3.1.6.3 Containment Integrity and Containment Systems

Containment integrity and containment systems operability are integrally tied to the determination of seismic induced vulnerabilities. Containment structural integrity has been evaluated for Fermi 2 in the following:

- Section 6.2.1.3.5 of the UFSAR describes design basis challenges including high drywell temperature associated with small break LOCAs. The UFSAR challenges come in terms of pressure and temperature. The containment design is 56 psig and 340°F. The systems included in the Success Path Logic Diagram are capable of maintaining the containment within these design specifications for a seismic induced small break LOCA. The severe accident analysis examined in the IPE demonstrated that the Fermi 2 containment is capable of withstanding substantially higher challenges on a best estimate basis. **[3.88]**

- The IPE analysis for internal events examined containment challenges induced by severe accidents causing high pressure and temperature **[3.31]**. The IPE containment performance evaluation has demonstrated that even under severe accident conditions the containment can survive a broad spectrum of temperature and pressure challenges much more severe than those design basis conditions specified in the FSAR. The seismic induced accident sequences for the RLE do not produce containment challenge accident sequences that are more severe or of a different character than already evaluated in the IPE.

- NUREG/CR-5098 **[3.54]** examined containment seismic structural integrity. The results of this analysis indicates that there are no seismic induced structural vulnerabilities and no unique failure modes imposed by an earthquake of larger magnitude than the RLE on the Fermi 2 Mark I containment.

The IPE Levels 1 and 2 analysis was reviewed to identify those systems required to prevent early containment failures with high consequence potential. Containment systems in BWRs are intimately tied to the systems that prevent core damage. Since the SSEL includes systems to prevent core damage, these same systems will provide containment protection and can be grouped as follows:

- Reactivity Control
- Pressure Relief
- Injection

- Depressurization
- Vapor Suppression

These systems are on the SSEL and are found acceptable because they pass the RLE margins assessment. In addition to these systems, combustible gas control is assured with a high probability because of the inerted containment required by Technical Specifications during power operation.

3.1.6.4 Containment Bypass

Containment bypass induced by a seismic event could substantially alter the early/high release category if the conditional failure probability was high. Containment bypass has generally referred to one of two principal failure modes. These failure modes are:

- Unisolated breaks outside containment in systems connected to the primary system.

- Failure of vapor suppression to prevent rapid overpressure failure of the containment.

Both of these failure modes were addressed in the seismic evaluation. The primary systems that penetrate the containment boundary were reviewed for bypass vulnerabilities. This review resulted in the following insights:

- Shutdown Cooling suction lines are normally closed and interlocked shut (One of the valves is deenergized.)

- Low pressure injection lines all contain at least one check valve that is seismically rugged and will prevent over-pressurization of low pressure connected systems. These lines also have a normally closed MOV in series with the check valve.

- High pressure connected systems (e.g., main steam, HPCI, RCIC, RWCU) are designed for high pressure and have isolation valves that have passed the seismic margin assessment screening evaluation at the RLE level demonstrating successful isolation for breaks outside containment.

In addition, the vapor suppression system was evaluated with the following results:

- The vapor suppression system may be needed for some portion of the seismic challenge and subsequent mitigation of events to prevent an early catastrophic containment overpressure. The vapor suppression system is on the SSEL and was shown to have adequate seismic capacity. This includes the torus-to-drywell vacuum breakers, the suppression pool, and the fixed components (downcomers, ring headers).

Containment connections that can result in a release are discussed in Section 3.1.6.5. In summary, no vulnerabilities in the containment or its penetrations were identified due to the RLE event.

3.1.6.5 Containment Isolation

The containment isolation system is normally energized and interruptions in the electrical supply result in a containment isolation. In addition, many normally open isolation valves fail closed on loss of their actuator support. Other normally open paths are associated with closed systems. The seismic capability of these closed systems is expected to be high.

The Fermi 2 containment isolation function has been evaluated deterministically as part of the seismic IPEEE (see Sections 3.1.2.2.4 and 3.1.2.3). This included a seismic assessment of the valves, the containment isolation signals, and the potential for "bad actor" relay chatter. In some cases, one of the two isolation valves, in series in each line, requires air to close. For these valves, the control air system and control air tubing were reviewed and found to be seismically adequate for the RLE. However, for core damage events involving a Station Blackout, there are two lines (the reactor building to torus vacuum breaker lines) which would have one check valve as the containment isolation boundary. This is identical to the qualitative evaluation presented in the internal events IPE. No new insights are derived as part of the seismic evaluation. For the purposes of this evaluation, the closure of a single valve in the line is a successful isolation of that line. This is consistent with the assessment performed in the IPE Level 2 **[3.31]**.

In tests performed for the NRC **[3.90]** to demonstrate seismic capability on containment isolation valves, none of the valves tested experienced any difficulty cycling during or after the seismic motion. In terms of operability, all performed well and were unaffected, both during and after seismic excitation. No observable structural damage occurred to any of the piping, valves, supports, or penetrations.

In summary, no vulnerabilities in the containment isolation system, relays, or containment isolation valves were identified due to the RLE event.

3.1.6.6 Containment Penetrations

The IPE previously reviewed all containment penetrations for severe accident resiliency and determined that there were no vulnerabilities. This included hatches, pipes, and electrical penetration assemblies. Because all equipment and personnel hatches to the drywell/wetwell at Fermi 2 utilize mechanical closure mechanisms with no inflatable seats, there was no need to include any additional support systems for these closures. As with the hatches, the containment penetrations/penetration seals are passive, i.e., they do not rely on pneumatic pressure or electricity for function. SRT walkdown of containment penetrations did not identify any seismic interaction problems or other vulnerabilities. Some containment penetrations are provided with water cooling; however, the cooling is not considered essential for the safety function of the penetrations. Furthermore, during accident conditions, cooling water to these penetrations is normally isolated with activation of the EECW system. The walls and other structural components are considered seismically rugged and capable of withstanding the RLE without any adverse effects on containment performance.

Because the containment structure and piping/valves are expected to survive the RLE, no containment failure modes different than those identified in the IPE for internal events are found.

3.1.6.7 Containment Performance Insights

No seismic induced containment performance vulnerabilities were identified.

Based on a review of the seismic failure modes and the Fermi 2 IPE model, the dominant seismic contributors to core damage and also containment integrity are those which include a loss of offsite power combined with one of the following:

- Failure of on-site AC power sources and failure to recover on-site or offsite AC power in four to six hours when HPCI and RCIC fail due to battery depletion or RPV depressurization.

- Failure of containment heat removal system, i.e., RHR.

Therefore, accident sequences that dominate the seismic induced risk are Station Blackout and Loss of Containment Heat Removal sequences. These were both fully evaluated in the Internal Events IPE submittal, and no containment vulnerabilities were identified. Neither of these seismically induced dominant sequences result in an early radionuclide release of high consequence. Therefore, containment performance is found to be acceptable.

3.1.7 Peer Review

The NRC staff requested **[3.1]** licensees to conduct an independent peer review by individuals who are not associated with the initial IPEEE evaluation to ensure the accuracy of the documentation and to validate the review process and its results. The peer review process is intended to provide a quality control and quality assurance to the IPEEE process to ensure reliability of the evaluation and its conclusions.

The peer review of the Fermi 2 seismic IPEEE process included the use of in-house personnel and outside consultants. The main peer review was conducted at the conclusion of the evaluation; however, several other reviews were performed during the evaluation process to validate the approach and methodology used and ensure compliance with the intent of the evaluation guidelines in EPRI NP-6041 and NUREG-1407. The different areas of reviews are described below in chronological order. More information is provided in Section 6.

1. Dr. Robert P. Kennedy of Structural Mechanics Consulting, Inc. was retained, during the early stages of the program, as a general consultant for the Fermi 2 seismic margin

assessment program. Dr. Kennedy is a well known authority in the seismic evaluation and qualification of nuclear power plant structures and components. He is a co-author of the EPRI NP-6041 report and has been involved in the development of the USI A-46 resolution guidelines as a senior seismic advisory panel member. Dr. Kennedy participated in the Fermi 2 preliminary walkdowns where he pointed out major areas of potential weak links that may require some concentrated effort during the evaluation.

Dr. Kennedy was specifically involved with the generation of new RLE in-structure response spectra, scaling of design basis spectra, and interpretation of the EPRI screening criteria. Dr. Kennedy also prepared HCLPF calculations for the Fermi 2 reactor internals and reviewed the HCLPF calculations for the masonry and shield walls.

2. Mr. Paul Hayes of MPR Associates performed an independent review **[3.91]** of initial efforts in the development of Fermi 2's SSEL. Detroit Edison resolved **[3.92]** Mr. Hays review comments by either incorporating them in the process of developing the SSEL or by providing the appropriate disposition.

3. Mr. Jess Betlack of MPR Associates performed an independent assessment of the Fermi 2 relay evaluation effort. Based on the review, Mr. Betlack concluded **[3.93]** that the Fermi 2 approach for completing the relay review is sufficiently comprehensive for the IPEEE low ruggedness relay review required for a focused scope plant.

4. Mr. Steve Reichle of VECTRA Technologies performed an independent evaluation **[3.9]** of Fermi 2's containment performance review requested in NUREG-1407. The evaluation concluded that the containment performance aspects of the seismic IPEEE process are in compliance with the intent of the NRC guidelines and are consistent with the approach used in other plant studies.

5. Mr. Charbel Abou-Jaoude of VECTRA Technologies conducted a review **[3.95]** of the Fermi 2 IPEEE seismic evaluation program in March 1995. This review was intended to be an intermediate partial peer review of the evaluation which was about 60 percent complete at the time. This review concluded that the Fermi 2 seismic walkdowns and associated documentation were conducted at a very thorough and competent manner. The review also noted that the Fermi 2 plant has seismically rugged structures, systems and equipment compared to other plants.

6. All calculations and evaluations generated as part of the seismic IPEEE program have been independently reviewed by a second engineer knowledgeable in the seismic design and qualification area. This includes calculations and evaluations included with the screening and evaluation work sheets (SEWS) and ones filed separately. All evaluations performed by outside engineering support organizations were also independently reviewed in the same manner.

7. Nuclear operations conducted a review of the selected alternate shutdown paths for the seismic margin evaluation and the components on the SSEL. Additionally, the engineering I&C group reviewed the SSEL for the adequacy of instrumentation and for identifying secondary instruments required for the functioning of the main instruments. Both operations and I&C comments were resolved or incorporated in the final SSEL.

8. A draft version of the seismic IPEEE report was reviewed by several Detroit Edison engineers who have not been directly involved in the seismic evaluation process. Among the reviewers are Mr. A. D. Nayakwadi of the Mechanical and Civil group in Plant Support Engineering, and Mr. Earl Page from the Risk Analysis group. Additionally, sections of the final report were routed to different site organizations for review and comment before the final submittal of this report to the NRC.

   A presentation of the IPEEE study summary, conclusions and results was also given to Fermi 2's senior management staff. The purpose of the presentation was to brief the management staff on the insights and findings resulting from the IPEEE process and inform them of the plant improvements initiated during the course of the evaluation.

8. Dr. John D. Stevenson of Stevenson & Associates performed the final peer review of the Fermi 2 seismic IPEEE program. Dr. Stevenson is a senior seismic consultant in the structural and mechanical engineering area including probabilistic and dynamic analyses. Dr. Stevenson's peer review was performed in three steps. First, a copy of Fermi 2's draft seismic IPEEE report was provided for his review and comment. Second, Dr. Stevenson reviewed 16 SEWS prepared by the Fermi 2 SRT which included various mechanical and electrical components. Third, he walked down the general areas of the plant and the same 16 components reviewed in the SEWS.

   Dr. Stevenson provided several comments **[3.96 and 3.97]** on the draft report, the SEWS, and as a result of the plant walkdown. The main comments are discussed in Section 6. All comments and questions from this peer review were satisfactorily resolved **[3.98]**.

3.1.8 Summary and Conclusions

Fermi 2 has completed an individual plant evaluation for seismic events as requested in Supplement 4 of Generic Letter 88-20. The EPRI NP-6041 seismic margin methodology was used to perform the evaluation. Fermi 2 was classified as a "Focused" scope plant in NUREG-1407. As such, it was required to use a NUREG/CR-0098 median response spectrum anchored at 0.3g. Fermi 2 is not among older plants subject to the NRC Unresolved Safety Issue (USI) A-46; therefore, the IPEEE seismic study was performed independent from other seismic programs. Plant seismic design basis information was used extensively as a starting point in the seismic margin evaluation of structures and components for the IPEEE. In most cases, the design basis seismic information proved to be a very valuable source in the assessment of component seismic capability and in the identification of the margin available above the design basis.

Two alternate safe shutdown success paths were selected in compliance with plant automatic system actuation and operation procedures. The front line systems in the two selected success paths satisfy the four essential safety functions of reactivity control, reactor coolant pressure and inventory controls, and decay heat removal. The systems in each success path are capable of achieving and maintaining plant shutdown for 72 hours following a seismic event. All other support systems required for the functioning of the front line systems were also included in the seismic evaluation program.

A Safe Shutdown Equipment List (SSEL) was compiled by identifying all components required for the successful operation of front line and support systems. For each system, adequate instrumentation was selected to provide the control room operators with information for operating and monitoring the system. Any other instruments and power sources required for the proper functioning of selected control room instrumentation were also added to the SSEL. Primary containment isolation valves were included on the list to assure containment isolation function as required by NUREG-1407.

To address potential malfunction resulting from relay chatter during a seismic event, a screening was performed to locate any known low seismic ruggedness relays and switches used in the systems selected for the seismic IPEEE and for the containment isolation function. This approach is in accordance with the requirements for a focused scope plant evaluation as described in NUREG-1407.

New in-structure demand response spectra were generated for the RLE using slightly modified versions of the design basis seismic models to better account for building torsional response. A synthetic time history consistent with NUREG/CR-0098 rock spectrum was used in the analysis. For a few locations where new demand spectra were not generated, design basis spectra were scaled, in accordance with provisions in EPRI NP-6041, to generate RLE demand spectra.

The screening approach described in EPRI NP-6041 was used in the seismic assessment of structures, systems, and components included in the margin program. Detailed plant walkdowns of the items and areas involved were performed mainly by in-house experienced seismic engineers, trained on the use and application of the EPRI seismic margin method. Several contractors and consultants were also involved in the seismic evaluation to ensure the accuracy of the results and to assist in outlier resolution. Particular emphasis was put on equipment anchorage and identification of potential spatial interaction problems. A bounding anchorage evaluation was prepared to evaluate the capability of SSEL component anchorage to resist the RLE loads. HCLPF calculations were performed for several critical items including masonry block and shield walls, and reactor internals.

By letter submitted in February 1995 **[3.8]**, Detroit Edison informed the NRC of a change in scope for the Fermi 2 seismic IPEEE program from that described in NUREG-1407 for a "focused" scope plant. Detroit Edison indicated that only simple evaluations, mostly based on design basis seismic qualification documentation, will be performed to calculate outlier component HCLPFs. With the exception of calculations performed prior to February 1995, no other highly sophisticated HCLPF calculations would be generated for program completion. This change in scope was considered commensurate with the revised seismic hazard estimates published in NUREG-1488. Thus, for components that do not meet the RLE seismic demand requirements using the simple evaluation techniques, demonstration of seismic design basis compliance was considered adequate for the IPEEE program.

The seismic margin assessment included a containment performance evaluation, which was conducted to study early containment failure modes and evaluate its essential functions. The containment performance evaluation was conducted in accordance with the requirements and guidelines of Generic Letter 88-20, Supplement 4 and NUREG-1407.

At the conclusion of the seismic IPEEE study for Fermi 2, an independent peer review was performed by a known seismic expert to further validate the results of the evaluation. The peer review involved an evaluation of the approach and methodology used in performing the Fermi 2 IPEEE study, as well as conclusions drawn from the program.

3.1.8.1 Results of Evaluation

All structures, systems and components included in the seismic margin evaluation were assessed for their capability to withstand the RLE and perform their intended function in the plant shutdown scenario. Several outlier conditions were identified during the seismic capability walkdowns. Many of these conditions involved component mounting hardware deficiencies that were addressed through normal plant corrective maintenance procedures. Another significant number of outliers was for spatial interaction conditions that presented potential effects on component functionality during and after a seismic event. Most spatial interaction issues were resolved analytically with the exception of one which was handled through a maintenance work request by eliminating the interaction.

The seismic evaluation also resulted in several potential plant modifications. An EDP **[3.75]** was prepared to tie adjacent relay panels together to eliminate potential sensitive relay malfunction resulting from panel interaction. A TSR **[3.41]** has been approved to replace four "bad-actor" relays found in the emergency diesel generator (EDG) control panels. Another TSR **[3.99]** was initiated to strengthen the anchorage of a non-safety related tank in the vicinity of SSEL components. A DER **[3.81]** was issued to address a weak seismic load path in two instrumentation panels.

In addition to the items above, several design basis documentation deficiencies were identified. Documentation update to reflect plant conditions was initiated in accordance with plant procedures.

Despite the change in scope of the Fermi 2 seismic IPEEE program **[3.8]**, with the completion of the plant modifications and corrective maintenance activities resulting from the program, all outliers identified during the seismic evaluation and walkdowns are shown to have adequate capability to withstand the prescribed RLE without degradation of the components or pertinent systems. As a result, this study has demonstrated, by using the above-described methodology, that the plant seismic HCLPF at Fermi 2 is equal to or greater than 0.3g. This conclusion is reached from the screening results of all SSEL components in addition to other structures and distribution systems and from demonstrating a HCLPF value of 0.3g or greater for all evaluated items.

With the exception of the four EDG bad-actor relays, relay screening for known low ruggedness contact devices found no applications of bad-actor relays that would have any adverse effect on the components and systems required for plant safe shutdown.

3.1.8.2 Problem Areas and Insights Gained

The following are the main insights gained from the completion of the seismic margin assessment study at Fermi 2:

1. As a result of the implementation timetable of the seismic margin program and the plant schedule for refueling outages, it was necessary to perform seismic walkdowns of areas of the plant that are normally inaccessible during plant operation, such as inside containment and in the main steam tunnel, before any other walkdowns were completed. This arrangement proved to be somewhat inconvenient because the SRT had to spend more time in radiation areas than what would have been the case if similar components had been walked down in the general plant areas. It was also necessary to perform follow-up walkdowns in these normally inaccessible areas because not all the required information was obtained the first time around. Therefore, if possible, it would be preferable to perform radiation area walkdowns after completing the walkdowns in other areas where the SRT can gain experience learning the walkdown evaluation process.

2. Operations personnel at Fermi 2 are routinely trained on accident scenarios such as Loss of Offsite Power and Station Blackout. However, all training scenarios assume that the Combustion Turbine Generator (CTG) number 11 located near the plant is restorable a short time after the accident to provide offsite power to the plant. For the seismic margin evaluation, the plant is assumed to lose off-site power. Additionally, the reliability of the CTG-11 after a seismic event is greatly in question; therefore, the plant is required to achieve and maintain safe shutdown for 72 hours, using the EDGs as the only power source available. This situation may present different challenges to the plant operators for which they have received no training.

As a result, the nuclear training department will incorporate the seismic margin accident scenario in their future operator training plans.

3. Relay bad-actor screening indicates that, during a seismic event, it is possible that control room operators would receive a large number of alarms resulting from relay chatter of bad-actor relays. Such chatter was considered insignificant in the evaluation of its impact on component and system functions. However, it is realized that it may cause significant confusion in the control room due to the potentially large number of annunciators involved. Therefore, the nuclear training department will also include this scenario in the operator training plans.

4. As a result of the review of Table 3-7, which summarizes the corrective maintenance activities initiated throughout this program, it was realized that many of the discovered discrepancies involve missing, loose or damaged mounting hardware. Furthermore, it seemed like, for the most part, the deficiencies were the result of corrective maintenance activities rather than original mounting installation of the item. It was considered prudent to enhance maintenance personnel training and awareness of the importance of the installation and restoration of mounting hardware. Therefore, lessons were incorporated **[3.82]** in the maintenance personnel continuing training sessions for the second quarter of 1996 to emphasize this point.

3.2 USI A-45 and Other Seismic Safety Issues

In Generic Letter 88-20, Supplement 4 and NUREG-1407, the NRC identified several other external events programs to be coordinated with the IPEEE. Three of these programs were considered subsumed in the IPEEE. They are USI A-45, GI-131, and the eastern U.S. seismicity issue. Three other programs were considered either resolved or nearing completion; however, some coordination with the IPEEE may still be required. These programs are USI A-17, USI A-40 and USI A-46. The applicability of all six programs to Fermi 2 and their resolution are discussed below.

1) **USI A-45**, "Shutdown Decay Heat Removal Requirements." The objective of this program is to determine whether the decay heat removal (DHR) function at operating plants is adequate and if cost-effective improvements can be identified. The Fermi 2 IPE for internal initiating events evaluated the DHR system adequacy as reported in Section 3.4.3 of the IPE report **[3.31]**. The conclusion of the IPE study with respect to the DHR system includes the following statement: "*The IPE evaluation supports the conclusion that no vulnerabilities exist at Fermi 2 to adversely affect the operator's ability to accomplish the DHR function during an accident.*"

   With respect to seismic events, the SMA approach is based on the selection of two alternate safe shutdown success paths. Both success paths must include the capability of decay heat removal as one of the four essential safety functions required. Both success paths selected for the Fermi 2 SMA include one or more DHR modes of the RHR system. Therefore, an assessment of seismic adequacy of the suppression pool cooling and shutdown cooling modes of the RHR system was included in the seismic IPEEE program. All necessary support systems for RHR were also included in the seismic assessment.

   All components required for the decay heat removal functions were identified on the SSEL and were seismically evaluated for the RLE. No vulnerabilities were identified as a result of the seismic evaluation. Minor seismic interaction issues related to components in the RHR system were identified and satisfactorily resolved during the program. The conclusion of the seismic IPEEE study is that the DHR system at Fermi 2 is capable of performing its intended safety function without any degradation resulting from the RLE seismic event.

2) **GI-131**, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants." The Fermi 2 utilizes a General Electric Boiling Water Reactor design; therefore, the GI-131 program is not applicable to Fermi 2.

3) **The Eastern U.S. Seismicity (The Charleston Earthquake Issue)**. The objective of this program is to resolve concerns related to the possibility of the occurrence of large earthquakes at nuclear plant sites east of the Rocky Mountains. In the resolution steps of this issue, probabilistic seismic hazard estimates were developed by both

NRC/LLNL and EPRI for all affected sites. These estimates were utilized by the NRC in the determination of the seismic scope of IPEEE review for each plant. Hence, this IPEEE submittal provides a resolution of the Eastern U.S. seismicity issue for Fermi 2 without any additional work or documentation.

4) **USI A-17**, "System Interactions in Nuclear Power Plants." This unresolved safety issue deals with possible system interactions that could affect redundancy and independence of safety systems. Seismic spatial system interaction has been addressed at Fermi 2 as part of the plant design and licensing. As discussed in Section 3.1.1.1.3.2, Fermi 2 maintains a "rattlespace" program to identify and disposition system interactions that involve safety-related components in the plant. Additionally, the seismic IPEEE program addressed spatial interactions as part of the seismic margin assessment screening and evaluation walkdowns of the structures, systems and components included in the program.

5) **USI A-40**, "Seismic Design Criteria." This program deals with the concern regarding seismic adequacy of large safety-related, above-ground, flat-bottom storage tanks for SSE loading. The seismic IPEEE success paths did not take credit for any large flat-bottom tanks. Furthermore, there are no such safety-related tanks at Fermi 2; therefore, USI A-40 is not relevant to Fermi 2.

6) **USI A-46**, "Verification of Seismic Adequacy of Equipment in Operating Plants." Implementation of the USI A-46 program involves plants with construction permit applications docketed before about 1972. The construction permit for Fermi 2 was docketed on September 26, 1972; therefore, Fermi 2 was not included in the subset of nuclear plants requested to perform a USI A-46 review. In 1981, during the licensing process of Fermi 2, the NRC conducted a "SQRT audit" to review the seismic qualification program. The audit concluded that the Fermi 2 seismic qualification program meets all the applicable NRC requirements. Therefore, USI A-46 is not applicable to Fermi 2.

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3.94 VECTRA Technologies S. Reichle Letter to Detroit Edison's A. I. Hassoun, "Containment Evaluation Review for Fermi Seismic IPEEE," Letter No. 0102-00058.000, November 18, 1994. (Detroit Edison DSN: 0102-00058.000)

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3.97 Stevenson & Associates J. D. Stevenson Letter to Detroit Edison's A. I. Hassoun, "Comment Result of Review of a Sample of SEWS," February 19, 1996. (Detroit Edison DSN: 96C1910A 2.13)

3.98 A. I. Hassoun Memorandum to K. E. Howard, Detroit Edison Memorandum No. TMPE-96-0103, "Resolution of the Seismic IPEEE Peer Review Comments," March 8, 1996.

3.99 Detroit Edison Technical Service Request No. 28,195, "Restraint of ILRT Air Dryer Tank on Second Floor Reactor Building," February 28, 1996.

Table 3-1 Seismically Rugged Items

The following items are considered to be rugged and not vulnerable to a seismic event. A seismic evaluation of these component types is not required for the IPEEE.

1 - Piping

2 - Manual Valves

3 - Check Valves

4 - Restricting Orifices

5 - Flexible Hoses

6 - Filters

7 - Strainers

8 - Pressure Taps

9 - Pressure Elements

Table 3-2 Success Paths Safe Shutdown System Functions

Function	Preferred Success Path	Alternate Success Path
Reactor reactivity control	Reactor reactivity is controlled by inserting the control rods into the core. The control rod drive mechanisms and their corresponding hydraulic control units operate independently of one another.	Same as for the preferred success path. Since the standby liquid control system is not considered as a viable option given a SME, there are no alternate systems for reactor reactivity control.
Reactor coolant system pressure control	Safety relief valves in the safety mode are selected for primary pressure control. (Five ADS valves are specifically chosen.) Also, HPCI and RCIC use reactor steam to run steam turbines, condensing to the torus, and thus provide additional limited pressure control, adequate for lower decay heats.	If high pressure injection is not available, the primary system can be depressurized using the ADS SRVs. These valves are used since they have nitrogen accumulators and additional pneumatic support is not required for their operation.
Reactor coolant system inventory control	The reactor coolant inventory is controlled by RCIC or HPCI. It is assumed that makeup is from the torus. Therefore, the CST need not be evaluated.	Given successful depressurization of the reactor vessel, the LPCI mode of RHR would provide adequate inventory control to the reactor.
Decay heat removal	Decay heat can be removed via the suppression pool cooling mode of RHR.	In addition to suppression pool cooling, the shutdown cooling mode of RHR could be used to remove decay heat from the reactor given that the reactor vessel has been depressurized.

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
1	B2100F010A	7011	FEEDWATER A INBD AOV	DW	1	595 0		DW-01
2	B2100F010B	7008	FEEDWATER B INBD AOV	DW	1	595 0		DW-01
3	B2100F076A	7012	FEEDWATER A OUTBD AOV	RB	1	593 0		STNL-01
4	B2100F076B	7009	FEEDWATER B OUTBD AOV	RB	1	589 6		STNL-01
5	B2103F022A	7000	MSIV A INBD AOV	DW	1	58800		DW-18
6	B2103F022B	7002	MSIV B INBD AOV	DW	1	58800		DW-18
7	B2103F022C	7004	MSIV C INBD AOV	DW	1	58800		DW-18
8	B2103F022D	7006	MSIV D INBD AOV	DW	1	58800		DW-18
9	B2103F028A	7001	MSIV A OUTBD AOV	RB	1	59806		STNL-02
10	B2103F028B	7003	MSIV B OUTBD AOV	RB	1	59806		STNL-02
11	B2103F028C	7005	MSIV C OUTBD AOV	RB	1	59806		STNL-02
12	B2103F028D	7007	MSIV D OUTBD AOV	RB	1	59806		STNL-02
13	B2104A003A	3007	ACCUMULATOR TANK	DW	1	61603	B2104F013R	DW-10
14	B2104A003B	3052	ACCUMULATOR TANK	DW	1	61903	B2104F013H	DW-10
15	B2104A003C	3067	ACCUMULATOR TANK	DW	1	61903	B2104F013P	DW-10
16	B2104A003D	3037	ACCUMULATOR TANK	DW	1	61706	B2104F013J	DW-10
17	B2104A003E	3022	ACCUMULATOR TANK	DW	1	61509	B2104F013E	DW-10
18	B2104F013E	3016	SRV	DW	1	612 9		DW-09
19	B2104F013H	3046	SRV	DW	1	612 9		DW-09
20	B2104F013J	3031	SRV	DW	1	612 9		DW-09
21	B2104F013P	3061	SRV	DW	1	612 9		DW-09
22	B2104F013R	3001	SRV	DW	1	612 9		DW-09
23	B21F013E	3017	SOLENOID VALVE	DW	1	61209	B2104F013E	
24	B21F013H	3047	SOLENOID VALVE	DW	1	61209	B2104F013H	
25	B21F013J	3032	SOLENOID VALVE	DW	1	61209	B2104F013J	
26	B21F013P	3062	SOLENOID VALVE	DW	1	61209	B2104F013P	
27	B21F013R	3002	SOLENOID VALVE	DW	1	61209	B2104F013R	
28	B21F022A	8500	MSIV A INB.ISO.VLV. SOL.VLV.	DW	1	589-6	B2103F022A	DW-23
29	B21F022B	8501	MSIV B INB.ISO.VLV. SOL.VLV.	DW	1	589-6	B2103F022B	DW-23
30	B21F022C	8502	MSIV C INB.ISO.VLV. SOL.VLV.	DW	1	589-6	B2103F022C	DW-23
31	B21F022D	8503	MSIV D INB.ISO.VLV. SOL.VLV.	DW	1	589-6	B2103F022D	DW-23
32	B21F028A	8504	MSIV D OTB.ISO.VLV. SOL.VLV.	RB	1	598-6	B210[illegible]028A	STNL-08
33	B21F028B	8505	MSIV D OTB.ISO.VLV. SOL.VLV.	RB	1	598-6	B2103[illegible]28B	STNL-08
34	B21F028C	8506	MSIV D OTB.ISO.VLV. SOL.VLV.	RB	1	598-6	B2103F028C	STNL-08
35	B21F028D	8507	MSIV D OTB.ISO.VLV. SOL.VLV.	RB	1	598-6	B2103F028D	STNL-08
36	B21K401	8508	ERIS RPV H2O LVL.SIG.CND.	AB	2	613-6	H11P612	
37	B21K402A	8509	ERIS RPV H2O LVL.SIG.CND.	AB	2	613-6	H11P613	
38	B21K402B	8510	ERIS RPV H2O LVL.SIG.CND.	AB	2	613-6	H11P612	
39	B21K609A	3246	OUTPUT PS - B31N111A	AB	4	65906	H21P080	
40	B21K609B	3247	OUTPUT PS - B31N112A	AB	4	65906	H21P081	
41	B21K609C	3248	OUTPUT PS - B31N113A	AB	4	65906	H21P080	
42	B21K609D	3249	OUTPUT PS - B31N114A	AB	4	65906	H21P081	
43	B21K610A	3214	POWER SUPPLY	AB	4	65906	H21P082	
44	B21K610B	3216	POWER SUPPLY	AB	4	65906	H21P083	
45	B21K610C	3215	POWER SUPPLY	AB	4	65906	H21P082	
46	B21K610D	3217	POWER SUPPLY	AB	4	65906	H21P083	
47	B21K613A	3212	POWER SUPPLY	AB	2	61306	H11P613	
48	B21K613B	3213	POWER SUPPLY	AB	2	61306	H11P612	
49	B21K801A	3296	OUTPUT PS - E41N602A	AB	2	61306	H11P614	
50	B21K801B	3297	OUTPUT PS - E41N602B	AB	2	61306	H11P614	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
51	B21K815	3250	INTERM. INST. - B21R802	AB	2	61306	H11P917A	
52	B21K816	3252	INTERM. INST. - B21R801	AB	2	61306	H11P917A	
53	B21K817	3253	INTERM. INST. - B21R801	AB	2	61306	H11P917A	
54	B21K827	3254	INTERM. INST. - B21R803	AB	2	61306	H!1P917B	
55	B21K828	3256	INTERM. INST. - B21R804	AB	2	61306	H11P917B	
56	B21K829	3257	INTERM. INST. - B21R804	AB	2	61306	H11P917B	
57	B21K839	3258	INTERM. PS - B21R801	AB	2	61306	H11P917A	
58	B21K842	3259	INTERM PS - B21R802	AB	2	61306	H11P917B	
59	B21K845	3260	INTERM. INST. - B21R802	AB	2	61306	H11P917A	
60	B21K846	3262	INTERM. INST. - B21R801	AB	2	61306	H11P917A	
61	B21K847	3264	INTERM. PS - B21R803	AB	2	61306	H11P917B	
62	B21K848	3266	INTERM. INST. - B21R804	AB	2	61306	H11P917B	
63	B21K849	3268	FEED - B21R807	AB	2	61306	H11P917A	
64	B21K850	3269	FEED - B21R803	AB	2	61306	H11P917B	
65	B21K857A	8511	ERIS RPV H2O LVL.SIG.CND.	AB	4	659-6	H21P082	
66	B21N080A	3166	LEVEL TRANSMITTER	RB	2	61306	H21P004	
67	B21N080B	3165	LEVEL TRANSMITTER	RB	2	61306	H21P004	
68	B21N080C	3188	LEVEL TRANSMITTER	RB	2	61306	H21P005	
69	B21N080D	3189	LEVEL TRANSMITTER	RB	2	61306	H21P005	
70	B21N081A	3168	LEVEL TRANSMITTER	RB	2	61306	H21P004	
71	B21N081B	3167	LEVEL TRANSMITTER	RB	2	61306	H21P004	
72	B21N081C	3194	LEVEL TRANSMITTER	RB	2	61306	H21P005	
73	B21N081D	3195	LEVEL TRANSMITTER	RB	2	61306	H21P005	
74	B21N085A	3105	LEVEL TRANSMITTER	RB	1	58306	H21P009	
75	B21N085B	3118	LEVEL TRANSMITTER	RB	1	58306	H21P010	
76	B21N090A	3101	PRESSURE TRANSMITTER	RB	2	61306	H21P004	
77	B21N090B	3114	PRESSURE TRANSMITTER	RB	2	61306	H21P005	
78	B21N090C	3103	PRESSURE TRANSMITTER	RB	1	58306	H21P009	
79	B21N090D	3116	PRESSURE TRANSMITTER	RB	1	58306	H21P010	
80	B21N091A	3106	LEVEL TRANSMITTER	RB	2	61306	H21P004	
81	B21N091B	3119	LEVEL TRANSMITTER	RB	2	61306	H21P005	
82	B21N091C	3109	LEVEL TRANSMITTER	RB	2	61306	H21P004	
83	B21N091D	3122	LEVEL TRANSMITTER	RB	2	61306	H21P005	
84	B21N094A	3128	PRESSURE TRANSMITTER	RB	2	61306	H21P004	
85	B21N094B	3140	PRESSURE TRANSMITTER	RB	2	61306	H21P005	
86	B21N094C	3134	PRESSURE TRANSMITTER	RB	2	61306	H21P004	
87	B21N094D	3146	PRESSURE TRANSMITTER	RB	2	61306	H21P005	
88	B21N094E	3130	PRESSURE TRANSMITTER	RB	2	61306	H21P004	
89	B21N094F	3142	PRESSURE TRANSMITTR	RB	2	61306	H21P005	
90	B21N094G	3136	PRESSURE TRANSMITTER	RB	2	61306	H21P004	
91	B21N094H	3148	PRESSURE TRANSMITTER	RB	2	61306	H21P005	
92	B21N110A	3153	PRESSURE TRANSMITTER	RB	2	61306	H21P004	
93	B21N110B	3176	PRESSURE TRANSMITTER	RB	2	61306	H21P005	
94	B21N110C	3154	PRESSURE TRANSMITTER	RB	2	61306	H21P004	
95	B21N110D	3177	PRESSURE TRANSMITTER	RB	2	61306	H21P005	
96	B21N111A	3156	PRESSURE TRANSMITTER	RB	2	61306	H21P004	
97	B21N111B	3180	PRESSURE TRANSMITTER	RB	2	61306	H21P005	
98	B21N111C	3157	PRESSURE TRANSMITTER	RB	2	61306	H21P004	
99	B21N111D	3179	PRESSURE TRANSMITTER	RB	2	61306	H21P005	
100	B21N410E	3027	PRESSURE SWITCH	DW	1	61006	B21P402E	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
101	B21N410H	3057	PRESSURE SWITCH	DW	1	61100	B21P402H	
102	B21N410J	3042	PRESSURE SWITCH	DW	1	61100	B21P402J	
103	B21N410P	3072	PRESSURE SWITCH	DW	1	61006	B21P402P	
104	B21N410R	3012	PRESSURE SWITCH	DW	1	61100	B21P402R	
105	B21N411E	3028	PRESSURE SWITCH	DW	1	61006	B21P402E	
106	B21N411H	3058	PRESSURE SWITCH	DW	1	61100	B21P402H	
107	B21N411J	3043	PRESSURE SWITCH	DW	1	61100	B21P402J	
108	B21N411P	3073	PRESSURE SWITCH	DW	1	61006	B21P402P	
109	B21N411R	3013	PRESSURE SWITCH	DW	1	61100	B21P402R	
110	B21N450	3270	FEED - B21R803	RB	1	58306	H21P423B	
111	B21N451	3271	SOURCE INST. - B21R802	RB	1	58306	H21P423A	
112	B21N610A	3277	OUTPUT INST. - B21N110A	AB	4	65906	H21P082	
113	B21N610B	3278	OUTPUT INST. - B21N110B	AB	4	65906	H21P083	
114	B21N610C	8512	LPCI TRIP UNIT SIG.COND.	AB	4	659-6	H21P082	
115	B21N610D	8513	LPCI TRIP UNIT SIG.COND.	AB	4	659-6	H21P083	
116	B21N680A	3279	OUTPUT INST. - B21N080A	AB	4	65906	H21P084	
117	B21N680B	3280	OUTPUT INST. - B21N080B	AB	4	65906	H21P086	
118	B21N680C	3281	OUTPUT INST. - B21N080C	AB	4	65906	H21P085	
119	B21N680D	3282	OUTFUT INST. - B21N080D	AB	4	65906	H21P087	
120	B21N681A	3283	OUTPUT INST. - B21N081A	AB	4	65906	H21P084	
121	B21N681B	3284	OUTPUT INST. - B21N081B	AB	4	65906	H21P086	
122	B21N681C	3285	OUTPUT INST. - B21N081C	AB	4	65906	H21P085	
123	B21N681D	3286	OUTPUT INST. - B21N081D	AB	4	65906	H21P087	
124	B21N684A	3292	OUTPUT INST. - B21N081A	AB	4	65906	H21P084	
125	B21N684B	3293	OUTPUT INST. - B21N081B	AB	4	65906	H21P086	
126	B21N684C	3294	OUTPUT INST. - B21N081C	AB	4	65906	H21P085	
127	B21N684D	3295	OUTPUT INST. - B21N081D	AB	4	65906	H21P087	
128	B21N685A	3287	INTERM. INST. - B21N085A	AB	4	65906	H21P080	
129	B21N685B	3288	INTERM. INST. - B21N085B	AB	4	65906	H21P081	
130	B21N690A	3102	PRESSURE INDICATOR	AB	4	65906	H21P082	
131	B21N690B	3115	PRESSURE INDICATOR	AB	4	65906	H21P083	
132	B21N690C	3104	PRESSURE INDICATOR	AB	4	65906	H21P080	
133	B21N690D	3117	PRESSURE INDICATOR	AB	4	65906	H21P081	
134	B21N691A	3107	LEVEL INDICATOR	AB	4	65906	H21P082	
135	B21N691B	3120	LEVEL INDICATOR	AB	4	65906	H21P083	
136	B21N691C	3110	LEVEL INDICATOR	AB	4	65906	H21P082	
137	B21N691D	3123	LEVEL INDICATOR	AB	4	65906	H21P083	
138	B21N692A	3108	LEVEL SWITCH	AB	4	65906	H21P082	
139	B21N692B	3121	LEVEL SWITCH	AB	4	65906	H21P083	
140	B21N692C	3111	LEVEL SWITCH	AB	4	65906	H21P082	
141	B21N692D	3124	LEVEL SWITCH	AB	4	65906	H21P083	
142	B21N693A	8514	RPV LEVEL LEVEL SWITCH	AB	4	659-6	H21P082	
143	B21N693B	8515	RPV LEVEL LEVEL SWITCH	AB	4	659-6	H21P082	
144	B21N693C	8516	RPV LEVEL LEVEL SWITCH	AB	4	659-6	H21P083	
145	B21N693D	8517	RPV LEVEL LEVEL SWITCH	AB	4	659-6	H21P083	
146	B21N694A	3129	TRIP UNIT	AB	4	65906	H21P082	
147	B21N694B	3141	TRIP UNIT	AB	4	65906	H21P083	
148	B21N694C	3135	TRIP UNIT	AB	4	65906	H21P082	
149	B21N694D	3147	TRIP UNIT	AB	4	65906	H21P083	
150	B21N694E	3131	TRIP UNIT	AB	4	65906	H21P082	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
151	B21N694F	3143	TRIP UNIT	AB	4	65906	H21P083	
152	B21N694G	3137	TRIP UNIT	AB	4	65906	H21P082	
153	B21N694H	3149	TRIP UNIT	AB	4	65906	H21P083	
154	B21P400	1830	RELAY PANEL	AB	3	64306	WALL	SGR2-07
155	B21P401	1831	RELAY PANEL	AB	3	64306	WALL	SGR2-07
156	B21P402E	3298	INSTRUMENT RACK	DW	1	61100	B21P402E	DW-04
157	B21P402H	3299	INSTRUMENT RACK	DW	1	61006	B21P402H	DW-04
158	B21P402J	3300	INSTRUMENT RACK	DW	1	61100	B21P402J	DW-04
159	B21P402P	3301	INSTRUMENT RACK	DW	1	61100	B21P402P	DW-04
160	B21P402R	3302	INSTRUMENT RACK	DW	1	61100	B21P402R	DW-04
161	B21R615	3290	OUTPUT INST. - B21N085B	AB	3	64306	H11P602	
162	B21R623A	3237	POST ACCIDENT RPV LEVEL/PRESSU	AB	3	64306	H11P601	
163	B21R623B	3238	POST ACCIDENT RPV LEVEL/PRESSU	AB	3	64306	H11P602	
164	B3100F014A	7034	RECIRC PUMP SEAL INBD ISO VA	DW	B	57500	V8-3710	DW-19
165	B3100F014B	7032	RECIRC PUMP SEAL INBD ISO VA	DW	B	580 0	V8-3590	DW-19
166	B3100F016A	7035	RECIRC PUMP SEAL OUTBD ISO VA	RB	1	592 0	V8-3767	RB1-03
167	B3100F016B	7033	RECIRC PUMP SEAL OUTBD ISO VA	RB	1	595 0	V8-3768	RB1-03
168	B31N111A	3504	SOURCE INST. - B31N611A	RB	B	56200	H21P006	
169	B31N111B	3505	SOURCE INST. - B31N611B	RB	B	56200	H21P022	
170	B31N611A	2101	OUTPUT INST. - B31N111A	AB	4	65906	H21P080	
171	B31N611B	2102	OUTPUT INST. - B31N111B	AB	4	65906	H21P081	
172	C1100F010	2004	SDV VENT VALVE	RB	1	600 7		RB1-11
173	C1100F011	2005	SDV DRAIN VALVE	RB	B	57311	V30-0012	RBTR-08
174	C1100F180	2006	SDV VENT VALVE	RB	1	600 7		RB1-12
175	C1100F181	2007	SDV DRAIN VALVE	RB	B	57311	V30-0011	RBTR-08
176	C1102D001	2019	CRD 1 THRU 185	DW	1	587 0		
177	C1103D001	2018	HCU 1 THRU 185	RB	1	586 6		RB1-21
178	C11F160A	2008	ARI SOLENOID VALVE	RB	1	59300		RB1-07
179	C11F160B	2009	ARI SOLENOID VALVE	RB	1	59300		RB1-07
180	C11F162A	2010	ARI SOLENOID VALVE	RB	1	59100		RB1-06
181	C11F162B	2011	ARI SOLENOID VALVE	RB	1	58600		RB1-06
182	C11F162C	2012	ARI SOLENOID VALVE	RB	1	59008		RB1-06
183	C11F162D	2013	ARI SOLENOID VALVE	RB	1	59008		RB1-06
184	C11F163A	2014	ARI SOLENOID VALVE	RB	1	58702		RB1-05
185	C11F163B	2015	ARI SOLENOID VALVE	RB	1	58600		RB1-05
186	C11F182A	2001	SDV VENT AND DRAIN SOLENOID VA	RB	1	58306	C11P401	RB1-08
187	C11F182B	2002	SDV VENT AND DRAIN SOLENOID VA	RB	1	58306	C11P401	RB1-08
188	C11F409 A/B	2003	SDV VENT AND DRAIN SOLENOID VA	RB	1	58700	C11P401	RB1-09
189	C11P401	3613	INSTRUMENT RACK	RB	1	58306	FLOOR	RB1-10
190	C35K410	8518	PRV LEVEL SIG. COND.	AB	4	659-6	H21P082	
191	C35K800	8519	DW PR.RSD PR.SIG.COND.	AB	2	613-6	H11P612	
192	C35K801	8520	RPV LVL.RSD LVL.SIG.COND.	AB	2	613-6	H11P612	
193	C35K803	8521	HPCI HDR.FL.RSD SIG.COND.	AB	2	613-6	H11P612	
194	C35R001	8522	DW PRES. INDICATOR	RB	2	613-6	H21P100	
195	C71K609A	2111	OUTPUT PS - B21N080A	AB	4	65906	H21P084	
196	C71K609B	2112	OUTPUT PS - B21N080B	AB	4	65906	H21P086	
197	C71K609C	2113	OUTPUT PS - B21N080C	AB	4	65906	H21P085	
198	C71K609D	2114	OUTPUT PS - B21N080D	AB	4	65906	H21P087	
199	C71K610A	2115	OUTPUT PS - B21N080A	AB	4	65906	H21P084	
200	C71K610B	2116	OUTPUT PS - B21N080B	AB	4	65906	H21P086	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
201	C71K610C	2117	OUTPUT PS - B21N080C	AB	4	65906	H21P085	
202	C71K610D	2118	OUTPUT PS - B21N080D	AB	4	65906	H21P087	
203	C71N050A	2119	SOURCE INSTRUMENT	RB	2	61306	H21P004	
204	C71N050B	2120	SOURCE INSTRUMENT	RB	2	61306	H21P004	
205	C71N050C	2121	SOURCE INSTRUMENT	RB	2	61306	H21P005	
206	C71N050D	2122	SOURCE INSTRUMENT	RB	2	61306	H21P005	
207	C71N650A	2123	OUTPUT INST. - C71N050A	AB	4	65906	H21P084	
208	C71N650B	2124	OUTPUT INST. - C71N050B	AB	4	65906	H21P086	
209	C71N650C	2125	OUTPUT INST. - C71N050C	AB	4	65906	H21P085	
210	C71N650D	2126	OUTPUT INST. - C71N050D	AB	4	65906	H21P087	
211	C71N651	3514	OUTPUT INST. - C71N050A	AB	4	65906	H21P084	
212	C71N653	3515	OUTPUT INST. - C71N050A	AB	4	65906	H21P084	
213	E1100F001A	828	RHR HX THERMAL RELIEF VA	RB	2	629 6		RHRHX-03
214	E1100F001B	824	RHR HX THERMAL RELIEF VA	RB	2	629 6		RHRHX-02
215	E1100F025A	5039	RELIEF VALVE	RB	B	576 0		RBTR-02
216	E1100F025B	5143	RELIEF VALVE	RB	B	57611		RBTR-02
217	E1100F029	6031	SDC SUCTION RELIEF VALVE	RB	B	579 0	PIPE	RBTR-07
218	E1100F030A	5001	RELIEF VALVE	RB	SB	546 0		RBTR-15
219	E1100F030B	5105	RELIEF VALVE	RB	SB	547 0		RBTR-15
220	E1100F030C	5074	RELIEF VALVE	RB	SB	546 0		RBTR-15
221	E1100F030D	5177	RELIEF VALVE	RB	SB	546 0		RBTR-15
222	E1100F050A	5061	TESTABLE CHECK VALVE	DW	1	59906		DW-02
223	E1100F050B	5168	TESTABLE CHECK VALVE	DW	1	59906		DW-02
224	E1100F056A	829	RHR HX RELIEF VALVE	RB	1	608 0		RHRHX-02
225	E1100F056B	823	RHR HX RELIEF VALVE	RB	1	608 0		RHRHX-02
226	E1100F060A	5064	MANUAL ISOLATION VALVE	DW	1	60000		
227	E1100F060B	5170	LPCI LOOP B MANUAL ISO VALVE	DW	1	60000		
228	E1100F078	5141	RHRSW XTIE CHECK VALVE	RB	1	60106		RHRHX-01
229	E1101B001A	711	DIV 1 RHR HX	RB	2	60306		RHRHX-04
230	E1101B001B	764	RHR DIV 2 HEAT EXCHANGER	RB	2	503 6		RHRHX-04
231	E1102C002A	5012	RHR PUMP A	RB	SB	540 0		RBSB-01
232	E1102C002B	5116	RHR PUMP B	RB	SB	540 0		RBSB-01
233	E1102C002C	5083	RHR PUMP C	RB	SB	540 0		RBSB-01
234	E1102C002D	5188	RHR PUMP D	RB	SB	540 0		RBSB-01
235	E1150F004A	5002	TORUS SUCTION VALVE	RB	SB	543 0	V8-2099	RBTR-19
236	E1150F004B	5106	TORUS SUCTION VALVE	RB	SB	543 0	V8-2102	RBTR-19
237	E1150F004C	5075	TORUS SUCTION VALVE	RB	SB	542 4	V8-2101	RBTR-19
238	E1150F004D	5178	TORUS SUCTION MOV	RB	SB	542 4	V8-2100	RBTR-19
239	E1150F006A	5005	SDC ISOLATION VALVE	RB	SB	546 0	V8-2095	RBTR-19
240	E1150F006B	5109	SDC ISOLATION VALVE	RB	SB	546 0	V8-2098	RBTR-19
241	E1150F006C	5077	SDC ISOLATION VALVE	RB	SB	546 0	V8-2097	RBTR-19
242	E1150F006D	5181	SDC ISOLATION VALVE	RB	SB	546 0	V8-2096	RBTR-19
243	E1150F007A	5026	MINIMUM FLOW MOV	RB	B	578 6	V8-2154	RBTR-03
244	E1150F007B	5129	MIN FLOW ISOLATION MOV	RB	B	578 6	V8-2134	RBTR-03
245	E1150F008	6019	SDC SUCTION OUTBD ISO MOV	RB	1	591 0	V8-2092	RB1-04
246	E1150F009	6005	SDC INBD SUCTION ISO MOV	DW	1	600 0	V8-2091	DW-07
247	E1150F015A	5056	LPCI INBD ISOLATION MOV	RB	1	594 4	V8-2161	RB1-17
248	E1150F015B	5163	LPCI LOOP B INBD INJECTION MOV	RB	1	594 4	V8-2162	RB1-17
249	E1150F017A	5052	LPCI OUTBD ISO MOV	RB	B	578 5	V8-2159	RBTR-20
250	E1150F017B	5160	LPCI LOOP B OUTBD INJECTION MO	RB	B	578 5	V8-2160	RBTR-20

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
251	E1150F024A	6207	DIV 1 TC ISOLATION MOV	RB	B	578 7	V8-2135	RBTR-04
252	E1150F024B	6219	DIV 2 TC ISOLATION MOV	RB	B	578 7	V8-2136	RBTR-04
253	E1150F028A	6201	TC DIV 1 ISOLATION MOV	RB	B	578 7	V8-2155	RBTR-05
254	E1150F028B	6213	TC DIV 2 ISOLATION MOV	RB	B	578 7	V8-2156	RBTR-05
255	E1150F048A	5036	DIV I RHR HX BYPASS	RB	1	590 0	V8-2139	RHRHX-06
256	E1150F048B	5139	DIV II RHR HX BYPASS	RB	1	60511	V8-2140	RHRHX-06
257	E1150F068A	716	DIV 1 RHRSW HX FLOW CONTROL MO	RB	2	617 3	V15-2018	RHRHX-05
258	E1150F068B	765	DIV 2 RHRSW HX FLOW CONTROL MO	RB	2	617 3	V15-2019	RHRHX-05
259	E1150F608	6012	F009 BYPASS MOV	DW	1	608 0	V8-3407	DW-08
260	E1150F611A	5066	F017A BYPASS VALVE	RB	B	570 0	V8-4613	RBTR-01
261	E1150F611B	5171	F017B BYPASS MOV	RB	B	570 0	V8-4614	RBTR-01
262	E1151C001A	700	RHRSW PUMP A	RHR	1	590 0	PUMP	RHR1-12
263	E1151C001B	742	RHRSW PUMP B	RHR	1	590 0	PUMP	RHR1-12
264	E1151C001C	705	RHRSW PUMP C	RHR	1	590 0	PUMP	RHR1-12
265	E1151C001D	746	RHRSW PUMP D	RHR	1	590 0	PUMP	RHR1-12
266	E1156C001A	725	DIV 1 RHRSW FAN MOTOR	RHR	2	617 0		RHR2-09
267	E1156C001B	773	DIV 2 RHRSW FAN MOTOR	RHR	2	617 0		RHR2-09
268	E1156C001C	731	DIV 1 RHRSW FAN MOTOR	RHR	2	617 0		RHR2-09
269	E1156C001D	778	DIV 2 RHRSW FAN MOTOR	RHR	2	617 0		RHR2-09
270	E11F412	3139	SOLENOID VALVE	RB	2	61306		RB2-19
271	E11F413	3145	SOLENOID VALVE	RB	2	61306		RB2-19
272	E11F414	3127	SOLENOID VALVE	RB	2	62805		RB2-19
273	E11F415	3133	SOLENOID VALVE	RB	2	62805		RB2-19
274	E11F610A	5062	CHECK VALVE BYPASS	DW	1	59906		DW-03
275	E11F610B	5167	STEAM WARMUP BYPASS VALVE	DW	1	59906		DW-03
276	E11K600A	1966	INTERM. INST. - E11R603A	AB	2	61306	H11P613	
277	E11K600B	1967	INTERM. INST. - E11R603B	AB	2	61306	H11P612	
278	E11K603A	737	POWER SUPPLY	AB	2	61306	H11P613	
279	E11K603B	759	POWER SUPPLY	AB	2	61306	H11P612	
280	E11K817A	5317	INTERM. INST. - PT #72	AB	2	61306	H11P613	
281	E11K817B	5318	INTERM. INST. - PT #73	AB	2	61306	H11P612	
282	E11K826A	5321	INTERM. INST. - PT #80	AB	2	61306	H11P613	
283	E11K826B	5322	INTERM. INST. - PT #81	AB	2	61306	H11P612	
284	E11N007A	738	FLOW TRANSMITTER	RB	B	56200	H21P018	
285	E11N007B	760	FLOW TRANSMITTER	RB	B	56200	H21P021	
286	E11N015A	1970	SOURCE INST. - E11R608A	RB	B	56200	H21P018	
287	E11N015B	1971	SOURCE INST. - E11R608B	RB	B	56200	H21P021	
288	E11N055A	5031	PRESSURE TRANSMITTER	RB	B	56200	H21P018	
289	E11N055B	5135	PRESSURE TRANSMITTER	RB	B	56200	H21P021	
290	E11N055C	5100	PRESSURE TRANSMITTER	RB	B	56200	H21P018	
291	E11N055D	5203	PRESSURE TRANSMITTER	RB	B	56200	H21P021	
292	E11N655A	8115	RHR PMP.A PRMSSVE TO ADS TRIP	AB	4	659-6	H21P080	
293	E11N655B	8116	RHR PMP.B BLDN.PR.TRIP UNIT	AB	4	659-6	H21P081	
294	E11N655C	8117	RHR PMP.C BLDN.PR.TRIP UNIT	AB	4	659-6	H21P080	
295	E11N655D	8118	RHR PMP.D BLDN.PR.TRIP UNIT	AB	4	659-6	H21P081	
296	E11P400A	1832	RELAY PANEL	RHR	1	59000	RACKS	RHR1-13
297	E11P400B	1833	RELAY PANEL	RHR	1	59000	RACKS	RHR1-13
298	E11R003A	5029	PRESSURE INDICATOR	RB	B	56200	H21P018	
299	E11R003B	5133	PRESSURE INDICATOR	RB	B	56200	H21P021	
300	E11R003C	5098	PRESSURE INDICATOR	RB	B	56200	H21P018	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
301	E11R003D	5201	PRESSURE INDICATOR	RB	B	56200	H21P021	
302	E11R602A	710	FLOW INDICATOR	AB	3	64306	H11P601	
303	E11R602B	758	FLOW INDICATOR	AB	3	64306	H11P602	
304	E11R603A	5044	FLOW INDICATOR	RB	3	64306	H11P601	
305	E11R603B	5147	FLOW INDICATOR	RB	3	64306	H11P602	
306	E11R608A	5043	FLOW RECORDER	AB	3	64306	H11P601	
307	E11R608B	5146	FLOW RECORDER	AB	3	64306	H11P602	
308	E2150F031A	7057	CS MIN FLOW MOV	RB	B	568 0	V8-4683	RBB-05
309	E2150F031B	7049	CS MIN FLOW MOV	RB	B	56600	V8-2032	RBB-05
310	E21K601A	1972	FEED - B21N094A	AB	2	61306	H11P626	
311	E21K601B	1973	FEED - B21N094B	AB	2	61306	H11P627	
312	E4100B001	4020	BAROMETRIC CONDENSER	AB	SB	540 0		HPCI-08
313	E4100B002	4022	LUBE OIL COOLER	AB	SB	540 0		HPCI-13
314	E4100F020	4010	BOOSTER PUMP RELIEF VA	AB	SB	540 0		HPCI-01
315	E4100F026	4028	DRAIN LINE ISO VA	AB	SB	54107	VALVE	HPCI-03
316	E4100F028	4092	AOV	RB	SB	54200		HPCI-26
317	E4100F050	4021	LUBE OIL COOLER RELIEF VA	AB	SB	540 0		HPCI-01
318	E4100F053	4131	AOV	AB	SB	54200		HPCI-14
319	E4100F067	4109	H.O. STOP VALVES	AB	SB			HPCI-09
320	E4100F068	4110	H.O. CONTROL VALVE	AB	SB			HPCI-09
321	E4101C001A	4038	MAIN HPCI PUMP	AB	SB	540 0		HPCI-02
322	E4101C001B	4006	HPCI BOOSTER PUMP	AB	SB	545 0		HPCI-15
323	E4101C001C	8006	TURBINE-DRIVEN OIL PUMP	AB	SB	545 0	FLOOR	HPCI-23
324	E4101C001D	8007	HPCI PUMP GEAR REDUCER	AB	SB	545 0	FLOOR	
325	E4101C002	4113	HPCI TURBINE	AB	SB	545 0		HPCI-16
326	E4101C003	4036	VACUUM PUMP	AB	SB	540 0		HPCI-17
327	E4101C004	4025	COND PUMP	AB	SB	540 0		HPCI-18
328	E4101C005	8008	HPCI AUX OIL PUMP	AB	SB	540 0		HPCI-22
329	E4150F001	4108	MOV	AB	SB	549 9	V17-2022	HPCI-10
330	E4150F002	4074	STEAM SUPPLY INBD ISO	DW	1	586 6	V17-2020	DW-22
331	E4150F003	4076	STEAM SUPPLY OUTBD ISO	RB	1	586 6	V17-2021	STNL-03
332	E4150F004	8129	HPCI BSTR.PMP.SCTN.FRM.CST ISO	RB		541-1	V8-2191	HPCI-27
333	E4150F006	4060	MOV ISOLATION TO FW	RB	1	587 3	V8-2194	STNL-06
334	E4150F012	4053	MOV ISOLATION TO TORUS	RB	SB	555 6	V8-2196	HPCI-24
335	E4150F041	4004	HPCI SUCTION FROM TORUS MOV	AB	SB	54111	V8-2204	HPCI-05
336	E4150F042	4001	TORUS SUCTION MOV	RB	SB	54111	V8-2202	RBTR-18
337	E4150F059	4013	ISO VA BAROMETRIC CONDENSER	AB	SB	55000	V8-2218	HPCI-04
338	E4150F075	4177	TURBINE EXH OUTBD VAC BREAKER	RB	B	579 0	V11-2013	RBTR-13
339	E41F025	4068	SOLENOID VA	AB	SB	54400	H21P428	
340	E41F026	4067	SOLENOID VA	AB	SB	54400	H21P420	
341	E41F035	4015	PCV - BAROMETRIC COND	AB	SB	54103	PIPE	HPCI-06
342	E41F053	4132	SOLENOID FOR F053	AB	SB	54400	H21P420	
343	E41F200	8130	HPCI REMOTE TURB.TRIP SOL.VALV	RB		545-0	EQUIP.	HPCI-29
344	E41F428	4093	SOLDNOID FOR F028	AB	SB	54400	H21P428	
345	E41F429	4096	SOLENOID FOR F029	AB	SB	54400	H21P420	
346	E41F454	4088	SOLENOID FOR F054	AB	SB	54400	H21P420	
347	E41K200	1944	INTERM. INST. - E41R700	AB	2	61306	H11P929	
348	E41K201	3901	SOURCE INST. - E41R700	AB	2	61306	H11P929	
349	E41K202	8132	HPCI TURB.SPEED SIGNAL CONDITI	RB	2	613-6	H11P929	
350	E41K203	3902	SOURCE INST. - E41R700	AB	SB	54100	EQUIP	HPCI-20

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
351	E41K204	3903	SOURCE INST. - E41R700	AB	2	61306	H11P929	
352	E41K400	4066	PRESSURE CONTROL	RB	SB	54700	WALL	HPCI-28
353	E41K401	8523	HPCI TRS.H2O.LVL.SIG.COND	AB	4	659-6	H21P081	
354	E41K403	8524	HPCI TRS.H2O.LVL.SIG.COND	AB	4	659-6	H21P081	
355	E41K409	8525	HPCI TURB.SPD.SIG.COND.	RB	2	613-6	H11P929	
356	E41K411	8526	HPCI TURB.GOV.SIG.COND.	RB	2	613-6	H11P929	
357	E41K600	1940	PS - E41N014, 16, 09, 19	AB	2	61306	H11P612	
358	E41K601	1939	INTERM. INST. - E41R613	AB	2	61306	H11P612	
359	E41K603	3904	FEED - E41R613	AB	2	61306	H11P612	
360	E41K615	1942	INTERM. INST. - E41R614	AB	2	61306	H11P612	
361	E41K616	1943	PS - E41K615	AB	2	61306	H11P612	
362	E41K801	8527	HPCI PMP.FLOW.SIG.COND.	RB	2	613-6	H11P612	
363	E41K803	8691	HPCI TURB.SPD.SIG.COND.	RB	2	613-6	H11P612	
364	E41K805	8528	HPCI PMP.FL.RTE.SIG.COND.	RB	2	613-6	H11P612	
365	E41N006	4047	FLOW SWITCH	AB	SB	54000	H21P014	
366	E41N008	1941	SOURCE INST. - E41R613	AB	SB	54000	H21P014	
367	E41N009	1937	SOURCE INST. - E41R609	AB	SB	54000	H21P014	
368	E41N010	4065	PRESSURE SWITCH	AB	SB	54000	H21P014	
369	E41N013	1935	SOURCE INST. - E41R608	AB	SB	54000	H21P014	
370	E41N016	1936	SOURCE INST. - E41R608	AB	SB	54000	H21P014	
371	E41N017A	4118	PRESSURE SWITCH	AB	SB	54000	H21P014	
372	E41N017B	4119	PRESSURE SWITCH	AB	SB	54000	H21P014	
373	E41N019	1938	SOURCE INST. - E41R609	RB	SB	54000	H21P014	
374	E41N027	4041	PRESSURE SWITCH	RB	SB	54000	H21P014	
375	E41N030A	3910	SOURCE INST. - E41N602A	AB	SB	55100	CEILING	HPCI-07
376	E41N030B	3909	SOURCE INST. - E41N602B	AB	SB	55100	CEILING	HPCI-07
377	E41N055A	3935	SOURCE INST. - E41N655A	RB	B	56600	H21P034	
378	E41N055B	3936	SOURCE INST. - E41N655B	AB	SB	54000	H21P014	
379	E41N055C	3937	SOURCE INST. - E41N655C	RB	B	56600	H21P034	
380	E41N055D	3938	SOURCE INST. - E41N655D	AB	SB	54000	H21P014	
381	E41N057A	3912	SOURCE INST. - E41N657A	RB	B	56200	H21P016	
382	E41N057B	3913	SOURCE INST. - E41N657B	RB	B	56200	H21P036	
383	E41N058A	3931	SOURCE INST. - E41N658A	RB	B	56200	H21P016	
384	E41N058B	3932	SOURCE INST. - E41N658B	RB	B	56200	H21P036	
385	E41N058C	3933	SOURCE INST. - E41N658C	RB	B	56200	H21P016	
386	E41N058D	3934	SOURCE INST. - E41N658D	RB	B	56200	H21P036	
387	E41N061B	3939	SOURCE INST. - E41N661B	YD	1	58705	H21P492	
388	E41N061D	3940	SOURCE INST. - E41N661D	YD	1	58705	H21P492	
389	E41N062B	3941	SOURCE INST. - E41N662B	AB	SB	54904	WALL	RBSB-07
390	E41N062D	3944	SOURCE INST. - E41N662D	AB	SB	54904	WALL	RBSB-07
391	E41N203	3908	SOURCE INST. - E41R700	AB	SB	54910	EQUIP	HPCI-21
392	E41N212	3905	SOURCE INST. - E41R700	AB	SB	54400	EQUIP	
393	E41N500A	8529	HPCI STOP VLV.POS.SWITCH	RB		549-1	V17-2026	
394	E41N500B	8530	HPCI STOP VLV.POS.SWITCH	RB		549-1	V17-2026	
395	E41N602A	3915	OUTPUT INST. - E41N030A	AB	2	61306	H11P614	
396	E41N602B	3916	OUTPUT INST. - E41N030B	AB	2	61306	H11P614	
397	E41N655A	3924	OUTPUT INST. - E41N055A	AB	4	65906	H21P080	
398	E41N655B	3925	OUTPUT INST. - E41N055B	AB	4	65906	H21P081	
399	E41N655C	3926	OUTPUT INST. - E41N055C	AB	4	65906	H21P080	
400	E41N655D	3927	OUTPUT INST. - E41N055D	AB	4	65906	H21P081	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
401	E41N657A	3911	SOURCE INST. - PT #130	AB	4	65906	H21P080	
402	E41N657B	3917	OUTPUT INST. - E41N057B	AB	4	65906	H21P081	
403	E41N658A	3920	OUTPUT INST. - E41N058A	AB	4	65906	H21P080	
404	E41N658B	3921	OUTPUT INST. - E41N058B	AB	4	65906	H21P081	
405	E41N658C	3922	OUTPUT INST. - E41N058C	AB	4	65906	H21P080	
406	E41N658D	3923	OUTPUT INST. - E41N058D	AB	4	65906	H21P081	
407	E41N660A	3918	OUTPUT INST. - E41N057A	AB	4	65906	H21P080	
408	E41N660B	3919	OUTPUT INST. - E41N057B	AB	4	65906	H21P081	
409	E41N661B	3928	OUTPUT INST. - E41N061B	AB	4	65906	H21P081	
410	E41N661D	3929	OUTPUT INST. - E41N061D	AB	4	65906	H21P081	
411	E41N662B	3930	OUTPUT INST. - E41N062B	AB	4	65906	H21P081	
412	E41N662D	3943	OUTPUT INST. - E41N062D	AB	4	65906	H21P081	
413	E41NA01	8131	HPCI AUX./MAIN OIL PMP.TRIP PR	RB		546-0	EQUIP.	
414	E41R608	4080	PRESSURE INDICATOR	AB	3	64306	H11P602	
415	E41R609	4043	PRESSURE INDICATOR	AB	3	64306	H11P602	
416	E41R613	4049	FLOW INDICATOR	AB	3	64306	H11P602	
417	E41R614	4048	FLOW CONTROL	AB	3	64306	H11P602	
418	E41R700	3229	HPCI TURBINE SPEED INDICATION	RB	3	64306	H11P602	
419	E5100B001	4246	RCIC BAROMETRIC CONDENSER	RB	SB	540 0		RCIC-05
420	E5100B002	4242	RCIC LUBE OIL COOLER	RB	SB			RCIC-06
421	E5100F017	4212	RELIEF VALVE	RB	SB	54500		RCIC-04
422	E5100F018	4241	RELIEF VALVE	RB	SB	540 0		RCIC-01
423	E5101C001	4224	RCIC PUMP	RB	SB	543 0		RCIC-07
424	E5101C002	4295	RCIC TURBINE	RB	SB	543 0		RCIC-08
425	E5101C003	4318	RCIC CONDENSER PUMP	RB	SB	540 0		RCIC-09
426	E5101C004	4325	RCIC VACUUM PUMP	RB	SB	540 0		RCIC-10
427	E5150F007	4277	RCIC STEAM INBD ISO VA	DW	1	58610	V17-2030	DW-21
428	E5150F008	4280	RCIC STEAM OUTBD ISO VA	RB	1	58610	V17-2031	STNL-07
429	E5150F013	4274	RCIC PUMP DISCH INBD ISO VA	RB	1	586 6	V8-2228	STNL-04
430	E5150F019	4254	MINIMUM FLOW MOV	RB	B	578 6	V8-2230	RBTR-21
431	E5150F025	8531	RCIC ISO.VALVE	RB		541-3	PIPE	RCIC-20
432	E5150F029	4206	MOV	RB	SB	541 6		RCIC-16
433	E5150F031	4201	TORUS ISO MOV	RB	SB	545 0	V8-2225	RBTR-17
434	E5150F044	4294	RCIC TURBINE GOVERNING VA	RB	SB	59311		RCIC-21
435	E5150F045	4286	RCIC TURBINE ST INLET VA	RB	SB	544 0	V17-2032	RCIC-11
436	E5150F046	4229	RCIC TURBINE CW SUPPLY VA	RB	SB	547 0	V8-2239	RCIC-12
437	E5150F059	4290	RCIC TURBINE THROTTLE VA	RB	SB	544 0		RCIC-19
438	E5150F062	4312	RCIC VACUUM BREAKER ISO VA	RB	B	578 0	V11-2020	RBTR-12
439	E51F004	4324	RCIC CONDENSER PUMP DISCHARGE	RB	SB	54400	H21P485	RCIC-02
440	E51F015	4232	PCV	RB	SB	54606		RCIC-13
441	E51F025	8532	RCIC SOLENOID VALVE	RB		544-0	H21P485	RCIC-22
442	E51K200	1958	INTERM. INST. - E51R700	AB	2	61700	H11P923	
443	E51K201	1959	SOURCE INST. - E51R700	AB	2	61700	H11P923	
444	E51K203	8533	RCIC HYD.ACT.SP.SIG.COND.	RB		549-1		
445	E51K204	8534	RCIC TURB.SPEED CONTR.	AB	2	617-0	H11P923	
446	E51K400	4233	PRESSURE CONTROLLER	RB	SB	54406		
447	E51K409	8535	RCIC TURB.SP.SIG.COND.	AB	2	613-6	H11P923	
448	E51K411	8536	RCIC TURB.SP.SIG.COND.	AB	2	613-6	H11P923	
449	E51K600	1955	PS - E51N003, 4, 8, 9	AB	2	61306	H11P613	
450	E51K601	1953	INTERM. INST. - E51R613	AB	2	61306	H11P613	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
451	E51K603	1954	PS - E51K601	AB	2	61306	H11P613	
452	E51K615	1956	INTERM. INST. E51R614	AB	2	61306	H11P613	
453	E51K616	1957	PS - E51K615	AB	2	61306	H11P613	
454	E51K801	8537	RCICPMP.DSCH.FL.SIGCOND	AB	2	613-6	H11P613	
455	E51K803	8538	RCIC TURB.SP.SIG.COND.	AB	2	613-6	H11P613	
456	E51K805	8539	RCIC PMP.FL.SIG.COND.	AB	2	613-6	H11P613	
457	E51N002	4259	FLOW SWITCH	RB	SB	54000	H21P017	
458	E51N003	1952	SOURCE INST. - E51R613	RB	SB	54000	H21P017	
459	E51N004	1951	SOURCE INST. - E51R609	RB	SB	54000	H21P017	
460	E51N005	3503	SOURCE INST. - E51R609	RB	SB	54000	H21P017	
461	E51N006	4218	PRESSURE SWITCH	RB	SB	54000	H21P017	
462	E51N007	1949	SOURCE INST. - E51R608	RB	SB	54000	H21P017	
463	E51N008	1950	SOURCE INST. - E51R608	RB	SB	54000	H21P017	
464	E51N009A	4303	PRESSURE SWITCH	RB	SB	54000	H21P017	
465	E51N009B	4304	PRESSURE SWITCH	RB	SB	54000	H21P017	
466	E51N020	4250	PRESSURE SWITCH	RB	SB	54000	H21P017	
467	E51N023A	8112	STEAM LEAK THERMOCOUPLE	RB	SB	55000	CEILING	RCIC-18
468	E51N023B	8113	STEAM LEAK THERMOCOUPLE	RB	SB	55000	CEILING	RCIC-18
469	E51N055A	3550	SOURCE INST. - E51N655A	RB	SB	54000	H21P017	
470	E51N055B	3551	SOURCE INST. - E51N655B	RB	B	56600	H21P037	
471	E51N055C	3552	SOURCE INST. - E51N655C	RB	SB	54000	H21P017	
472	E51N055D	3553	SOURCE INST. - E51N655D	RB	B	56600	H21P037	
473	E51N057A	3554	SOURCE INST. - E51N657A	RB	1	58706	H21P035	
474	E51N057B	3555	SOURCE INST. - E51N657B	RB	B	56600	H21P038	
475	E51N058A	3558	SOURCE INST. - E51N658A	RB	1	58706	H21P035	
476	E51N058B	3559	SOURCE INST. - E51N658B	RB	B	56600	H21P038	
477	E51N058C	3560	SOURCE INST. - E51N658C	RB	1	58706	H21P035	
478	E51N058D	3561	SOURCE INST. - E51N658D	RB	B	56600	H21P038	
479	E51N205	8540	RCIC TURB.SP.SENS.ELEM.	RB		549-1	EQUIP	
480	E51N512	8541	RCIC VALVE POS.SWITCH	RB		543-1	V17-2023	
481	E51N602A	8110	STEAM LEAK TEMP SWITCH	AB	2	61306	H11P614	
482	E51N602B	8111	STEAM LEAK TEMP SWITCH	AB	2	61306	H11P614	
483	E51N655A	3542	OUTPUT INST. - E51N055A	AB	4	65906	H21P080	
484	E51N655B	3543	OUTPUT INST. - E51N055B	AB	4	65906	H21P081	
485	E51N655C	3544	OUTPUT INST. - E51N055C	AB	4	65906	H21P080	
486	E51N655D	3545	OUTPUT INST. - E51N055D	AB	4	65906	H21P081	
487	E51N657A	3546	OUTPUT INST. - E51N057A	AB	4	65906	H21P080	
488	E51N657B	3547	OUTPUT INST. - E51N057B	AB	4	65906	H21P081	
489	E51N658A	3566	OUTPUT INST. - E51N058A	AB	4	65906	H21P080	
490	E51N658B	3567	OUTPUT INST. - E51N058B	AB	4	65906	H21P081	
491	E51N658C	3568	OUTPUT INST. - E51N058C	AB	4	65906	H21P080	
492	E51N658D	3569	OUTPUT INST. - E51N058D	AB	4	65906	H21P081	
493	E51N660A	3548	OUTPUT INST. - E51N057A	AB	4	65906	H21P080	
494	E51N660B	3549	OUTPUT INST. - E51N057B	AB	4	65906	H21P081	
495	E51R608	4285	PRESSURE INDICATOR	AB	3	64306	H11P601	
496	E51R609	4215	PRESSURE INDICATOR	AB	3	64306	H11P601	
497	E51R613	4260	FLOW INDICATOR	AB	3	64306	H11P601	
498	E51R614	4261	FLOW CONTROL	AB	3	64306	H11P601	
499	G1154F018	7018	DW SUMP INBD ISOLATION MOV	DW	B	578 3	V9-2022	DW-15
500	G1154F600	7017	DW SUMP INBD ISOLATION MOV	DW	B	57910	V9-2044	DW-16

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
501	G3352F001	7027	RWCU FROM RR INBD ISO MOV	DW	1	60210	V8-2252	DW-05
502	G3352F004	7028	RWCU FROM RR OUTBD ISO MOV	RB	2	623 7	V8-2253	RWCU-01
503	G3352F220	7010	RWCU OUTBD ISOLATION MOV	RB	1	586 6	V8-4615	STNL-05
504	G5100F600	7044	TORUS WATER MANAGEMENT ISO MOV	RB	SB	540 7		RBTR-16
505	G5100F601	7043	TORUS WATER MANAGEMENT ISO MOV	RB	SB	540 7		RBTR-16
506	G5100F602	7045	TORUS WATER MANAGEMENT ISO MOV	RB	SB	540 7	V8-3831	RBTR-16
507	G5100F603	7046	TORUS WATER MANAGEMENT ISO MOV	RB	SB	540 7	V8-3833	RBTR-16
508	G5100F604	7041	TORUS WATER MANAGEMENT ISO VA	RB	B	577 4	V8-3849	RBTR-09
509	G5100F605	7040	TORUS WATER MANAGEMENT ISO VA	RB	B	577 4	V8-4680	RBTR-09
510	G5100F606	7050	TORUS WATER MANAGEMENT ISO MOV	RB	B	57011	V8-3850	RBTR-14
511	G5100F607	7051	TORUS WATER MANAGEMENT ISO MOV	RB	B	57011	V8-3848	RBTR-14
512	G51P400A	1869	RELAY PANEL	RB	1	58310	G51P400A	RB1-24
513	G51P400B	1870	RELAY PANEL	RB	1	58310	G51P400B	RB1-24
514	H11P601	739	CONTROL ROOM PANEL	AB	3	64306	FLOOR	CR-02
515	H11P602	761	CONTROL ROOM PANEL	AB	3	64306	FLOOR	CR-02
516	H11P609	1834	RELAY PANEL	AB	2	61306	FLOOR	RR-13
517	H11P611	1835	RELAY PANEL	AB	2	61306	FLOOR	RR-13
518	H11P612	3226	INSTRUMENT RACK	AB	2	61306	FLOOR	RR-03
519	H11P613	3227	INSTRUMENT RACK	AB	2	61306	FLOOR	RR-03
520	H11P614	1836	RELAY PANEL	AB	2	61306	FLOOR	RR-04
521	H11P617	5084	HGA/ - RELAY PANEL	AB	2	61306	FLOOR	RR-06
522	H11P618	5189	HGA - RELAY PANEL	AB	2	61306	FLOOR	RR-06
523	H11P620	1838	RELAY PANEL	AB	2	61306	FLOOR	RR-08
524	H11P621	1839	RELAY PANEL	AB	2	61306	FLOOR	RR-14
525	H11P622	1840	RELAY PANEL	AB	2	61306	FLOOR	RR-15
526	H11P623	1841	RELAY PANEL	AB	2	61306	FLOOR	RR-09
527	H11P626	1871	RELAY PANEL	AB	2	61306	FLOOR	RR-10
528	H11P627	1872	RELAY PANEL	AB	2	61306		RR-10
529	H11P628	3036	RELAY PANEL	AB	2	61306	FLOOR	RR-16
530	H11P809	1544	HAS SWITCH FOR R3001S001	AB	3	64306	FLOOR	CR-03
531	H11P810	1543	HAS SWITCH FOR R3001S004	AB	3	64306	FLOOR	CR-03
532	H11P857	373	RELAY PANEL	AB	2	61306	FLOOR	RR-11
533	H11P870	374	RELAY PANEL	AB	2	61306	FLOOR	RR-11
534	H11P898A	1842	RELAY PANEL	AB	2	61306	FLOOR	RR-17
535	H11P898B	1843	RELAY PANEL	AB	2	61306	FLOOR	RR-17
536	H11P900	1879	INTERM. PS - E11N027A	AB	2	61806	WALL	RR-01
537	H11P901	8123	120VAC DISTRIBUTION PANEL	AB	2	618-6	WALL	RR-19
538	H11P902	8124	120VAC DISTRIBUTION PANEL	AB	2	618-6	WALL	RR-19
539	H11P903	1880	OUTPUT PS - B21R803	AB	2	61806	WALL	RR-01
540	H11P914	1171	INSTRUMENT RACK	AB	2	61306	FLOOR	RR-18
541	H11P915	1176	INSTRUMENT RACK	AB	2	61306	FLOOR	RR-18
542	H11P917A	3626	INSTRUMENT RACK	AB	2	61306	FLOOR	RR-05
543	H11P917B	3627	INSTRUMENT RACK	AB	2	61306	FLOOR	RR-05
544	H11P923	3630	INSTRUMENT RACK	AB	2	61806	WALL	RR-07
545	H11P929	3631	INSTRUMENT RACK	AB	2	61806	WALL	RR-07
546	H21P004	3218	INSTRUMENT RACK	RB	2	61306	FLOOR	RB2-17
547	H21P005	3219	INSTRUMENT RACK	RB	2	61306	FLOOR	RB2-17
548	H21P006	3632	INSTRUMENT RACK	RB	B	56200	FLOOR	RBB-03
549	H21P009	3220	INSTRUMENT RACK	RB	1	58306	FLOOR	RB1-16
550	H21P010	3221	INSTRUMENT RACK	RB	1	58306	FLOOR	RB1-16

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
551	H21P014	3633	INSTRUMENT RACK	AB	SB	54000	FLOOR	HPCI-12
552	H21P016	3634	INSTRUMENT RACK	RB	B	56200	FLOOR	RBB-03
553	H21P017	3635	INSTRUMENT RACK	RB	SB	54000	FLOOR	RCIC-15
554	H21P018	741	INSTRUMENT RACK	RB	B	56200	FLOOR	RBB-03
555	H21P021	763	INSTRUMENT RACK	RB	B	56200	FLOOR	RBB-02
556	H21P022	3636	INSTRUMENT RACK	RB	B	56200	FLOOR	RBB-02
557	H21P034	3637	INSTRUMENT RACK	RB	B	56600	WALL	RBB-04
558	H21P035	3638	INSTRUMENT RACK	RB	1	58706	WALL	RB1-13
559	H21P036	3639	INSTRUMENT RACK	RB	B	56200	FLOOR	RBB-02
560	H21P037	3640	INSTRUMENT RACK	RB	B	56600	WALL	RBB-04
561	H21P038	3641	INSTRUMENT RACK	RB	B	56600	WALL	RBB-04
562	H21P080	3222	INSTRUMENT RACK	AB	4	65906	FLOOR	AB4-01
563	H21P081	3223	INSTRUMENT RACK	AB	4	65906	FLOOR	AB4-01
564	H21P082	3224	INSTRUMENT RACK	AB	4	65906	FLOOR	AB4-01
565	H21P083	3225	INSTRUMENT RACK	AB	4	65906	FLOOR	AB4-01
566	H21P084	3642	INSTRUMENT RACK	AB	4	65906	FLOOR	AB4-01
567	H21P085	3643	INSTRUMENT RACK	AB	4	65906	FLOOR	AB4-01
568	H21P086	3644	INSTRUMENT RACK	AB	4	65906	FLOOR	AB4-01
569	H21P087	3645	INSTRUMENT RACK	AB	4	65906	FLOOR	AB4-01
570	H21P090-1	1844	RELAY PANEL	AB	3	64706	WALL	BAT-06
571	H21P090-2	1845	RELAY PANEL	AB	3	64706	WALL	BAT-06
572	H21P100	8542	REMOTE RPV SHTDWN.PNL.	AB	2	613-6	FLOOR	SGR1-09
573	H21P285A	8543	CCHVAC AC CHILLER PANEL	AB	5	677-6	FLOOR	AB5-23
574	H21P285B	8544	CCHVAC AC CHILLER PANEL	AB	5	677-6	FLOOR	AB5-23
575	H21P296A	1134	INSTRUMENT RACK	AB	5	67706	FLOOR	AB5-02
576	H21P296B	1369	INSTRUMENT RACK	AB	5	67706	FLOOR	AB5-02
577	H21P296C	1846	RELAY PANEL	AB	5	67706	FLOOR	AB5-02
578	H21P296D	1847	RELAY PANEL	AB	5	67706	FLOOR	AB5-02
579	H21P296E	1132	INSTRUMENT RACK	AB	5	67706	FLOOR	AB5-02
580	H21P296F	1371	INSTRUMENT RACK	AB	5	67706	FLOOR	AB5-02
581	H21P350	1775	X41K002A MOUNTED ON THIS PANEL	RHR	2	61700	FLOOR	RHR2-07
582	H21P351	1848	RELAY PANEL	RHR	2	61700	FLOOR	RHR2-07
583	H21P352	1773	X41K002C/F MOUNTED ON THIS PAN	RHR	2	61700	FLOOR	RHR2-07
584	H21P353	1772	X41K002H MOUNTED ON THIS PANEL	RHR	2	61700	FLOOR	RHR2-07
585	H21P420	1300	INSTRUMENT RACK	AB	SB	54400	WALL	HPCI-11
586	H21P423A	3646	INSTRUMENT RACK	RB	1	58306	FLOOR	RB1-15
587	H21P423B	3647	INSTRUMENT RACK	RB	1	58306	FLOOR	RB1-15
588	H21P428	1301	INSTRUMENT RACK	AB	SB	54400	WALL	HPCI-11
589	H21P474	627	P44N425A MOUNTED ON THIS PANEL	RB	1	58306	H21P474	RB1-15
590	H21P475	626	P44N425B MOUNTED ON THIS PANEL	RB	1	58306	H21P475	RB1-15
591	H21P485	1287	INSTRUMENT RACK	RB	SB	54400	WALL	RCIC-03
592	H21P492	3650	INSTRUMENT RACK	YD		58604	EQUIP	YD-1
593	H21P501A	1079	INSTRUMENT RACK	AB	B	55100	FLOOR	ABB-04
594	H21P501B	1277	INSTRUMENT RACK	AB	B	55100	FLOOR	ABB-04
595	H21P517	1771	X41K003A MOUNTED ON THIS PANEL	RHR	1	59400	WALL	RHR1-11
596	H21P518	1770	X41K003B MOUNTED ON THIS PANEL	RHR	1	59400	WALL	RHR1-11
597	H21P527	1851	RELAY PANEL	AB	5	67706	FLOOR	AB5-04
598	H21P527A	1852	RELAY PANEL	AB	5	68108	WALL	AB5-04
599	H21P528	1853	RELAY PANEL	AB	4	65906	FLOOR	AB4-05
600	H21P529	1854	RELAY PANEL	AB	4	65906	FLOOR	AB4-05

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
601	H21P548	6035	E11N018 MOUNTED ON THIS PANEL	RB	B	56200	FLOOR	RBB-01
602	H21P557	1890	T41 INSTRUMENT RACK	AB	5	68104	WALL	AB5-08
603	H21P558	1891	T41 INSTRUMENT RACK	AB	5	68203	WALL	AB5-08
604	H21P559	8125	120VAC DISTRIBUTION PANEL	RB	2	617-6	WALL	RB2-24
605	H21P560	8126	120VAC DISTRIBUTION PANEL	RB	1	587-6	WALL	RB1-27
606	H21P561	8127	120VAC DISTRIBUTION PANEL	RB		566-3	WALL	RBB-06
607	H21P562	8128	120VAC DISTRIBUTION PANEL	RB		565-1	WALL	RBB-06
608	H21P572	3651	INSTRUMENT RACK	AB	5	68406	FLOOR	AB5-13
609	H21P573	3652	INSTRUMENT RACK	AB	5	68406	FLOOR	AB5-13
610	H21P595A	1092	INSTRUMENT RACK	RB	2	61800	H21P595A	RB2-23
611	H21P595B	1321	INSTRUMENT RACK	RB	2	61800	H21P595B	RB2-23
612	H21P614A	3653	INSTRUMENT RACK	RB	SB	54110	FLOOR	RBSB-03
613	H21P614B	3654	INSTRUMENT RACK	RB	SB	54110	FLOOR	RBSB-03
614	H21P631A	8692	TORUS PR.INSTR.RACK	RB	1	586-4		RB1-26
615	H21P631B	8545	TORUS PR.INSTR.RACK	RB	1	586-4		RB1-26
616	P34F401B	7056	POST ACCIDENT SAMPLE LINE VALV	RB	1	59406	PIPE	RB1-14
617	P4400A001	291	DIV 1 EECW MAKEUP TANK	RB	2	61803		RB2-09
618	P4400A002	355	EECW DIV 2 MAKEUP TANK	RB	2	61803		RB2-09
619	P4400B001	1	EECW DIV I HEAT EXCHANGER	RB	2	613 6		RB2-01
620	P4400B002	316	EECW DIV 2 HEAT EXCHANGER	RB	2	613 6		RB2-01
621	P4400C001A	26	EECW PUMP A	RB	2	613 6	FLOOR	RB2-16
622	P4400C001B	371	EECW DIV 2 PUMP	RB	2	613 6	FLOOR	RB2-16
623	P4400F125A	284	MU TANK RELIEF VALVE	RB	2	62000		RB2-10
624	P4400F125B	343	MU TANK RELIEF VALVE	RB	2	62000		RB2-10
625	P4400F126A	274	MU TANK RELIEF VALVE	RB	2	62000		RB2-11
626	P4400F126B	336	MU TANK RELIEF VALVE	RB	2	62000		RB2-11
627	P4400F142A	2	HX RELIEF VALVE	RB	2	62000		RB2-02
628	P4400F142B	317	EECW HX RELIEF VALVE	RB	2	62000		RB2-02
629	P4400F245A	219	RELIEF VALVE	DW	1	59001		DW-25
630	P4400F245B	604	SUPPLY HEADER RELIEF VA	DW	B	58011		DW-25
631	P4400F601A	258	RBCCW DIV 1 RETURN ISO MOV	RB	1	597 9	V8-2323	RB1-19
632	P4400F601B	499	RBCCW RETURN ISOLATION MOV	RB	1	600 0	V8-2314	RB1-20
633	P4400F602A	12	MAKEUP TANK OUTLET MOV	RB	2	630 9	V8-2407	RB2-03
634	P4400F602B	325	MU TANK OUTLET ISOLATION MOV	RB	2	625 3	V8-2374	RB2-03
635	P4400F603A	55	RBCCW SUPPLY ISO MOV	RB	1	600 0	V8-2324	RB1-19
636	P4400F603B	463	RBCCW SUPPLY ISOLATION MOV	RB	1	600 0	V8-2315	RB1-20
637	P4400F604	508	CRD COOLING ISOLATION MOV	AB	B	569 6	V8-2425	ABB-01
638	P4400F605A	184	RB SUMP HX ISOLATION MOV	RB	SB	545 6	V8-2427	RBSB-04
639	P4400F605B	523	RB SUMP HX SUPPLY ISO MOV	RB	SB	542 0	V8-2426	RBSB-05
640	P4400F606A	192	DRYWELL SUPPLY ISO MOV	RB	B	57800	V8-2485	RBTR-10
641	P4400F606B	553	EECW DIV 20W SUPPLY ISO VA	RB	B	578 0	V8-2484	RBTR-10
642	P4400F607A	254	OUTBD RETURN ISOLATION MOV	RB	B	57806	V8-2486	RBTR-11
643	P4400F607B	617	EECW DIV 2 DW RETURN OUTBD ISO	RB	B	578 6	V8-2483	RBTR-11
644	P4400F608	607	SUMP SUPPLY ISO VALVE	DW	B	580 6	V8-2487	DW-12
645	P4400F613	84	BATTERY CHARGER SUPPLY MOV	AB	3	645 0	P44F613	BAT-07
646	P4400F614	218	PENETRATION COOLING ISO VA	DW	B	582 6	V8-3058	DW-14
647	P4400F616	252	INBD RETURN ISOLATION MOV	DW	B	579 0	V8-3882	DW-13
648	P44F400A	827	EESW TCV	RB	2	61500		RB2-04
649	P44F400B	825	EESW TCV	RB	2	61500		RB2-04
650	P44F402A	282	LEVEL CONTROL VALVE	RB	2	61500		RB2-12

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
651	P44F402B	340	MU TANK LEVEL CONTROL VALVE	RB	2	61500		RB2-12
652	P44F403A	41	RECIRC LINE PRESSURE CONTROL V	RB	2	61500		RB2-05
653	P44F403B	386	PRESSURE DIFFERENTIAL CONTROL	RB	2	61500		RB2-05
654	P44K800A	8546	EECW HX TEMP.CONTROL	AB	3	643-6	H11P601	
655	P44K800B	321	TEMPERATURE ALARM	RB	3	64306	H11P602	
656	P44N401A	7	TEMPERATURE ELEMENT	RB	2	61806		RB2-06
657	P44N401B	622	TEMPERATURE ELEMENT	RB	2	62006		RB2-06
658	P44N425A	122	PRESSURE DIFFERENTIAL SWITCH	RB	1	58306	H21P474	
659	P44N425B	477	PRESSURE DIFFERENTIAL SWITCH	RB	1	58306	H21P475	
660	P44N426A	119	PRESSURE DIFFERENTIAL SWITCH	RB	1	58306	H21P474	
661	P44N426B	474	PRESSURE DIFFERENTIAL SWITCH	RB	1	58306	H21P475	
662	P4500C002A	783	EESW PUMP A	RHR	1	590 0		RHR1-14
663	P4500C002B	785	EESW PUMP B	RHR	1	590 0		RHR1-14
664	P4500F141A	831	EECW HX RELIEF VALVE	RB	2	62000		RB2-02
665	P4500F141B	832	EECW HX RELIEF VALVE	RB	2	62000		RB2-02
666	P45N415A	793	PRESSURE TRANSMITTER	RB	2	61506	WALL	RB2-07
667	P45N415B	795	PRESSURE TRANSMITTER	RB	2	61611	WALL	RB2-07
668	P5000F207A	1013	M/S RELIEF VALVE	AB	SM	551 0		
669	P5000F207B	1189	M/S RELIEF VALVE	AB	SM	551 0		
670	P5000F223A	1066	AIR RECEIVER RELIEF VALVE	AB	SB	540		ABB-07
671	P5000F223B	1258	RELIEF VALVE	AB	SB	540		ABB-07
672	P5000F440	1025	SA SUPPLY TO DIV I NIAS	AB	SM			ABB-05
673	P5000F441	1201	SA SUPPLY TO DIV II ISO VA	AB	SM			ABB-05
674	P5000F541A	1050	RELIEF VALVE	AB	SM			ABB-09
675	P5000F541B	1244	RELIEF VALVE	AB	SM			ABB-09
676	P5000F542A	1049	RELIEF VALVE	AB	SM			ABB-09
677	P5000F542B	1240	RELIEF VALVE	AB	SM			ABB-09
678	P5002A001	1065	NIAS DIV 1 AIR RECEIVER	AB	SB	540		ABB-08
679	P5002A002	1257	AIR RECEIVER	AB	SB	540		ABB-08
680	P5002A004A	1149	VOLUME CHAMBER	RB	1	58800		RB1-01
681	P5002A004B	1310	VOLUME CHAMBER	RB	1	58800		RB1-01
682	P5002B004	146	AFTERCOOLER	AB	SM	551		ABB-10
683	P5002B005	489	AFTERCOOLER	AB	SM	551		ABB-10
684	P5002D001	150	COMPRESSOR	AB	SM	551 0		ABB-14
685	P5002D002	493	COMPRESSOR	AB	SM	551 0		ABB-14
686	P5002D012	1039	NIAS DIV 1 DEHYDRATION UNIT	AB	SM	55100		ABB-11
687	P5002D013	1245	NIAS DIV 2 DEHYDRATION UNIT	AB	SM	55100		ABB-11
688	P5002D014	1036	PREFILTER	AB	SM	55100		ABB-12
689	P5002D015	1228	PRE FILTER	AB	SM	55208		ABB-12
690	P5002D016	1057	NIAS DIV 1 AFTERFILTER	AB	SM	55100		ABB-12
691	P5002D017	1246	AFTER FILTER	AB	SM	55100		ABB-12
692	P5002D029A	1018	DRAIN TANK	AB	SM	55300		ABB-13
693	P5002D029B	1195	DRAIN TANK	AB	SM	55300		ABB-13
694	P50F433A	1006	SOLENOID VALVE	AB	SM	55506	P50P401A	
695	P50F433B	1180	SOLENOID VALVE	AB	SM	56600	P50P401B	
696	P50F511A	1042	SOLENOID VALVE	AB	SM	55800	EQUIP	
697	P50F511B	1233	SOLENOID VALVE	AB	SM	55800	EQUIP	
698	P50F512A	1043	SOLENOID VALVE	AB	SM	55800	EQUIP	
699	P50F512B	1234	SOLENOID VALVE	AB	SM	55800	EQUIP	
700	P50F513A	1044	SOLENOID VALVE	AB	SM	55800	EQUIP	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
701	P50F513B	1235	SOLENOID VALVE	AB	SM	55800	EQUIP	
702	P50F514A	1045	SOLENOID VLAVE	AB	SM	55800	EQUIP	
703	P50F514B	1236	SOLENOID VALVE	AB	SM	55800	EQUIP	
704	P50F515A	1040	DRAIN SOLENOID VALVE	AB	SM	55200	WALL	
705	P50F515B	1231	SOLENOID VALVE	AB	SM	55106	WALL	
706	P50F516A	1026	SOLENOID VALVE	AB	SM	55100	H21P501A	
707	P50F516B	1217	SOLENOID VALVE	AB	SM	55100	H21P501B	
708	P50F518	1273	SOLENOID VALVE	AB	SM	55100	H21P501B	
709	P50F519A	1151	ISOLATION VALVE	RB	1	58800		RB1-28
710	P50F519B	1312	ISOLATION VALVE	RB	1	58800		RB1-28
711	P50N480A	3500	OUTPUT INST. CONTROL AIR	AB	2	61903		SGR1-01
712	P50N480B	3502	OUTPUT INST. CONTROL AIR	AB	2	61806		SGR1-01
713	P50N481A	1004	PRESSURE SWITCH	AB	B	55500	P50P401A	
714	P50N481B	1178	PRESSURE SWITCH	AB	B	56600	P50P401B	
715	P50N482A	1029	PRESSURE SWITCH	AB	B	55100	H21P501A	
716	P50N482B	1219	PRESSURE SWITCH	AB	B	55100	H21P501B	
717	P50N483A	1008	PRESSURE SWITCH	RB	B	55500	P50P401A	
718	P50N483B	1182	PRESSURE SWITCH	RB	B	56600	P50P401B	
719	P50P401A	3655	INSTRUMENT RACK	AB	B	55500	WALL	ABB-02
720	P50P401B	3656	INSTRUMENT RACK	AB	B	56600	WALL	ABB-02
721	P50P402A	1855	RELAY PANEL	AB	B	56600	WALL	ABB-03
722	P50P402B	1856	RELAY PANEL	AB	B	55500	WALL	ABB-03
723	R1400S001B	1486	ESS BUS 64B	AB	2	613 6	BUS 64B	SGR1-05
724	R1400S001C	1490	ESS BUS 64C	AB	2	613 6	BUS 64C	SGR1-05
725	R1400S001E	1494	ESS BUS 65E	AB	3	643 6	BUS 65E	SGR2-03
726	R1400S001F	1498	ESS BUS 65F	AB	3	643 6	BUS 65F	SGR2-03
727	R1400S002A	1484	ESS BUS 11EA	RHR	2	617	BUS 11EA	RHR2-06
728	R1400S002B	1488	ESS BUS 12EB	RHR	2	617 0	BUS 12EB	RHR2-06
729	R1400S002C	1492	ESS BUS 13EC	RHR	2	617 0	BUS 13EC	RHR2-06
730	R1400S002D	1496	ESS BUS 14ED	RHR	2	617 0	BUS 14ED	RHR2-06
731	R1400S020	1495	480V ESS BUS 72E	AB	3	643 6		SGR2-04
732	R1400S020A	1451	72E TRANSFORMER	AB	3	643 6	BUS 72E	SGR2-05
733	R1400S020B	1452	VOLTAGE REGULATOR	AB	3	643 6		SGR2-09
734	R1400S021	1499	480V ESS BUS 72F	AB	3	643 6		SGR2-04
735	R1400S021A	1470	72F TRANSFORMER	AB	3	643 6	BUS 72F	SGR2-05
736	R1400S021B	1471	VOLTAGE REGULATOR	AB	3	643 6	BUS 72F	SGR2-09
737	R1400S022	1487	480V ESS BUS 72B	AB	2	613 6		SGR1-06
738	R1400S022A	1414	72B TRANSFORMER	AB	2	613 6	BUS 72B	SGR1-07
739	R1400S023	1491	ESS BUS 72C	AB	2	613 6	POS 72C-3D	SGR1-06
740	R1400S023A	1433	72C TRANSFORMER	AB	2	613 6	BUS 72C	SGR1-07
741	R1400S036	1485	480V BUS ESS 72EA	RHR	2	617 0		RHR2-04
742	R1400S036A	1413	72EA TRANSFORMER	RHR	2	617 0	BUS 72EA	RHR2-05
743	R1400S037	1489	480V BUS ESS 72EB	RHR	2	617 0		RHR2-04
744	R1400S037A	1432	72EB TRANSFORMER	RHR	2	617 0	BUS 72EB	RHR2-05
745	R1400S038	1493	480V BUS ESS 72EC	RHR	2	617 0		RHR2-04
746	R1400S038A	1449	72EC TRANSFORMER	RHR	2	617 0	BUS 72EC	RHR2-05
747	R1400S038B	1450	VOLTAGE REGULATOR	RHR	2	617 0	BUS 72EC	RHR2-05
748	R1400S039	1497	480V BUS ESS 72ED	RHR	2	617 0		RHR2-04
749	R1400S039A	1468	72ED TRANSFORMER	RHR	2	617 0	BUS 72ED	RHR2-05
750	R1400S039B	1469	VOLTAGE REGULATOR	RHR	2	617 0		RHR2-05

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
751	R1600S002A	1800	MCC 72B-2A	AB	2	620 6	72B2A	SGR1-02
752	R1600S002B	1801	MCC 72B-3A	RB	1	683 6	72B3A	RB1-25
753	R1600S003A	1803	MCC 72C-2A	AB	5	677 6	72C2A	AB5-05
754	R1600S003B	1804	MCC 72C-3A	RB	2	61306	72C3A	RB2-22
755	R1600S003D	1806	MCC 72C-F	RB	2	616 6	72CF	RB2-22
756	R1600S004B	1810	MCC 72E-5A	RB	1	683 6	72E5A	RB1-25
757	R1600S005A	1813	MCC 72F-2A	AB	3	643 6	72F2A	SGR2-08
758	R1600S005C	1815	MCC 72F-4A	RB	2	61306	72F4A	RB2-22
759	R1600S005D	1816	MCC 72F-5A	AB	5	67706	72F5A	AB5-05
760	R1600S016A	1821	MCC 72EA-2C	RHR	2	617 0	72EA2C	RHR2-02
761	R1600S017A	1823	MCC 72EB-2D	RHR	2	617 0	72EB2D	RHR2-02
762	R1600S018A	1824	MCC 72EC-2C	RHR	2	617 0	72EC2C	RHR2-02
763	R1600S019A	1826	MCC 72ED-2D	RHR	2	617 0	72ED2D	RHR2-02
764	R3000A001	1400	FUEL OIL STORAGE TANK	RHR	1	590 0		RHR1-20
765	R3000A002	1415	FUEL OIL STORAGE TANK	RHR	1	590 0		RHR1-20
766	R3000A003	1434	FUEL OIL STORAGE TANK	RHR	1	590 0		RHR1-20
767	R3000A004	1453	FUEL OIL STORAGE TANK	RHR	1	590 0		RHR1-20
768	R3000A005	1421	EXPANSION TANK	RHR	1	60300		RHR1-07
769	R3000A006	1422	EXPANSION TANK	RHR	1	60300		RHR1-07
770	R3000A007	1439	EXPANSION TANK	RHR	1	60300		RHR1-07
771	R3000A008	1458	EXPANSION TANK	RHR	1	60300		RHR1-07
772	R3000A009	1411	EAST AIR RECEIVER	RHR	1	590 0		RHR1-04
773	R3000A010	1428	EAST AIR RECEIVER	RHR	1	590 0		RHR1-04
774	R3000A011	1409	WEST AIR RECEIVER	RHR	1	590 0		RHR1-04
775	R3000A012	1426	WEST AIR RECEIVER	RHR	1	590 0		RHR1-04
776	R3000A013	1446	EAST AIR RECEIVER	RHR	1	590 0		RHR1-04
777	R3000A014	1465	EAST AIR RECEIVER	RHR	1	590 0		RHR1-04
778	R3000A015	1444	WEST AIR RECEIVER	RHR	1	590 0		RHR1-04
779	R3000A016	1463	WEST AIR RECEIVER	RHR	1	590 0		RHR1-04
780	R3000A017	1403	FUEL OIL DAY TANK	RHR	1	590 0		RHR1-21
781	R3000A018	1418	FUEL OIL DAY TANK	RHR	1	590 0		RHR1-21
782	R3000A019	1437	FUEL OIL DAY TANK	RHR	1	590 0		RHR1-21
783	R3000A020	1456	FUEL OIL DAY TANK	RHR	1	590 0		RHR1-21
784	R3000A021	1404	LUBE OIL TANK	RHR	1	590 0		RHR1-22
785	R3000A022	1419	LUBE OIL TANK	RHR	1	590 0		RHR1-22
786	R3000A023	1438	LUBE OIL TANK	RHR	1	590 0		RHR1-22
787	R3000A024	1457	LUBE OIL TANK	RHR	1	590 0		RHR1-22
788	R3000C001	1401	FUEL OIL TRANSFER PUMP	RHR	1	592 0		RHR1-19
789	R3000C002	1416	FUEL OIL TRANSFER PUMP	RHR	1	592 0		RHR1-19
790	R3000C003	1402	FUEL OIL TRANSFER PUMP	RHR	1	592 0		RHR1-19
791	R3000C004	1417	FUEL OIL TRANSFER PUMP	RHR	1	592 0		RHR1-19
792	R3000C009	1435	FUEL OIL TRANSFER PUMP	RHR	1	592 0	EDG13N	RHR1-19
793	R3000C010	2209	F.O. TRANSFER PUMP MOTOR	RHR	1	592 0		RHR1-19
794	R3000C011	2210	F.O. TRANSFER PUMP MOTOR	RHR	1	592 0		RHR1-19
795	R3000C012	2211	F.O. TRANSFER PUMP MOTOR	RHR	1	592 0		RHR1-19
796	R3000D001	1405	STARTING AIR COMPRESSOR	RHR	1	59000		RHR1-05
797	R3000D002	1420	STARTING AIR COMPRESSOR	RHR	1	59000		RHR1-05
798	R3000D003	1440	STARTING AIR COMPRESSOR	RHR	1	59000		RHR1-05
799	R3000D004	1459	STARTING AIR COMPRESSOR	RHR	1	59000		RHR1-05
800	R3000F012A	2505	J.C. TEMP REGULATING VALVE	RHR	1		SKID	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
801	R3000F012B	2506	J.C. TEMP REGULATING VALVE	RHR	1		SKID	
802	R3000F012C	2507	J.C. TEMP REGULATING VALVE	RHR	1		SKID	
803	R3000F012D	2508	J.C. TEMP REGULATING VALVE	RHR	1		SKID	
804	R3000F023A	2509	A.C. TEMP CONTROL VALVE	RHR	1	59300	R3000F023A	RHR1-10
805	R3000F023B	2510	A.C. TEMP CONTROL VALVE	RHR	1	59300	R3000F023B	RHR1-10
806	R3000F023C	2511	A.C. TEMP CONTROL VALVE	RHR	1	59300	R3000F023C	RHR1-10
807	R3000F023D	2512	A.C. TEMP CONTROL VALVE	RHR	1	59300	R3000F023D	RHR1-10
808	R3000F035A	1410	RELIEF VALVE	RHR	1	590 0	R3000A011	RHR1-01
809	R3000F035B	1427	RELIEF VALVE	RHR	1	590 0	R3000A015	RHR1-01
810	R3000F035C	1445	RELIEF VALVE	RHR	1	590 0	R3000A012	RHR1-01
811	R3000F035D	1464	RELIEF VALVE	RHR	1	590 0	R3000A016	RHR1-01
812	R3000F036A	1429	RELIEF VALVE	RHR	1	590 0	R3000A009	RHR1-01
813	R3000F036B	1430	RELIEF VALVE	RHR	1	590 0	R3000A013	RHR1-01
814	R3000F036C	1447	RELIEF VALVE	RHR	1	590 0	R3000A010	RHR1-01
815	R3000F036D	1466	RELIEF VALVE	RHR	1	590 0	R3000A014	RHR1-01
816	R3000F096A	9012	FUEL OIL PUMP DISCHARGE VALVE	RHR	1	590 0		RHR1-26
817	R3000F096B	9013	FUEL OIL PUMP DISCHARGE VALVE	RHR	1	590 0		RHR1-26
818	R3000F096C	9014	FUEL OIL PUMP DISCHARGE VALVE	RHR	1	590 0		RHR1-26
819	R3000F096D	9015	FUEL OIL PUMP DISCHARGE VALVE	RHR	1	590 0		RHR1-26
820	R3000F109A	2497	FLOAT LUBE OIL REGULATOR	RHR	1		SKID	
821	R3000F109B	2498	FLOAT LUBE OIL REGULATOR	RHR	1		SKID	
822	R3000F109C	2499	FLOAT LUBE OIL REGULATOR	RHR	1		SKID	
823	R3000F109D	2500	FLOAT LUBE OIL REGULATOR	RHR	1		SKID	
824	R3000F111A	9016	TURBO CHARGER DISCHARGE VALVE	RHR	1	590 0		RHR1-26
825	R3000F111B	9017	TURBO CHARGER DISCHARGE VALVE	RHR	1	590 0		RHR1-26
826	R3000F111C	9018	TURBO CHARGER DISCHARGE VALVE	RHR	1	590 0		RHR1-26
827	R3000F111D	9019	TURBO CHARGER DISCHARGE VALVE	RHR	1	590 0		RHR1-26
828	R3000F123A	2501	L.O. TEMP REGULATING VALVE	RHR	1		SKID	
829	R3000F123B	2502	L.O. TEMP REGULATING VALVE	RHR	1		SKID	
830	R3000F123C	2503	L.O. TEMP REGULATING VALVE	RHR	1		SKID	
831	R3000F123D	2504	L.O. TEMP REGULATING VALVE	RHR	1		SKID	
832	R3000F601	1480	EMERGENCY DRAIN MOV	RHR	1	592 2		RHR1-23
833	R3000F603	1483	EMERGENCY DRAIN MOV	RHR	1	592 2		RHR1-23
834	R3000F605	1474	EMERGENCY DRAIN MOV	RHR	1	592 2		RHR1-23
835	R3000F607	1477	EMERGENCY DRAIN MOV	RHR	1	592 2		RHR1-23
836	R3000S005	2236	TRANSFORMER, EDG FIELD	RHR	2	617 0	R3000S005	RHR2-01
837	R3000S006	2237	TRANSFORMER, EDG FIELD	RHR	2	617 0	R3000S006	RHR2-01
838	R3000S007	2238	TRANSFORMER, EDG FIELD	RHR	2	617 0	R3000S007	RHR2-01
839	R3000S008	2239	TRANSFORMER, EDG FIELD	RHR	2	617 0	R3000S008	RHR2-01
840	R3000S009A	2216	VOLTAGE REGULATOR	RHR	2	61700	R3000S005	RHR2-01
841	R3000S009B	2217	VOLTAGE REGULATOR	RHR	2	61700	R3000S007	RHR2-01
842	R3000S009C	2218	VOLTAGE REGULATOR	RHR	2	61700	R3000S006	RHR2-01
843	R3000S009D	2219	VOLTAGE REGULATOR	RHR	2	61700	R3000S008	RHR2-01
844	R3001B001	9004	EDG12 LUBE OIL HX	RHR	1	590 0		RHR1-27
845	R3001B002	9005	EDG11 LUBE OIL HX	RHR	1	590 0		RHR1-27
846	R3001B003	9006	EDG13 LUBE OIL HX	RHR	1	590 0		RHR1-27
847	R3001B004	9007	EDG14 LUBE OIL HX	RHR	1	590 0		RHR1-27
848	R3001B017	9000	EDG11 JACKET CCOLANT HX	RHR	1	590 0		RHR1-27
849	R3001B018	9001	EDG13 JACKET CCOLANT HX	RHR	1	590 0		RHR1-27
850	R3001B019	9002	EDG12 JACKET CCOLANT HX	RHR	1	590 0		RHR1-27

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
851	R3001B020	9003	EDG14 JACKET COOLANT HX	RHR	1	590 0		RHR1-27
852	R3001B025	9008	EDG11 AIR COOLANT HX	RHR	1	590 0		RHR1-27
853	R3001B026	9009	EDG13 AIR COOLANT HX	RHR	1	590 0		RHR1-27
854	R3001B027	9010	EDG12 AIR COOLANT HX	RHR	1	590 0		RHR1-27
855	R3001B028	9011	EDG14 AIR COOLANT HX	RHR	1	590 0		RHR1-27
856	R3001C005	2212	DGSW PUMP MOTOR	RHR	1	594 0		RHR1-15
857	R3001C006	2213	DGSW PUMP MOTOR	RHR	1	594 0		RHR1-15
858	R3001C007	2214	DGSW PUMP MOTOR	RHR	1	594 0		RHR1-15
859	R3001C008	2215	DGSW PUMP MOTOR	RHR	1	594 0		RHR1-15
860	R3001S001	2401	GOVERNOR ACTUATOR	RHR	1	595 0	EDG 11	RHR1-08
861	R3001S002	2402	GOVERNOR ACTUATOR	RHR	1	595 0	EDG 12	RHR1-08
862	R3001S003	2403	GOVERNOR ACTUATOR	RHR	1	595 0	EDG 13	RHR1-08
863	R3001S004	2404	GOVERNOR ACTUATOR	RHR	1	595 0	EDG 14	RHR1-08
864	R30FA04A	2328	AIR START SOLENOID VALVES	RHR	1	59200		RHR1-09
865	R30FA04B	2329	AIR START SOLENOID VALVES	RHR	1	59200		RHR1-09
866	R30FA04C	2330	AIR START SOLENOID VALVES	RHR	1	59200		RHR1-09
867	R30FA04D	2331	AIR START SOLENOID VALVES	RHR	1	59200		RHR1-09
868	R30FA05A	2332	AIR START SOLENOID VALVES	RHR	1	59200	EQUIP	RHR1-09
869	R30FA05B	2333	AIR START SOLENOID VALVES	RHR	1	59200		RHR1-09
870	R30FA05C	2334	AIR START SOLENOID VALVES	RHR	1	59200	EQUIP	RHR1-09
871	R30FA05D	2335	AIR START SOLENOID VALVES	RHR	1	59200	EQUIP	RHR1-09
872	R30N563A	2357	F.O. DAY TANK LEVEL SWITCH	RHR	1	61100	EQUIP	
873	R30N563B	2358	F.O. DAY TANK LEVEL SWITCH	RHR	1	61100	EQUIP	
874	R30N563C	2359	F.O. DAY TANK LEVEL SWITCH	RHR	1	61100	EQUIP	
875	R30N563D	2360	F.O. DAY TANK LEVEL SWITCH	RHR	1	61100	EQUIP	
876	R30N568A	801	FLOW TRANSMITTER	RHR	1	59500	PSD-35	
877	R30N568B	813	FLOW TRANSMITTER	RHR	1	59500	PSD-48	
878	R30N568C	807	FLOW TRANSMITTER	RHR	1	59500	PSD-47	
879	R30N568D	819	FLOW TRANSMITTER	RHR	1	59500	PSD-36	
880	R30NA08A	2361	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
881	R30NA08B	2362	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
882	R30NA08C	2363	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
883	R30NA08D	2364	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
884	R30NA09A	2365	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
885	R30NA09B	2366	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
886	R30NA09C	2367	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
887	R30NA09D	2368	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
888	R30NA10A	2369	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
889	R30NA10B	2370	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
890	R30NA10C	2371	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
891	R30NA10D	2372	CRANKCASE PRESSURE SWITCH	RHR	1	59200	EQUIP	
892	R30NA11A	2373	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
893	R30NA11B	2374	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
894	R30NA11C	2375	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
895	R30NA11D	2376	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
896	R30NA12A	2377	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
897	R30NA12B	2378	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
898	R30NA12C	2379	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
899	R30NA12D	2380	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
900	R30NA13A	2381	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
901	R30NA13B	2382	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
902	R30NA13C	2383	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
903	R30NA13D	2384	LUBE OIL LOW PRESSURE SWITCH	RHR	1	59200	EQUIP	
904	R30NA16A	2340	COOLANT PRESSURE SWITCH	RHR	1	59300	R30P310	
905	R30NA16B	2341	COOLANT PRESSURE SWITCH	RHR	1	59300	R30P320	
906	R30NA16C	2342	COOLANT PRESSURE SWITCH	RHR	1	59300	R30P330	
907	R30NA16D	2343	COOLANT PRESSURE SWITCH	RHR	1	59300	R30P340	
908	R30NA17A	2348	SIGNAL GENERATOR	RHR	1	59700	SKID	
909	R30NA17B	2349	SIGNAL GENERATOR	RHR	1	59700	SKID	
910	R30NA17C	2350	SIGNAL GENERATOR	RHR	1	59700	SKID	
911	R30NA17D	2351	SIGNAL GENERATOR	RHR	1	59700	SKID	
912	R30NA18A	2517	A.C. PNEUMATIC TEMP TRANS	RHR	1	59800	R30P310	
913	R30NA18B	2518	A.C. PNEUMATIC TEMP TRANS	RHR	1	59800	R30P320	
914	R30NA18C	2519	A.C. PNEUMATIC TEMP TRANS	RHR	1	59800	R30P330	
915	R30NA18D	2520	A.C. PNEUMATIC TEMP TRANS	RHR	1	59800	R30P340	
916	R30NA19A	2513	A.C. PNEUMATIC TEMP CONTROLLER	RHR	1	59800	R30P310	
917	R30NA19B	2514	A.C. PNEUMATIC TEMP CONTROL	RHR	1	59800	R30P320	
918	R30NA19C	2515	A.C. PNEUMATIC TEMP CONTROLL	RHR	1	59800	R30P330	
919	R30NA19D	2516	A.C. PNEUMATIC TEMP CONTROLLER	RHR	1	59800	R30P340	
920	R30P310	1861	EDG GAUGE/RELAY PANEL	RHR	1	590 0		RHR1-02
921	R30P311	1862	RELAY PANEL	RHR	2	61700	WALL	RHR2-03
922	R30P312	1886	INSTRUMENT RACK	RHR	1	59000	R3001S001	RHR1-03
923	R30P320	1863	RELAY PANEL	RHR	1	590 0		RHR1-02
924	R30P321	1864	RELAY PANEL	RHR	2	61700	WALL	RHR2-03
925	R30P322	1887	INSTRUMENT RACK	RHR	1		R3001S002	RHR1-03
926	R30P330	1865	RELAY PANEL	RHR	1	590 0		RHR1-02
927	R30P331	1866	RELAY PANEL	RHR	2	61700	WALL	RHR2-03
928	R30P332	1888	INSTRUMENT RACK	RHR	1	59000	R3001S003	RHR1-03
929	R30P340	1867	RELAY PANEL	RHR	1	590 0		RHR1-02
930	R30P341	1868	RELAY PANEL	RHR	2	61700	WALL	RHR2-03
931	R30P342	1889	INSTRUMENT RACK	RHR	1		R3001S004	RHR1-03
932	R30P405A	8009	PIPE STAND (REF R30N568@)	RHR	1	59000	FLOOR	RHR1-06
933	R30P405B	8010	PIPE STAND (REF R30N568@)	RHR	1	59000	FLOOR	RHR1-06
934	R30P405C	8011	PIPE STAND (REF R30N568@)	RHR	1	59000	FLOOR	RHR1-06
935	R30P405D	8012	PIPE STAND (REF R30N568@)	RHR	1	59000	FLOOR	RHR1-06
936	R30R003A	2493	RUN HOUR METER	RHR	2	61700	R3000S005	
937	R30R003B	2494	RUN HOUR METER	RHR	2	61700	R3000S007	
938	R30R003C	2495	RUN HOUR METER	RHR	2	61700	R3000S006	
939	R30R003D	2496	RUN HOUR METER	RHR	2	61700	R3000S008	
940	R30R008A	2481	WATTMETER	RHR	2	61700	R3000S005	
941	R30R008B	2482	WATTMETER	RHR	2	61700	R3000S007	
942	R30R008C	2483	WATTMETER	RHR	2	61700	R3000S006	
943	R30R008D	2484	WATTMETER	RHR	2	61700	R3000S008	
944	R30R009A	2489	FREQUENCY METER	RHR	2	61700	R3000S005	
945	R30R009B	2490	FREQUENCY METER	RHR	2	61700	R3000S007	
946	R30R009C	2491	FREQUENCY METER	RHR	2	61700	R3000S006	
947	R30R009D	2492	FREQUENCY METER	RHR	2	61700	R3000S008	
948	R3101S001	1875	MPU #1	AB	2	613 6		SGR1-03
949	R3101S002	1876	MPU #2	AB	3	643 6		SGR2-01
950	R31K001	2127	FEED - B21N094A	AB	2	61306	H11P613	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
951	R31K002	2128	OUTPUT PS - B21R623A	AB	2	61306	H11P613	
952	R31K003	8000	1KVA INVERTER-DEP. INST.	AB	2	6136	H11P626	
953	R31K004	2129	FEED - B21N091B	AB	2	61306	H11P612	
954	R31K005	2130	OUTPUT PS - B21R623B	AB	2	61306	H11P612	
955	R3200S003	900	2PA BATTERIES	AB	3	64306	DIV I	BAT-04
956	R3200S004	913	2PB BATTERIES	AB	3	64306	DIV II	BAT-04
957	R3200S015	904	MCC	AB	3	643 6	2PA-1	BAT-03
958	R3200S016	917	MCC	AB	3	643 6	2PB-1	BAT-03
959	R3200S020A	906	2A-1 BATTERY CHARGER	AB	3	643 6	DIV I	BAT-01
960	R3200S020B	907	2A-2 BATTERY CHARGER	AB	3	643 6	DIV I	BAT-01
961	R3200S020C	908	2A1-2 BATTERY CHARGER	AB	3	643 6	DIV I	BAT-01
962	R3200S021A	919	2B-1 BATTERY CHARGER	AB	3	643 6	DIV I	BAT-01
963	R3200S021B	920	2B-2 BATTERY CHARGER	AB	3	643 6	DIV I	BAT-01
964	R3200S021C	921	2B1-2 BATTERY CHARGER	AB	3	643 6	DIV I	BAT-01
965	R3200S026	905	DISTRIBUTION CABINET	AB	3	643 6	2PA-2	BAT-02
966	R3200S027	918	DISTRIBUTION CABINET	AB	3	643 6	2PB-2	BAT-02
967	R3200S061A	911	DISTRIBUTION CABINET	AB	2	61306	2PA2-5	RR-02
968	R3200S061B	912	DISTRIBUTION CABINET	AB	2	613 6	2PA2-6	RR-02
969	R3200S062	909	DISTRIBUTION CABINET	AB	2	61306		SGR1-08
970	R3200S063	910	DISTRIBUTION CABINET	RHR	2	617 0	2PA2-13	RHR2-08
971	R3200S064A	924	DISTRIBUTION CABINET	AB	2	613 6	2PB2-5	RR-02
972	R3200S064B	925	DISTRIBUTION CABINET	AB	2	613 6	2PB2-6	RR-02
973	R3200S065	922	DISTRIBUTION CABINET	AB	3	643 6	2PB2-15	SGR2-06
974	R3200S066	923	DISTRIBUTION CABINET	RHR	2	617 0	2PB2-14	RHR2-08
975	T2300F400	8005	TORUS TO DRYWELL VACUUM BRKRS.	RB	B	56401	TORUS	DW-17
976	T2300F409	8547	PC VAC.BRKR.VALVE	RB			V21-2015	RBTR-23
977	T2300F410	8548	PC VAC.BRKR.VALVE	RB			V21-2016	
978	T2300X000	8004	PRIMARY CONT. PENETRATIONS	DW	1	600 0	DRYWELL	
979	T23F409	1163	SOLENOID VALVE	RB	B	57300	WALL	RBTR-22
980	T23F410	1325	SOLENOID VALVE	RB	B	57300	WALL	RBTR-22
981	T23N010A	8549	PR.DIFF.IND.SWITCH	RB		548-0		RBSB-06
982	T23N010B	8550	PR.DIFF.IND.SWITCH	RB		548-0		RBSB-06
983	T4100B002	68	SWGR RM SPACE COOLER	AB	2	613 6		SGR1-04
984	T4100B003	73	SWGR RM SPACE COOLER	AB	2	613 6		SGR1-04
985	T4100B004	409	SWGR ROOM SPACE COOLER	AB	3	643 6		SGR2-02
986	T4100B005	426	SWGR RM SPACE COOLER	AB	3	643 6		SGR2-02
987	T4100B006	1523	DIV 2 AC FAN	AB	5	680 0		AB5-11
988	T4100B007	1504	DIV 1 AC FAN	AB	5	680 0		AB5-11
989	T4100B008	440	A/C COOLER	AB	5	677 6		AB5-06
990	T4100B008A	8100	COOLER OIL PUMP	AB	5	677 6		AB5-26
991	T4100B009	95	A/C SPACE COOLER	AB	5	677 6		AB5-06
992	T4100B009A	8101	COOLER OIL PUMP	AB	5	677 6		AB5-26
993	T4100B018	128	RHR SPACE COOLER	RB	SB	540 0		RBSB-02
994	T4100B019	468	RHR DIV 2 SPACE COOLER	RB	SB	540 0		RBSB-02
995	T4100B021	154	CS AND RCIC CORNER RM SPACE CO	RB	SB	555 0		RCIC-14
996	T4100B022	512	HPCI SPACE COOLER	AB	SB	540 0		HPCI-19
997	T4100B029	140	SPACE COOLER	AB	B	554 0		ABB-06
998	T4100B030	484	SPACE COOLER	AB	B	554 0		ABB-06
999	T4100B034	49	EECW SPACE COOLER	RB	2	613 6		RB2-13
1000	T4100B035	396	EECW SPACE COOLER	RB	2	613 6		RB2-13

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
1001	T4100B043	89	BATTERY CHARGER SPACE COOLER	AB	3	65100		BAT-05
1002	T4100B044	432	BATTERY CHARGER SPACE COOLER	AB	3	65106		BAT-05
1003	T4100C030	1535	CC-HVAC RETURN AIR FAN	AB	5	677 6		AB5-09
1004	T4100C031	1516	CC-HVAC RETURN AIR FAN	AB	5	677 0		AB5-09
1005	T4100C040	1541	DIV 2 CHILLED WATER PUMP	AB	5	677 6		AB5-01
1006	T4100C041	1538	DIV 1 CHILLED WATER PUMP	AB	5	677 6		AB5-01
1007	T4100F031A	1545	DIV 1 RETURN DAMPER	AB	4			CR-05
1008	T4100F031B	1546	DIV 2 RETURN DAMPER	AB	4			CR-05
1009	T4100F033A	1502	DIV 1 OUTSIDE AIR DAMPER	AB	4			CR-05
1010	T4100F033B	1521	DIV 2 OUTSIDE AIR DAMPER	AB	4			CR-05
1011	T4100F035	1503	DIV 1 SHUTOFF DAMPER	AB	4			CR-05
1012	T4100F038	1522	DIV 2 SHUTOFF DAMPER	AB	4			AB4-04
1013	T4100F039B	1517	RETURN FAN SHUTOFF DAMPER	AB	5			AB5-07
1014	T4100F040B	1536	RETURN FAN SHUTOFF DAMPER	AB	5			AB5-07
1015	T4100F041	1501	CC-HVAC INTAKE VALVE	AB	4			AB4-03
1016	T4100F042	1500	CC-HVAC INTAKE VALVE	AB	4			AB4-03
1017	T4100F068A	1506	DIV 1 SHUTOFF DAMPER	AB	5	68600		AB5-10
1018	T4100F068B	1507	DIV 1 SHUTOFF DAMPER	AB	5	68600		AB5-10
1019	T4100F069A	1525	DIV 2 SHUTOFF DAMPER	AB	5			AB5-10
1020	T4100F069B	1526	DIV 2 SHUTOFF DAMPER	AB	5			AB5-10
1021	T4100F083A	8551	HVAC EXH.AIR FIRE DAMPER	AB	4			
1022	T4100F083B	8552	HVAC EXH.AIR FIRE DAMPER	AB	4			
1023	T4100F084A	8553	HVAC NML.AIR INT.FR.DMPR.	AB	4			
1024	T4100F084B	8554	HVAC NML.AIR INT.FR.DMPR.	AB	4			
1025	T4100F085	8555	HVAC INT.SHTOFF.FR.DMPR.	AB	4			
1026	T4100F086	8556	HVAC RR INLT.FR.DMPR.	AB	2			
1027	T4100F087	8557	HVAC RR EXH.FR.DMPR.	AB	2			
1028	T4100F088	8558	HVAC SPR.RM.EX.FR.DMPR.	AB	3			
1029	T4100F089	8559	HVAC SPR.RM.INL.FR.DMPR.	AB	3			
1030	T4100F099	8560	HVAC MLTZN.AC IN.FR.DMPR	AB	5			
1031	T4100F100	8561	HVACMLTZN.AC.EX.FR.DMPR	AB	5			
1032	T4100F101	8562	HVACMLTZN.AC.EX.FR.DMPR	AB	5			
1033	T4100F102	8563	HVAC CR AIR SPLY.FR.DMPR	AB	5			
1034	T4100F109	8564	HVAC RTN.AIR FN.IN.FR.DMP	AB	5	677-0	DUCT	
1035	T4100F110	8565	HVAC RTN.AIR FN.IN.FR.DMP	AB	5	677-0	DUCT	
1036	T4100F157A	1508	ZONE 1 MODULATING DAMPER	AB	5	677 6		AB5-12
1037	T4100F157B	1527	ZONE 1 MODULATING DAMPER	AB	5	677 6		AB5-12
1038	T4100F158A	1509	ZONE 2 MODULATING DAMPER	AB	5	677 6		AB5-12
1039	T4100F158B	1528	ZONE Z MODULATING DAMPER	AB	5	677 6		AB5-12
1040	T4100F159A	1510	ZONE 3 MODULATING DAMPER	AB	5	677 6		AB5-12
1041	T4100F159B	1529	ZONE 3 MODULATING DAMPER	AB	5	677 6		AB5-12
1042	T4100F160A	1511	ZONE 4 MODULATING DAMPER	AB	5	677 6		AB5-12
1043	T4100F160B	1530	ZONE 4 MODULATING DAMPER	AB	5	677 6		AB5-12
1044	T4100F161A	1512	ZONE 6 MODULATING DAMPER	AB	5	677 6		AB5-12
1045	T4100F161B	1531	ZONE 6 MODULATING DAMPER	AB	5	677 6		AB5-12
1046	T4100F162A	1513	ZONE 7 MODULATING DAMPER	AB	5	677 6		AB5-12
1047	T4100F162B	1532	ZONE 7 MODULATING DAMPER	AB	5	677 6		AB5-12
1048	T4100F163A	1514	ZONE 8 MODULATING DAMPER	AB	5	677 6		AB5-12
1049	T4100F163B	1533	ZONE 8 MODULATING DAMPER	AB	5	677 6		AB5-12
1050	T4100F903	8566	HVACRTN.AIR FN.SO.FR.DMP	AB	5	677-0	DUCT	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
1051	T41F025A	3574	SOLENOID VALVE FOR T4100F035	AB	5	67706	H21P296A	
1052	T41F025B	3592	SOLENOID VALVE FOR T4100F038	AB	5	67706	H21P296B	
1053	T41F026A	3584	SOLENOID VALVE FOR T4100F039B	AB	5	67706	H21P296A	
1054	T41F026B	3602	SOLENOID VALVE FOR T4100F040B	AB	5	67706	H21P296B	
1055	T41F071A	8567	CCHVAC CH.CMPR.SOL.VLV.	AB	5	684-0	EQUIP	AB5-31
1056	T41F071B	8568	CCHVAC CH.CMPR.SOL.VLV.	AB	5	684-0	EQUIP	AB5-31
1057	T41F072A	8102	FLOW CONTROL VALVE	AB	5	67706	T4100B009	AB5-14
1058	T41F072B	8103	FLOW CONTROL VALVE	AB	5	67706	T4100B008	AB5-14
1059	T41F073A	8569	CCHVAC CMPR.FL.CTRL.VLV.	AB	5	682-0	EQUIP	AB5-22
1060	T41F073B	8570	CCHVAC CMPR.FL.CTRL.VLV.	AB	5	682-0	EQUIP	AB5-22
1061	T41F074A	8571	CCHVAC CMPR.FL.CTRL.VLV.	AB	5	684-0	EQUIP	AB5-27
1062	T41F074B	8572	CCHVAC CMPR.FL.CTRL.VLV.	AB	5	684-0	EQUIP	AB5-27
1063	T41F083B	8573	SGT RM.ISO.VLV.SOL.VLV.	AB	5	677-6	H21P296B	AB5-28
1064	T41F084A	3571	SOLENOID VALVE FOR T4100F041	AB	4	66600	EQUIP	AB4-02
1065	T41F084B	3570	SOLENOID VALVE FOR T4100F042	AB	4	66600	EQUIP	AB4-02
1066	T41F085A	3585	SOLENOID VALVE FOR T4100F043	AB	4	67406	EQUIP	AB4-09
1067	T41F086A	8574	CCHVAC ISO.VLV.SOL.VLV.	AB	5	677-6	H21P296A	AB5-29
1068	T41F086B	8575	CCHVAC ISO.VLV.SOL.VLV.	AB	5	677-6	H21P296B	AB5-29
1069	T41F088A	8576	CCHVAC ISO.VLV.SOL.VLV.	AB	5	677-6	H21P296A	AB5-29
1070	T41F089A	8577	CCHVAC ISO.VLV.SOL.VLV.	AB	5	677-6	H21P296A	AB5-29
1071	T41F092B	3593	SOLENOID VALVE FOR T4100F069A	AB	5	67706	H21P296F	
1072	T41F093A	3575	SOLENOID VALVE FOR T4100F068A	AB	5	67706	H21P296E	
1073	T41F099A	3603	SOLENOID VALVE FOR T4100F031A	AB	5	67706	H21P296A	
1074	T41F099B	3606	SOLENOID VALVE FOR T4100F031B	AB	5	67706	H21P296B	
1075	T41F100A	3604	SOLENOID VALVE FOR T4100F031A	AB	5	67706	H21P296A	
1076	T41F100B	3607	SOLENOID VALVE FOR T4100F031B	AB	5	67706	H21P296B	
1077	T41F101A	3572	SOLENOID VALVE FOR T4100F033A	AB	5	67706	H21P296A	
1078	T41F101B	3591	SOLENOID VALVE FOR T4100F033B	AB	5	67706	H21P296B	
1079	T41F102A	3573	SOLENOID VALVE FOR T4100F033A	AB	5	67706	H21P296A	
1080	T41F103A	8578	CCHVAC RCRC.DMP.SL.VLV.	AB	5	677-6	H21P296A	AB5-30
1081	T41F103B	8579	CCHVAC RCRC.DMP.SL.VLV.	AB	5	677-6	H21P296B	AB5-30
1082	T41F104A	8580	CCHVAC RCRC.DMP.SL.VLV.	AB	5	677-6	H21P296A	AB5-30
1083	T41F104B	8581	CCHVAC RCRC.DMP.SL.VLV.	AB	5	677-6	H21P296B	AB5-30
1084	T41F107A	3576	SOLENOID VALVE FOR T4100F068B	AB	5	67706	H21P296E	
1085	T41F107B	3594	SOLENOID VALVE FOR T4100F069B	AB	5	67706	H21P296F	
1086	T41F111A	8582	CCHVAC IN.RK.PR.CTRL.VLV.	AB	5	677-6	H21P296A	AB5-18
1087	T41F111B	8583	CCHVAC IN.RK.PR.CTRL.VLV.	AB	5	677-6	H21P296B	AB5-18
1088	T41F114A	8584	CCHVAC IN.RK.PR.CTRL.VLV.	AB	5	677-6	H21P296E	AB5-19
1089	T41F114B	8585	CCHVAC IN.RK.PR.CTRL.VLV.	AB	5	677-6	H21P296F	AB5-19
1090	T41F132	8586	RBHVAC DMPR.PILOT VLV.	AB	4	673-0	T4100F062A	AB4-08
1091	T41F134	8587	RBHVAC DMPR.PILOT VLV.	AB	4	673-0	T4100F063A	AB4-08
1092	T41F142	8588	RBHVAC DMPR.PILOT VLV.	AB	5	684-0	T4100F054	AB5-17
1093	T41F143	8589	RBHVAC DMPR.PILOT VLV.	AB	5	684-0	T4100F053	AB5-17
1094	T41F144	8590	RBHVAC IS.DMPR.SOL.VLV.	AB	5	677-6	T4100F037	AB5-16
1095	T41F145	8591	RBHVAC DMPR.PILOT VLV.	AB	5	692-0	T4100F036	AB5-17
1096	T41F160	8592	RBHVAC REGULATOR VALVE	AB	5	687-0	T4100F036	AB5-15
1097	T41F161	8593	RBHVAC PR.CNTRL.VALVE	AB	5	692-0	T4100F037	AB5-15
1098	T41F162	8594	RBHVAC PR.CNTRL.VALVE	AB	5	670-6	T4100F041	AB5-15
1099	T41F164	8595	RBHVAC PR.CNTRL.VALVE	AB	4	673-0	T4100F043	AB4-07
1100	T41F181	8596	RB DMPR.SWITCH.VALVE	AB	5	684-0	T4100F053	AB5-20

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
1101	T41F182	8597	RB DMPR.SWITCH.VALVE	AB	5	684-0	T4100F054	AB5-20
1102	T41F183	8598	RB DMPR.SWITCH.VALVE	AB	4	673-0	T4100F062A	AB4-06
1103	T41F185	8599	RB DMPR.SWITCH.VALVE	AB	4	673-0	T4100F063A	AB4-06
1104	T41F187	8600	RB DMPR.SWITCH.VALVE	AB	5	687-0	T4100F036	AB5-20
1105	T41F188	8601	RB DMPR.SWITCH.VALVE	AB	5	691-0	T4100F037	AB5-20
1106	T41F189	8602	RB DMPR.SWITCH.VALVE	AB	4	670-6	T4100F041	AB4-06
1107	T41F191	8603	RB DMPR.SWITCH.VALVE	AB	4	673-0	T4100F043	AB4-06
1108	T41F382A	8104	REGULATING VALVE	AB	5	67706		AB5-32
1109	T41F382B	8105	REGULATING VALVE	AB	5	67706		AB5-32
1110	T41F384A	8106	REGULATING VALVE	AB	5	67706		AB5-32
1111	T41F384B	8107	REGULATING VALVE	AB	5	67706		AB5-21
1112	T41K001A	8604	CCHVAC TEMP.CONTROLLER	AB	5	677-6	H21P296A	
1113	T41K001B	8605	CCHVAC TEMP.CONTROLLER	AB	5	677-6	H21P296B	
1114	T41K007A	3605	TEMP CONTROLLER FOR T4100F031A	AB	5	67706	H21P296A	
1115	T41K007B	3608	TEMP CONTROLLER FOR T4100F031B	AB	5	67706	H21P296B	
1116	T41K030A	8606	CCHVAC COMPR.CONTROLL.	AB	5	677-6	H21P285A	
1117	T41K030B	8607	CCHVAC COMPR.CONTROLL.	AB	5	677-6	H21P285B	
1118	T41K032A	8608	CCHVAC COMPR.CONTROLL.	AB	5	677-6	T4100B009	
1119	T41K032B	8609	CCHVAC COMPR.CONTROLL.	AB	5	677-6	T4100B008	
1120	T41K032C	8610	CCHVAC COMPR.CONTROLL.	AB	5	677-6	T4100B009	
1121	T41K032D	8611	CCHVAC COMPR.CONTROLL.	AB	5	677-6	T4100B008	
1122	T41K039A	8612	CCHVAC DMP.CONVERTER	AB	5	677-6	H21P296A	
1123	T41K039B	8613	CCHVAC DMP.CONVERTER	AB	5	677-6	H21P296B	
1124	T41N059A	8614	CCHVAC PMP.DIFF.SWITCH	AB	5	677-6	MOPS	AB5-24
1125	T41N059B	8615	CCHVAC PMP.DIFF.SWITCH	AB	5	677-6	MOPS	AB5-24
1126	T41N060A	100	PRESSURE DIFFERENTIAL SWITCH	AB	5	67706	H21P572	
1127	T41N060B	446	PRESSURE DIFFERENTIAL SWITCH	AB	5	67706	H21P573	
1128	T41N061A	3577	TEMP SWITCH FOR T4100F157A	AB	2	61800	WALL	RR-12
1129	T41N061B	3595	TEMP SWITCH FOR T4100F157B	AB	2	61800	WALL	RR-12
1130	T41N062A	3578	TEMP SWITCH FOR T4100F158A	AB	2	63500	WALL	RR-12
1131	T41N062B	3596	TEMP SWITCH FOR T4100F158B	AB	2	63500	WALL	RR-12
1132	T41N063A	3579	TEMP SWITCH FOR T4100F159A	AB	3	64800	WALL	CR-01
1133	T41N063B	3597	TEMP SWITCH FOR T4100F159B	AB	3	64800	WALL	CR-01
1134	T41N065A	3582	TEMP SWITCH FOR T4100F162A	AB	3	64806	WALL	CR-01
1135	T41N065B	3600	TEMP SWITCH FOR T4100F162B	AB	3	64800	WALL	CR-01
1136	T41N066A	3581	TEMP SWITCH FOR T4100F161A	AB	3	64806	WALL	CR-01
1137	T41N066B	3599	TEMP SWITCH FOR T4100F161B	AB	3	64800	WALL	CR-01
1138	T41N067A	3580	TEMP SWITCH FOR T4100F160A	AB	4	66006	WALL	CR-04
1139	T41N067B	3598	TEMP SWITCH FOR T4100F160B	AB	4	66406	WALL	CR-04
1140	T41N068A	3583	TEMP SWITCH FOR T4100F163A	AB	5	68200	WALL	AB5-03
1141	T41N068B	3601	TEMP SWITCH FOR T4100F163B	AB	5	68200	WALL	AB5-03
1142	T41N117A	8616	CCHVAC TEMP.TRNSMITTER.	AB	5	686-0		AB5-33
1143	T41N117B	8617	CCHVAC TEMP.TRNSMITTER.	AB	5	683-0		AB5-33
1144	T41N132A	8618	CCHVAC PR.DIFF.SWITCH	AB	5	677-6	H21P296A	
1145	T41N132B	8619	CCHVAC PR.DIFF.SWITCH	AB	5	677-6	H21P296B	
1146	T41N134A	8620	CCHVAC PR.DIFF.SWITCH	AB	5	677-6	H21P296A	
1147	T41N134B	8621	CCHVAC PR.DIFF.SWITCH	AB	5	677-6	H21P296B	
1148	T41N222A	8622	CCHVAC EVAP.TMP.SWTCH.	AB	5	677-6	H21P285A	
1149	T41N222B	8623	CCHVAC EVAP.TMP.SWTCH.	AB	5	677-6	H21P285B	
1150	T41N260	8624	RBHVAC DMP.PNEU.CNTRLR	AB	5	687-0	T4100F036	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
1151	T41N261	8625	RBHVAC DMP.PNEU.CNTRLR	AB	5	691-0	T4100F037	
1152	T41N262	8626	RBHVAC DMP.PNEU.CNTRLR	AB	4	670-6	T4100F041	
1153	T41N264	8627	RBHVAC DMP.PNEU.CNTRLR	AB	4	674-0	T4100F043	
1154	T41N309A	8628	CCHVAC DMP.PNEU.XMTR	AB	5	677-6	H21P296A	
1155	T41N309B	8629	CCHVAC DMP.PNEU.XMTR.	AB	5	677-6	H21P296B	
1156	T41N310A	8630	CCHVAC PR.CTR.PNEU.XMTR	AB	5	677-6	H21P296A	
1157	T41N310B	8631	CCHVAC PR.CTR.PNEU.XMTR	AB	5	677-6	H21P296B	
1158	T41N322A	8632	CCHVAC CMPR.TEMP.SWCH.	AB	5	677-6	H21P285A	
1159	T41N322B	8633	CCHVAC CMPR.TEMP.SWCH.	AB	5	677-6	H21P285B	
1160	T41N323A	8634	CCHVAC PR.DIFF.SWITCH	AB	5	677-6	H21P285A	
1161	T41N323B	8635	CCHVAC PR.DIFF.SWITCH	AB	5	677-6	H21P285B	
1162	T41N324A	8636	CCHVAC OIL TEMP.SWITCH	AB	5	677-6	H21P285A	
1163	T41N324B	8637	CCHVAC OIL TEMP.SWITCH	AB	5	677-6	H21P285B	
1164	T41N325A	8638	CCHVAC OIL TEMP.SWITCH	AB	5	677-6	H21P285A	
1165	T41N325B	8639	CCHVAC OIL TEMP.SWITCH	AB	5	677-6	H21P285B	
1166	T41N326A	8640	CCHVAC COND.PR.SWITCH	AB	5	677-6	H21P285A	
1167	T41N326B	8641	CCHVAC COND.PR.SWITCH	AB	5	677-6	H21P285B	
1168	T41N327A	8642	CCHVAC MTR.TEMP.SWITCH	AB	5	677-6	H21P285A	
1169	T41N327B	8643	CCHVAC MTR.TEMP.SWITCH	AB	5	677-6	H21P285B	
1170	T41N328A	8644	CHVAC PRG.SYS.TMP.SWCH	AB	5	682-0	EQUIP	
1171	T41N328B	8645	CHVAC PRG.SYS.TMP.SWCH	AB	5	682-0	EQUIP	
1172	T41N334A	8646	CCHVAC EVAP.TMP.ELEMNT	AB	5	677-6	EQUIP	
1173	T41N334B	8647	CCHVAC EVAP.TMP.ELEMNT	AB	5	677-6	EQUIP	
1174	T41N369A	8648	CCHVAC PRES.ELEMENT	AB	5	682-0	DUCT	
1175	T41N369B	8649	CCHVAC PRES.ELEMENT	AB	5	682-0	DUCT	
1176	T41N371A	8650	CCHVAC PRES.ELEMENT	AB	5	684-6	DUCT	
1177	T41N371B	865i	CCHVAC PRES.ELEMENT	AB	5	684-6	DUCT	
1178	T41N456A	8652	CURRENT TRANSFORMER	AB	5	684-0	EQUIP	AB5-25
1179	T41N456B	8653	CURRENT TRANSFORMER	AB	5	684-0	EQUIP	AB5-25
1180	T41N463A	8654	CCHVAC DMP.PNEU.XMTR.	AB	5	677-6	H21P296A	
1181	T41N463B	8655	CCHVAC DMP.PNEU.XMTR.	AB	5	677-6	H21P296B	
1182	T4800F451	7029	NITROGEN INERTING ISO VALVE	RB	2			RB2-18
1183	T4800F453	7023	INERTING BYPASS ISOLATION VA	RB	B	57600	EQUIP	RBTR-06
1184	T4800F454	7022	NITROGEN SUPPLY TO DW OUTBD IS	DW	B	57600	EQUIP	RBTR-06
1185	T4800F455	7021	NITROGEN INERTING INBD ISO VA	DW	B	58002	EQUIP	DW-20
1186	T4800F456	7036	NITROGEN INERTING SUPPLY VA	RB	B	57600	EQUIP	RBTR-06
1187	T4800F457	7037	NITROGEN INERTING ISOLATION VA	RB	B		EQUIP	RBTR-06
1188	T4800F458	7038	NITROGEN INERTING BYPASS ISO V	RB	B		EQUIP	RBTR-06
1189	T4901F021	3010	RELIEF VALVE	DW	1	61406		DW-24
1190	T4901F024	3025	RELIEF VALVE	DW	1	61600		DW-24
1191	T4901F027	3040	RELIEF VALVE	DW	1	61706		DW-24
1192	T4901F030	3055	RELIEF VALVE	DW	1	61803		DW-24
1193	T4901F033	3070	RELIEF VALVE	DW	1	61803		DW-24
1194	T4901F465	7020	NITROGEN OUTBD ISOLATION MOV	RB	1	60006		RB1-02
1195	T4901F468	7024	NITROGEN OUTBD ISOLATION VA	RB	2	63300		RB2-20
1196	T4901F601	7019	NITROGEN INBD ISOLATION MOV	DW	1	602 0	V4-2080	DW-06
1197	T4901F602	7025	NITROGEN INBD ISOLATION VA	DW	2	63006	V4-2188	DW-11
1198	T49P400A	1157	INSTRUMENT RACK	RB	1	58306	FLOOR	RB1-18
1199	T49P400B	1319	INSTRUMENT RACK	RB	2	61306	FLOOR	RB2-21
1200	T5000F455	7048	PCMC ISOLATION VALVE	RB	2	62006	H21P284	RB2-14

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
1201	T5000F456	7031	CAS OUTBD ISOLATION VALVE	RB	2	62103	H21P284	RB2-14
1202	T50F450	7030	CAS OUTBD ISOLATION VALVE	RB	2	62103	H21P284	
1203	T50F451	7047	PCMC ISOLATION VALVE	RB	2	62006	H21P284	RB2-15
1204	T50K001A	8656	PC TRS.PR.MOD.SIG.COND.	AB	2	613-6	H11P613	
1205	T50K800A	1946	PS - T50N406A	AB	2	61306	H11P914	
1206	T50K800B	1948	PS - T50N406B	AB	2	61306	H11P915	
1207	T50K801A	8657	PCMS DW PR.SIG.COND.	AB	2	613-6	H11P613	
1208	T50K801B	8658	PCMS DW PR.SIG.COND.	AB	2	613-6	H11P612	
1209	T50K802A	8659	PCMS DW PR.SIG.COND.	AB	2	613-6	H11P613	
1210	T50K802B	8660	PCMS DW PR.SIG.COND.	AB	2	613-6	H11P612	
1211	T50N400A	3518	SOURCE INST. - T50R800A	DW	4	66407	T50R800A	
1212	T50N400B	3519	SOURCE INST. - T50R800B	DW	4	66407	T50R800B	
1213	T50N401A	8119	PC DW PR.NARROW RANGE PR.TRANS	RB	2	618-0	H21P595A	
1214	T50N401B	8661	PCMS DW PR.XMTR.	RB	2	618-0	H21P595B	
1215	T50N402A	3533	SOURCE INST. - T50R800A	RB	B	56806	T50R800A	
1216	T50N402B	3534	SOURCE INST. - T50R800B	RB	B	56806	T50R800B	
1217	T50N403A	3535	SOURCE INST. - T50R800A	RB	B	56806	T50R800A	
1218	T50N403B	3536	SOURCE INST. - T50R800B	RB	B	56806	T50R800B	
1219	T50N404A	3537	SOURCE INST. - T50R800A	RB	B	55104	T50R800A	
1220	T50N404B	3538	SOURCE INST. - T50R800B	RB	B	55104	T50R800B	
1221	T50N405B	3540	SOURCE INST. - T50R800B	RB	B	55104	T50R800B	
1222	T50N406A	1945	SOURCE INST. - T50R804A	RB	SB	54000	H21P614A	
1223	T50N406B	1947	SOURCE INST. - T50R804B	RB	SB	54000	H21P614B	
1224	T50N407A	3520	SOURCE INST. - T50R800B	DW	1	59700	T50R800B	
1225	T50N407B	3521	SOURCE INST. - T50R800A	DW	1	59700	T50R800A	
1226	T50N408A	3528	SOURCE INST. - T50R800B	DW	1	59700	T50R800B	
1227	T50N408B	3529	SOURCE INST. - T50R800B	DW	1	59700	T50R800B	
1228	T50N409A	3522	SOURCE INST. - T50R800A	DW	1	59700	T50R800A	
1229	T50N409B	3523	SOURCE INST. - T50R800B	DW	1	59700	T50R800B	
1230	T50N410A	3524	SOURCE INST. - T50R800A	DW	1	59700	T50R800A	
1231	T50N410B	3525	SOURCE INST. - T50R800A	DW	1	59700	T50R800A	
1232	T50N411A	3526	SOURCE INST. - T50R800B	DW	1	59700	T50R800B	
1233	T50N411B	3527	SOURCE INST. - T50R800A	DW	1	59506	T50R800A	
1234	T50N412A	3530	SOURCE INST. - T50R800A	DW	1	59700	T50R800A	
1235	T50N412B	3531	SOURCE INST. - T50R800B	DW	1	59700	T50R800B	
1236	T50N413A	8001	DIV 1 THERMOCOUPLE	DW	1	59700	T50R800A	
1237	T50N413B	3532	SOURCE INST. - T50R800B	DW	1	59700	T50R800B	
1238	T50N414A	8120	PCAM TORUS WIDE RANGE PR.TRANS	RB	1	589-1	H21P631A	
1239	T50N414B	8662	PCAM TORUS PR.XMTR.	RB	1	589-1	H21P631B	
1240	T50N415A	8121	PC DW WIDE RANGE PR.TRANSMITTE	RB	2	618-0	H21P595A	
1241	T50N415B	8663	PCPM DW PR.XMTR.	RB	2	618-0	H21P595B	
1242	T50N499A	8122	PCAM TORUS NARROW RANGE PR.TRN	RB	1	589-1	H21P631A	
1243	T50N499B	8664	PCAM TORUS PR.XMTR.	RB	1	589-1	H21P631B	
1244	T50R800A	8002	DW/TORUS TEMP RECORDER	AB	3	64306	H11P601	
1245	T50R800B	8003	DW/TORUS TEMP RECORDER	AB	3	64306	H11P602	
1246	T50R802A	8108	DW/TORUS PRESS RECORDER	AB	3	64306	H11P601	
1247	T50R802B	8109	DW/TORUS PRESS RECORDER	AB	3	64306	H11P602	
1248	T50R804A	3230	TORUS LEVEL INDICATION	AB	3	64306	H11P601	
1249	T50R804B	3231	TORUS LEVEL INDICATION	AB	3	64306	H11P602	
1250	X4103C001	1627	EDG 11 HAVE FAN	RHR	2	617 0		RHR2-13

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
1251	X4103C002	1628	EDG 11 HVAC FAN	RHR	2	61700		RHR2-13
1252	X4103C003	1676	EDG 12 HVAC FAN	RHR	2	617 0		RHR2-13
1253	X4103C004	1677	EDG 12 HVAC FAN	RHR	2	61700		RHR2-13
1254	X4103C005	1711	EDG13 HVAC FAN	RHR	2	617 0		RHR2-13
1255	X4103C006	1712	EDG13 HVAC FAN	RHR	2	617 0		RHR2-13
1256	X4103C007	1760	EDG 14 HVAC FAN	RHR	2	617 0		RHR2-13
1257	X4103C008	1761	EDG 14 HVAC FAN	RHR	2	61700		RHR2-13
1258	X4103C009	1603	EDG1 HVAC FAN	RHR	2	617 0		RHR2-14
1259	X4103C010	1607	EDG1 HVAC FAN	RHR	2	617 0		RHR2-14
1260	X4103C011	1654	EDG 12 HVAC FAN	RHR	2	[illegible]17 0		RHR2-14
1261	X4103C012	1658	EDG 12 HVAC FAN	RHR	2	[illegible]		RHR2-14
1262	X4103C013	1687	EDG 13 HVAC FAN	RHR	2	[illegible]		RHR2-14
1263	X4103C014	1691	EDG 13 HVAC FAN	RHR	2	617 0		RHR2-14
1264	X4103C015	1737	EDG 14 HVAC FAN	RHR	2	617 0		RHR2-14
1265	X4103C016	1741	EDG 14 HVAC FAN	RHR	2	617 0		RHR2-14
1266	X4103C017	1638	RHR COMPLEX HVAC FAN	RHR	1	590 0		RHR1-18
1267	X4103C018	1642	RHR COMPLEX HVAC FAN	RHR	1	59000		RHR1-18
1268	X4103C019	1722	RHR COMPLEX HVAC FAN	RHR	1	590 0		RHR1-18
1269	X4103C020	1726	RHR COMPLEX HVAC FAN	RHR	1	590 0		RHR1-18
1270	X4103F101	8665	RHRHVAC EXH.GRAV.DMPR.	RHR	2			RHR2-16
1271	X4103F102	8666	RHRHVAC EXH.GRAV.DMPR.	RHR	2			RHR2-16
1272	X4103F103	1600	AIR INTAKE MO DAMPER	RHR	2	632 0		RHR2-10
1273	X4103F104	1611	SWG RECIR MO DAMPER	RHR	2	632 0		RHR2-10
1274	X4103F106	1604	FAN DISCHARGE MO DAMPER	RHR	2	629 0		RHR2-12
1275	X4103F108	1608	FAN DISCHARGE MO DAMPER	RHR	2	625 0		RHR2-12
1276	X4103F109	1614	SWCR EXHAUST MO DAMPER	RHR	2	632 0		RHR2-12
1277	X4103F113	8667	RHRHVAC EXH.GRAV.DMPR.	RHR	2			RHR2-16
1278	X4103F114	8668	RHRHVAC EXH.GRAV.DMPR.	RHR	2			RHR2-16
1279	X4103F115	1651	AIR INTAKE MO DAMPER	RHR	2	623 0		RHR2-10
1280	X4103F116	1662	SWG RECIR MO DAMPER	RHR	2	623 0		RHR2-10
1281	X4103F118	1655	FAN DISCHARGE MO DAMPER	RHR	2	623 0		RHR2-12
1282	X4103F120	1659	FAN DISCHARGE MO DAMPER	RHR	2	623 6		RHR2-12
1283	X4103F121	1665	SWGR EXHAUST MO DAMPER	RHR	2	626 0		RHR2-12
1284	X4103F125	8669	RHRHVAC EXH.GRAV.DMPR.	RHR	2			RHR2-16
1285	X4103F126	8670	RHRHVAC EXH.GRAV.DMPR.	RHR				RHR2-16
1286	X4103F127	1684	AIR INTAKE MO DAMPER	RHR	2	630 0		RHR2-10
1287	X4103F128	1695	SWG RECIRC MO DAMPER	RHR	2	632 0		RHR2-10
1288	X4103F130	1688	FAN DISCHARGE MO DAMPER	RHR	2	625 0		RHR2-12
1289	X4103F132	1692	FAN DISCHARGE MO DAMPER	RHR	2	625 0		RHR2-12
1290	X4103F133	1698	SWGR EXHAUST MO DAMPER	RHR	2	627 0		RHR2-12
1291	X4103F137	8671	RHRHVAC EXH.GRAV.DMPR.	RHR	2			RHR2-16
1292	X4103F138	8672	RHRHVAC EXH.GRAV.DMPR.	RHR	2			RHR2-16
1293	X4103F139	1734	AIR INTAKE MO DAMPER	RHR	2	632 0		RHR2-10
1294	X4103F140	1745	SWG RECIRC MO DAMPER	RHR	2	632 0		RHR2-10
1295	X4103F142	1738	FAN DISCHARGE MO DAMPER	RHR	2	625 0		RHR2-12
1296	X4103F144	1742	FAN DISCHARGE MO DAMPER	RHR	2	62410		RHR2-12
1297	X4103F145	1748	SWGR EXHAUST MO DAMPER	RHR	2	630 0		RHR2-12
1298	X4103F149A	1619	AIR INTAKE MO DAMPER	RHR	2	629 0		RHR2-11
1299	X4103F149B	1620	AIR INTAKE MO DAMPER	RHR	2	62900		RHR2-11
1300	X4103F149C	1621	AIR INTAKE MO DAMPER	RHR	2	629 0		RHR2-11

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
1301	X4103F149D	1622	AIR INTAKE MO DAMPER	RHR	2	62900		RHR2-11
1302	X4103F150	1632	EDG RM RECIR MO DAMPER	RHR	2	626 0		RHR2-10
1303	X4103F151A	1668	AIR INTAKE MO DAMPER	RHR	2	632 0		RHR2-11
1304	X4103F151B	1669	AIR INTAKE MO DAMPER	RHR	2	63100		RHR2-11
1305	X4103F151C	1670	AIR INTAKE MO DAMPER	RHR	2	632 0		RHR2-11
1306	X4103F151D	1671	AIR INTAKE MO DAMPER	RHR	2	63100		RHR2-11
1307	X4103F152	1678	EDG RM RECIR MO DAMPER	RHR	2	62500		RHR2-10
1308	X4103F153A	1703	AIR INTAKE MO DAMPER	RHR	2	624 0		RHR2-11
1309	X4103F153B	1704	AIR INTAKE MO DAMPER	RHR	2	62700		RHR2-11
1310	X4103F153C	1705	AIR INTAKE MO DAMPER	RHR	2	624 0		RHR2-11
1311	X4103F153D	1706	AIR INTAKE MO DAMPER	RHR	2	62700		RHR2-11
1312	X4103F154	1713	EDG RECIR MO DAMPER	RHR	2	630 0		RHR2-11
1313	X4103F155A	1753	AIR INTAKE MO DAMPER	RHR	2	623 0		RHR2-11
1314	X4103F155B	1754	AIR INTAKE MO DAMPER	RHR	2	62300		RHR2-11
1315	X4103F155C	1755	AIR INTAKE MO DAMPER	RHR	2	623 0		RHR2-11
1316	X4103F155D	1756	AIR INTAKE MO DAMPER	RHR	2	62300		RHR2-11
1317	X4103F156	1762	EDG RM RECIR MO DAMPER	RHR	2	630 0		RHR2-10
1318	X4103F157	1635	AIR INTAKE MO DAMPER	RHR	1	601 0		RHR1-16
1319	X4103F158	8673	RHRHVAC VENT.FAN DMP.	RHR	1			RHR1-24
1320	X4103F159	1639	FAN DISCHARGE MO DAMPER	RHR	1	592 0		RHR1-17
1321	X4103F160	8674	RHRHVAC VENT.FAN DMP.	RHR	1			RHR1-24
1322	X4103F161	1643	FAN DISCHARGE MO DAMPER	RHR	1	595 0		RHR1-17
1323	X4103F162	1648	RHR COMPLEX RECIR MO DAMPER	RHR	1	601 0		RHR1-16
1324	X4103F163	8675	RHRHVAC EXH.GRAV.DMP.	RHR	1			RHR2-16
1325	X4103F164	1719	AIR INTAKE MO DAMPER	RHR	1	605 0		RHR1-16
1326	X4103F165	8676	RHRHVAC VENT.FAN DMP.	RHR	1			RHR1-24
1327	X4103F166	1723	FAN DISCHARGE MO DAMPER	RHR	1	600 0		RHR1-17
1328	X4103F167	8677	RHRHVAC VENT.FAN DMP.	RHR	1			RHR1-24
1329	X4103F168	1768	FAN DISCHARGE MO DAMPER	RHR	1	593 3		RHR1-17
1330	X4103F169	1729	PUMP RM RECIR MO DAMPER	RHR	1	600 0		RHR1-16
1331	X4103F170	8678	RHRHVAC EXH.GRAV.DMP.	RHR	1			RHR2-16
1332	X4103F171	8679	RHRHVAC EXH.FIRE DMP.	RHR	2	625-6		
1333	X4103F172	8680	RHRHVAC INL.FIRE DMP.	RHR	2	625-6		
1334	X4103F173	8681	RHRHVAC VENT.FAN DMP.	RHR	2	625-6		
1335	X4103F177	8682	RHRHVAC EXH.FIRE DMP.	RHR	2			
1336	X4103F178	8683	RHRHVAC INL.FIRE DMP.	RHR	2			
1337	X4103F179	8684	RHRHVAC VENT.FAN DMP.	RHR	2			
1338	X4103F183	8685	RHRHVAC EXH.FIRE DMP.	RHR	2			
1339	X4103F184	8686	RHRHVAC INL.FIRE DMP.	RHR	2			
1340	X4103F185	8687	RHRHVAC VENT.FAN DMP.	RHR	2			
1341	X4103F189	8688	RHRHVAC EXH.FIRE DMP.	RHR	2			
1342	X4103F190	8689	RHRHVAC INL.FIRE DMP.	RHR	2			
1343	X4103F191	8690	RHRHVAC INL.FIRE DMP.	RHR	2			
1344	X41K001A	1618	SWGR TEMPERATURE ELEMENT	RHR	2	61700	H21P350	
1345	X41K001B	1702	SWGR TEMP CONTROL	RHR	2	61700	H21P352	
1346	X41K001C	1667	SWGR TEMPERATURE CONTROL	RHR	2	61700	H21P351	
1347	X41K001D	1752	SWGR TEMPERATURE CONTROL	RHR	2	61700	H21P353	
1348	X41K002A	1630	EDG RM TEMP CONTROL	RHR	2	61700	H21P350	
1349	X41K002B	1717	EDG RM TEMP CONTROL	RHR	2	61700	H21P352	
1350	X41K002C	1682	EDG RM TEMP CONTROL	RHR	2	61700	H21P351	

Table 3-3 Safe Shutdown Equipment List (SSEL)

No	PIS No	Line No	Description	Bldg	Flr	Elev	Mounted on	SEWS No
1351	X41K002D	1766	EDG RM TEMP CONTROL	RHR	2	61700	H21P353	
1352	X41K002E	1631	EDG RM TEMP CONTROL	RHR	2	61700	H21P350	
1353	X41K002F	1718	EDG RM TEMP CONTROL	RHR	2	61700	H21P352	
1354	X41K002G	1683	EDG RM TEMP CONTROL	RHR	2	61700	H21P351	
1355	X41K002H	1767	EDG RM TEMP CONTROL	RHR	2	61700	H21P353	
1356	X41K003A	1647	PUMP RM TEMP CONTROL	RHR	1	59000	H21P517	
1357	X41K003B	1733	RHR COMPLEX TEMP CONTROL	RHR	1	59000	H21P518	
1358	X41N056A	1617	SWGR TEMPERATURE ELEMENT	RHR	2	62200	EQUIP	RHR2-15
1359	X41N056B	1701	SWGR TEMP ELEMENT	RHR	2	62200	EQUIP	RHR2-15
1360	X41N056C	1666	SWGR TEMP ELEMENT	RHR	2	62200	EQUIP	RHR2-15
1361	X41N056D	1751	SWGR TEMPERATURE ELEMENT	RHR	2	62200	EQUIP	RHR2-15
1362	X41N057A	1629	EDG RN TEMP ELEMENT	RHR	2	62200	EQUIP	RHR2-15
1363	X41N057B	1716	EDG RM TEMP ELEMENT	RHR	2	62200	EQUIP	RHR2-15
1364	X41N057C	1681	EDG RM TEMP ELEMENT	RHR	2	62200	EQUIP	RHR2-15
1365	X41N057D	1765	EDG RM TEMP ELEMENT	RHR	2	62200	EQUIP	RHR2-15
1366	X41N058A	1646	PUMP RM TEMP ELEMENT	RHR	1	59500	EQUIP	RHR1-25
1367	X41N058B	1732	RHR COMPLEX TEMP ELEMENT	RHR	1	59500	EQUIP	RHR1-25

Table 3-4 Plant System and Corresponding SSEL Line Numbers

Plant System/Function	SSEL Line Numbers
EECW	1–627
RHRSW, EDGSW, EESW	700–832
DC Power	900–925
NIAS	1000–1371
EDG	1400–1499, 9000–9019
HVAC	1500–1546
EDG HVAC	1600–1775
MCCs, relay and distribution panels	1800–1891
System instrumentation	1900–1973
RS/RPS/CRD	2000–2019
Reactor recirc. instrumentation	2100–2130
EDG permissives	2200–2520
ADS	3000–3073
B21 instrumentation	3100–3302
Additional instruments, instrument racks	3500–3656
HPCI instrumentation	3900–3944
HPCI	4000–4177
RCIC	4200–4325
RHR	5000–5203
RHR instrumentation	5300–5322
Shutdown cooling	6000–6035
Torus cooling	6200–6219
Containment isolation	7000–7057
Additional misc. items	8000–8012
Components from Ops. review	8100–8132
Additional dependency instruments	8500–8692

Table 3-5 Normally Open Valves Which Close on Receipt of Isolation Signal

Valve	Isolation Signal	Closes on loss of power?
B2103F022A	Reactor Vessel Low Level 1	yes
B2103F028A	Reactor Vessel Low Level 1	yes
B2103F022B	Reactor Vessel Low Level 1	yes
B2103F028B	Reactor Vessel Low Level 1	yes
B2103F022C	Reactor Vessel Low Level 1	yes
B2103F028C	Reactor Vessel Low Level 1	yes
B2103F022D	Reactor Vessel Low Level 1	yes
B2103F028D	Reactor Vessel Low Level 1	yes
G1154F600	Reactor Vessel Low Level 2 High Drywell Pressure	no
G1154F018	Reactor Vessel Low Level 2 High Drywell Pressure	no
T4901F601	Reactor Vessel Low Level 2 High Drywell Pressure	no
T4901F465	Reactor Vessel Low Level 2 High Drywell Pressure	yes
T4800F455	Reactor Vessel Low Level 2 High Drywell Pressure Reactor Building Exhaust Radiation High	yes
T4800F454	Reactor Vessel Low Level 2 High Drywell Pressure Reactor Building Exhaust Radiation High	yes
T4800F453	Reactor Vessel Low Level 2 High Drywell Pressure Reactor Building Exhaust Radiation High	yes
T4901F468	Reactor Vessel Low Level 2 High Drywell Pressure	no
T4901F602	Reactor Vessel Low Level 2 High Drywell Pressure	no
G3352F001	Reactor Vessel Low Level 2	no
G3352F004	Reactor Vessel Low Level 2	no
T5000F450	Reactor Vessel Low Level 2 High Drywell Pressure	yes
T5000F456	Reactor Vessel Low Level 2 High Drywell Pressure	yes
B3100F014B	Reactor Vessel Low Level 2 High Drywell Pressure	yes
B3100F016B	Reactor Vessel Low Level 2 High Drywell Pressure	yes
B3100F014A	Reactor Vessel Low Level 2 High Drywell Pressure	yes

Table 3-5 (continued) Normally Open Valves Which Close on Receipt of Isolation Signal

Valve	Isolation Signal	Closes on loss of power?
B3100F016A	Reactor Vessel Low Level 2 High Drywell Pressure	yes
T4800F456	Reactor Vessel Low Level 2 High Drywell Pressure Reactor Building Exhaust Radiation High	yes
T4800F457	Reactor Vessel Low Level 2 High Drywell Pressure Reactor Building Exhaust Radiation High	yes
T4800F458	Reactor Vessel Low Level 2 High Drywell Pressure Reactor Building Exhaust Radiation High	yes
G5100F605	Reactor Vessel Low Level 2 High Drywell Pressure	no
G5100F604	Reactor Vessel Low Level 2 High Drywell Pressure	no
E5150F062	High Drywell Pressure RCIC Steam Line Low Pressure	no
E5150F084	High Drywell Pressure RCIC Steam Line Low Pressure	no
E4150F075	High Drywell Pressure HPCI Steam Line Low Pressure	no
E4150F079	High Drywell Pressure HPCI Steam Line Low Pressure	no
G5100F601	Reactor Vessel Low Level 2 High Drywell Pressure	no
G5100F600	Reactor Vessel Low Level 2 High Drywell Pressure	no
G5100F602	Reactor Vessel Low Level 2 High Drywell Pressure	no
G5100F603	Reactor Vessel Low Level 2 High Drywell Pressure	no
T5000F451	Reactor Vessel Low Level 2 High Drywell Pressure	yes
T5000F455	Reactor Vessel Low Level 2 High Drywell Pressure	yes
G5100F606	Reactor Vessel Low Level 2 High Drywell Pressure	no
G5100F607	Reactor Vessel Low Level 2 High Drywell Pressure	no

Table 3-6 List of Equipment Categories Including Number of SSEL Items and Outliers in Each Category

Category No.	Description	SSEL Items	Outliers
1	motor control centers	15	6
2	low voltage switchgear	8	4
3	medium voltage switchgear	8	3
4	transformers and regulators	12	3
5	horizontal pumps	21	3
6	vertical pumps	16	1
7	fluid-operated valves	281	14
8	motor-operated valves	89	12
9	fans	26	0
10	air handlers	16	3
11	chillers	2	1
12	air compressors	6	0
13	motor generators	0	0
14	distribution panels	20	3
15	batteries and racks	2	1
16	battery chargers	6	2
17	engine generators	4	0
18	automatic transfer switches	0	0
19	instrument racks	50	8
20	local instruments/ temperature sensors	38	1
21	control and instrumentation cabinets	83	18
22A	other valves	1119	2
22B	tanks	413	3
22C	heat exchangers	20	4
22D	steam-driven turbines	2	0

Table 3-7 List of Maintenance Work Requests

No.	WR No	Date Initiated	Description	Status/ schedule	SEWS No
1	000Z947541	11/3/94	Mounting bolt missing on HCU	Complete 12/9/94	RB1-12
2	000Z947542	11/3/94	HCU N2 tank support rod bent	Complete 12/20/94	RB1-12
3	000Z947543	11/3/94	Missing/loose hardware and straps on HCUs	Complete 12/9/94	RB1-12
4	000Z947544	11/3/94	Missing safety cables for lights- RB1	Complete 11/19/94	RB1-15/18
5	000Z948669	12/6/94	Missing hardware- H21P021	Complete 12/9/94	RBB-02
6	000Z951276	1/27/95	Missing hardware- H21P017	Complete 6/8/95	RCIC-15
7	000Z951277	1/27/95	T4100B018 anchor nut may be missing	Verified 2/24/95	RBSB-02
8	000Z951289	2/24/95	Missing bolts- HPCI main oil pump	Complete 3/29/95	HPCI-23
9	000Z951305	4/3/95	Missing bolts- calvert bus box support	Complete 5/18/95	SGR1-05
10	000Z951311	4/27/95	Missing screws on H11P614 insert	Scheduled 4/6/96	RR-18
11	000Z951312	4/27/95	Missing overhead light safety cables- AB3	Complete 3/19/96	BAT-06
12	000Z951313	4/27/95	Missing emergency battery hooks	Scheduled 4/26/96	RB2-22
13	000Z951314	4/27/95	Missing bolts between MCC R3200S016 sections	Complete 7/24/95	BAT-03
14	000Z951315	4/27/95	Switchgear door bolts not engaged/ stripped	canceled, split into 5	SGR2-03
15	000Z951324	5/15/95	GEMAC modules not inserted in H11P612	Scheduled for RF05	RR-03
16	000Z951325	5/16/95	Missing overhead lights safety cables- AB2	Complete 3/96	SGR1-01
17	000Z952656	5/17/95	GEMAC modules not inserted in H11P613	Scheduled for RF05	RR-03
18	000Z953607	5/31/95	Mounting screws for Dwyer switch in H21P296B	Complete 7/25/95	AB5-02
19	000Z953608	5/31/95	Safety cables on overhead lights- AB5	Complete 3/7/96	AB5-02
20	000Z953614	6/22/95	Mounting screw missing on R30NA09D switch	Complete 2/27/96	RHR1-08
21	000Z953615	6/22/95	Actuator lid loose on R3000F023D, EDG 14	Complete 2/27/96	RHR1-10
22	000Z953616	6/22/95	Emergency lighting battery hook bolts R3600S199	Work w/ 000Z951313	RHR1-08
23	000Z953617	6/27/95	EDG gauge panel mounting nut torque	Canceled, split into 4	RHR1-02
24	000Z953621	7/5/95	MCC R1600S003D rattlespace w/ water shield	Complete 9/26/95	RB2-22
25	000Z953622	7/5/95	Torque jam nuts on gauge panel R30P310	Complete 8/10/95	RHR1-02
26	000Z953623	7/5/95	Torque jam nuts on gauge panel R30P320	Complete 8/15/95	RHR1-02
27	000Z953624	7/5/95	Torque jam nuts on gauge panel R30P330	Complete 8/22/95	RHR1-02
28	000Z953625	7/5/95	Torque jam nuts on gauge panel R30P340	Complete 8/30/95	RHR1-02
29	000Z954328	7/11/95	Switchgear door bolts not engaged R1400S001B	Complete 7/24/95	SGR2-03
30	000Z954329	7/11/95	Switchgear door bolts not engaged R1400S001B	Complete 7/24/95	SGR2-03
31	000Z954330	7/11/95	Switchgear door bolts not engaged R1400S001C	Complete 7/24/95	SGR2-03
32	000Z954331	7/11/95	Switchgear door bolts not engaged R1400S001E	Complete 7/24/95	SGR2-03
33	000Z954332	7/11/95	Switchgear door bolts not engaged R1400S021B	Complete 7/24/95	SGR2-03
34	000Z952645	8/2/95	RHR switchgear bolts not engaged R1400S002A	Complete 8/8/95	RHR2-06
35	000Z952646	8/2/95	RHR switchgear bolts not engaged R1400S002B	Complete 8/8/95	RHR2-06
36	000Z952647	8/2/95	RHR switchgear bolts not engaged R1400S002C	Complete 8/8/95	RHR2-06
37	000Z952648	8/2/95	RHR switchgear bolts not engaged R1400S002D	Complete 8/8/95	RHR2-06
38	000Z952649	8/2/95	RHR switchgear bolts not engaged R1400S038	Complete 8/8/95	RHR2-04
39	000Z952650	8/2/95	RHR switchgear bolts not engaged R1400S039	Complete 8/8/95	RHR2-04
40	000Z955153	8/2/95	Relay mounting screws missing - H21P350	Complete 10/3/95	RHR2-07
41	000Z955154	8/2/95	Relay mounting screws missing - H21P351	Complete 2/9/96	RHR2-07
42	000Z955432	11/28/95	Anchor nut missing for Distrib. Pnl. H21P561	Scheduled 6/5/96	RBB-06
43	000Z957665	12/11/95	Missing hardware for sig. conditioners, H11P612	Scheduled 7/15/96	RR-03
44	000Z957666	12/11/95	Recorder hardware deficiencies in H11P601	Complete 2/14/96	CR-02
45	000Z957668	12/18/95	Trolleys on top of 480V switchgears R1400S022	Complete 2/15/96	SGR1-06
46	000Z957669	12/18/95	Trolleys on top of 480V switchgears R1400S023	Complete 2/15/96	SGR1-06
47	000Z957670	12/18/95	Trolleys on top of 480V switchgears R1400S020	Complete 2/22/96	SGR2-04
48	000Z957671	12/18/95	Trolleys on top of 480V switchgears R1400S021	Complete 2/22/96	SGR2-04
49	000Z957672	12/18/95	Trolleys on top of 480V switchgears R1400S036	Complete 2/15/96	RHR2-04
50	000Z957673	12/18/95	Trolleys on top of 480V switchgears R1400S037	Complete 2/15/96	RHR2-04
51	000Z957674	12/18/95	Trolleys on top of 480V switchgears R1400S038	Complete 2/22/96	RHR2-04
52	000Z957675	12/18/95	Trolleys on top of 480V switchgears R1400S039	Complete 2/22/96	RHR2-04
53	000Z957680	12/20/95	Mounting screw missing for PCV T41F114B	Scheduled 4/23/96	AB5-02
54	000Z957681	12/20/95	Mntg. hardware deficiencies for T41N132B & 134B	Scheduled 4/23/96	AB5-02
55	000Z957682	12/22/95	Tighten transformer mounting bolt - T41N456B	Scheduled 4/6/96	AB5-25

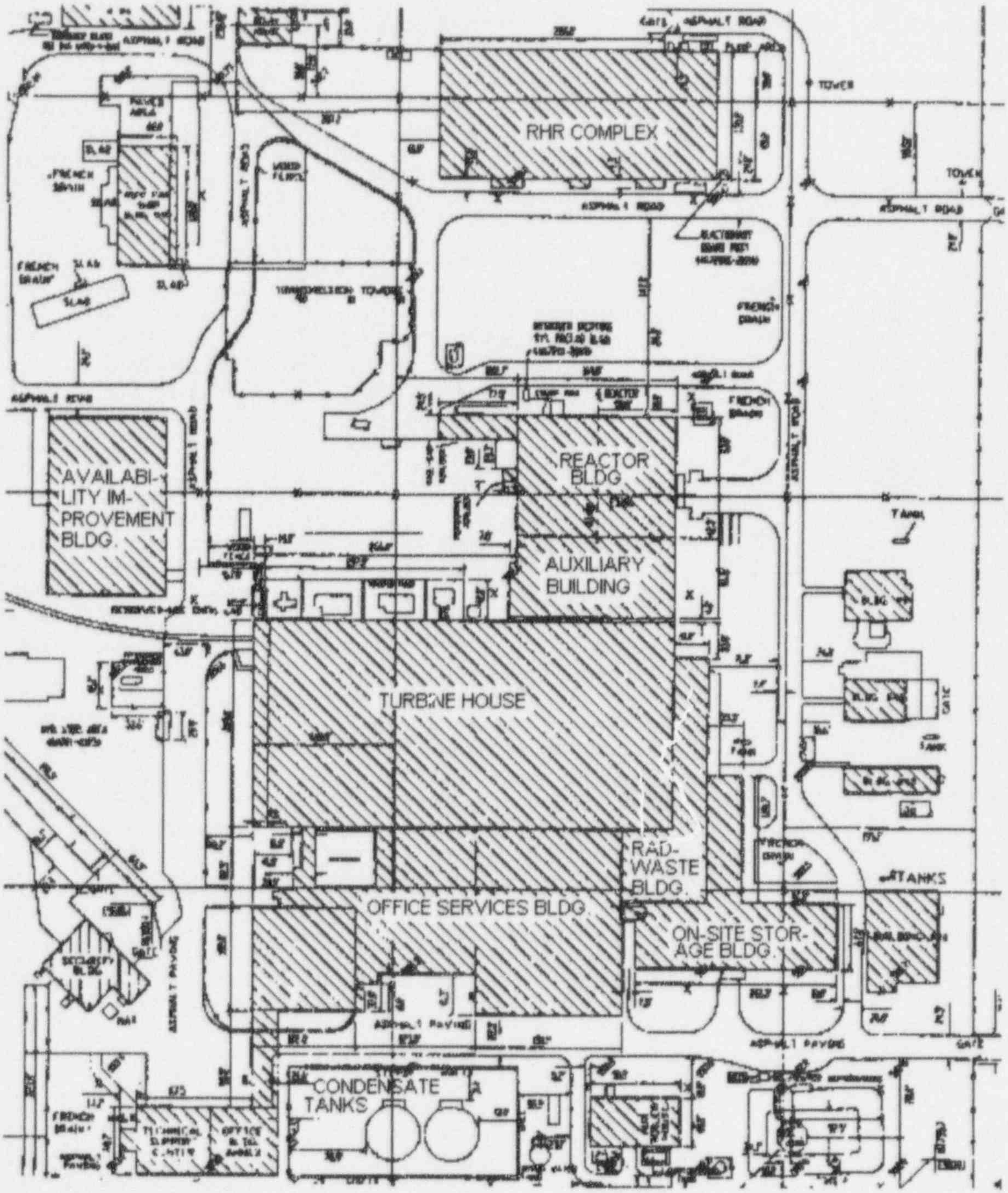


Figure 3-1 General Plant Building Arrangement

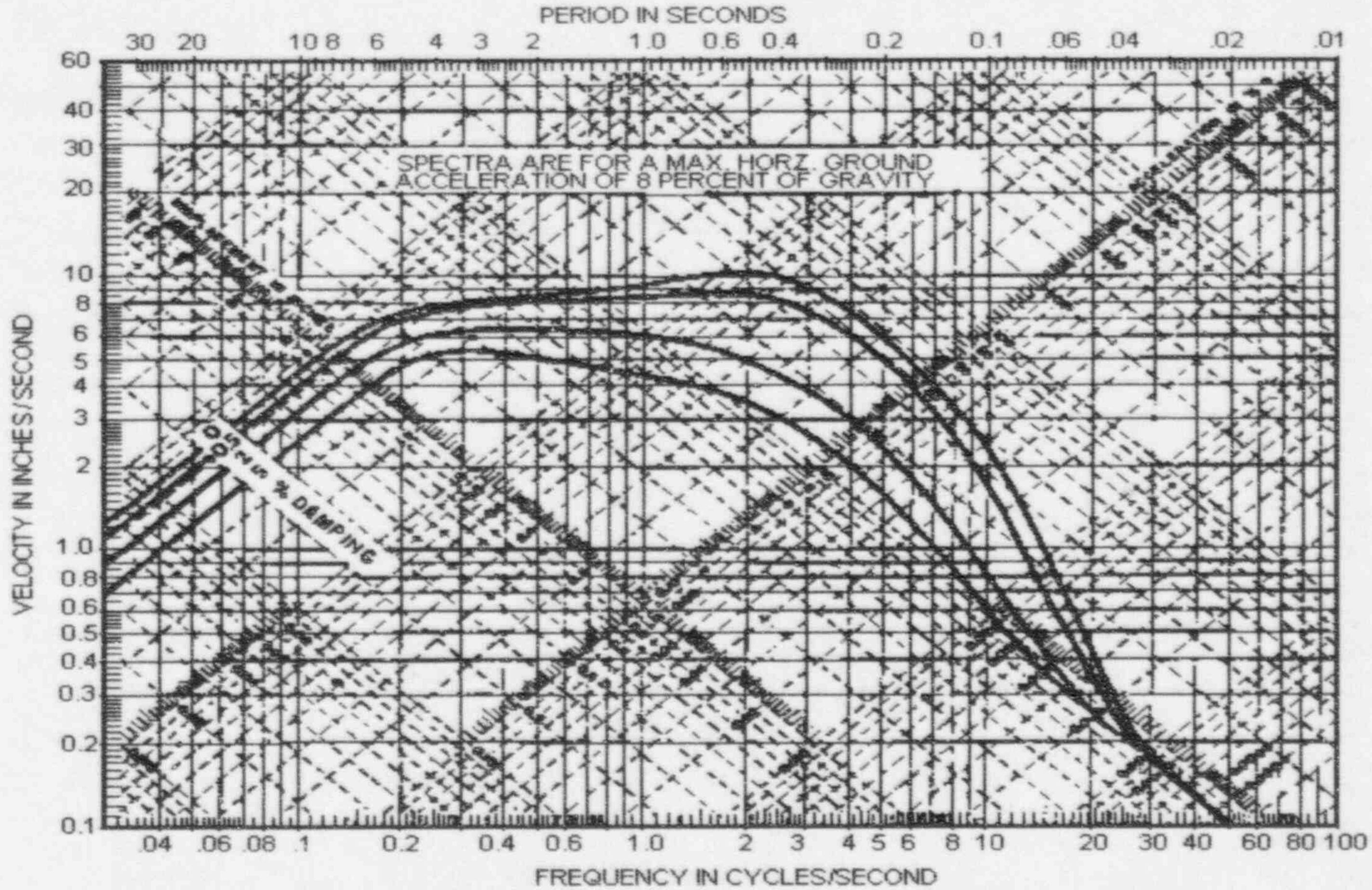


Figure 3-2 OBE Ground Response Spectra

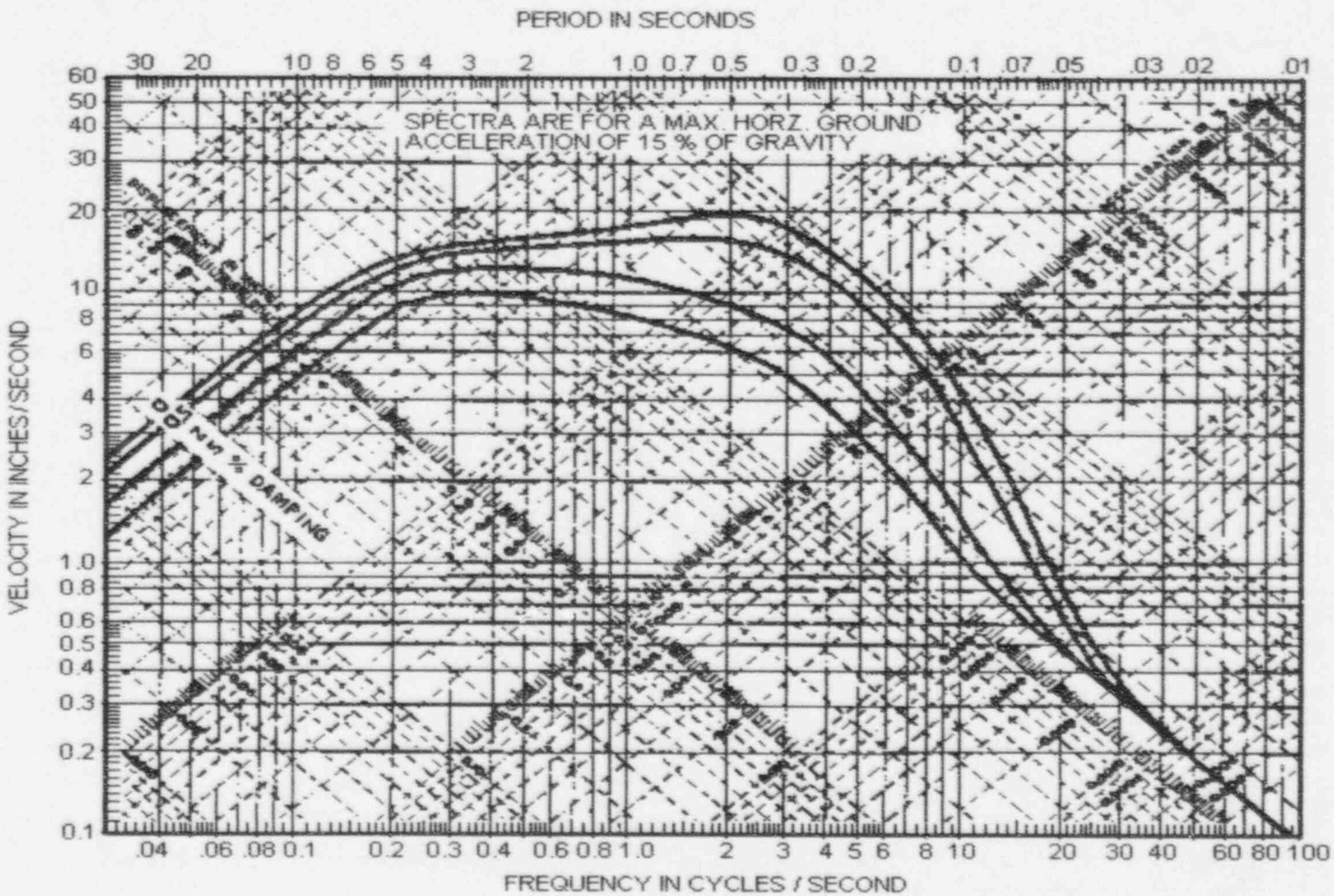


Figure 3-3 SSE Ground Response Spectra

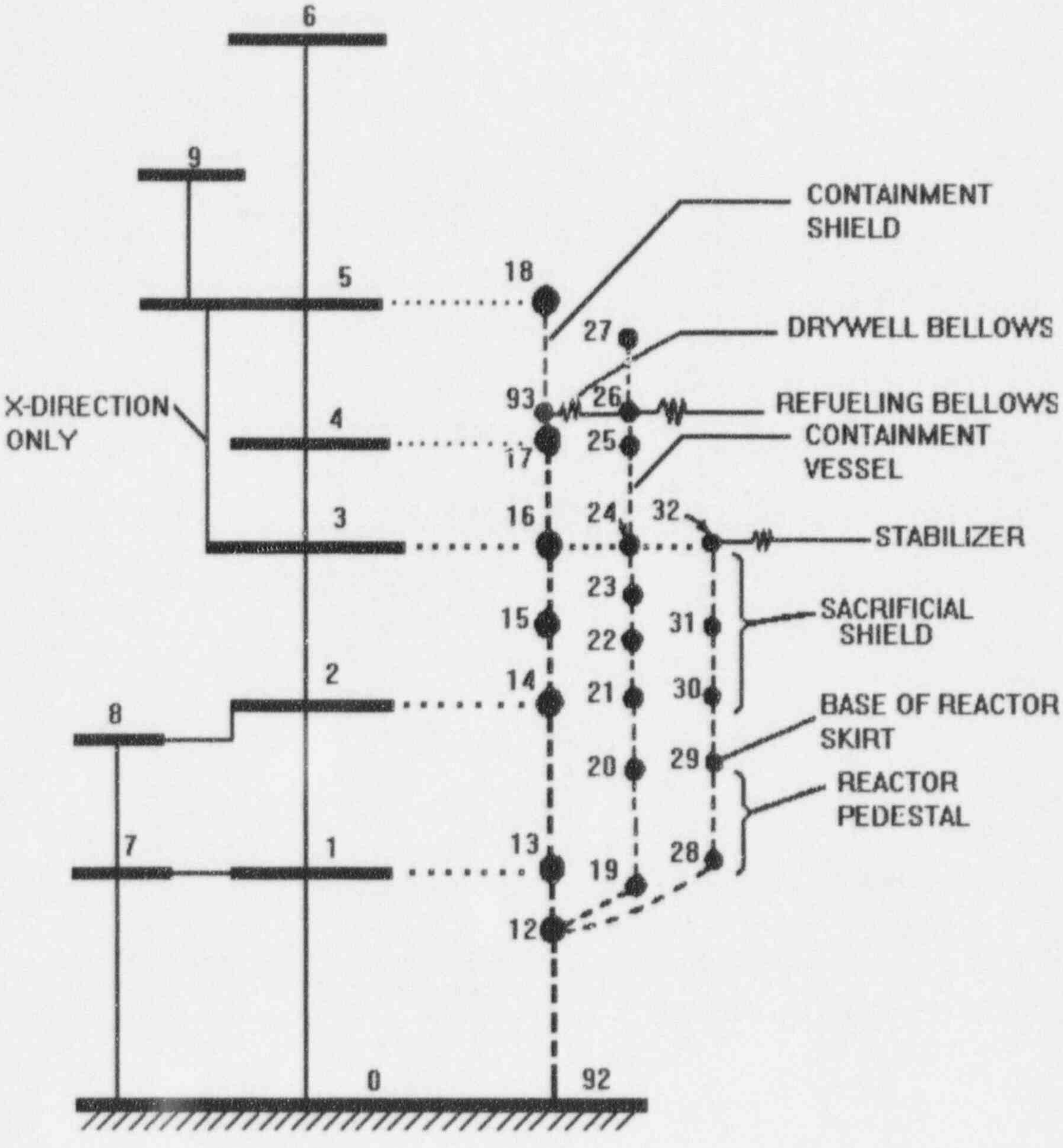


Figure 3-4 Reactor/Auxiliary Building Horizontal Dynamic Model

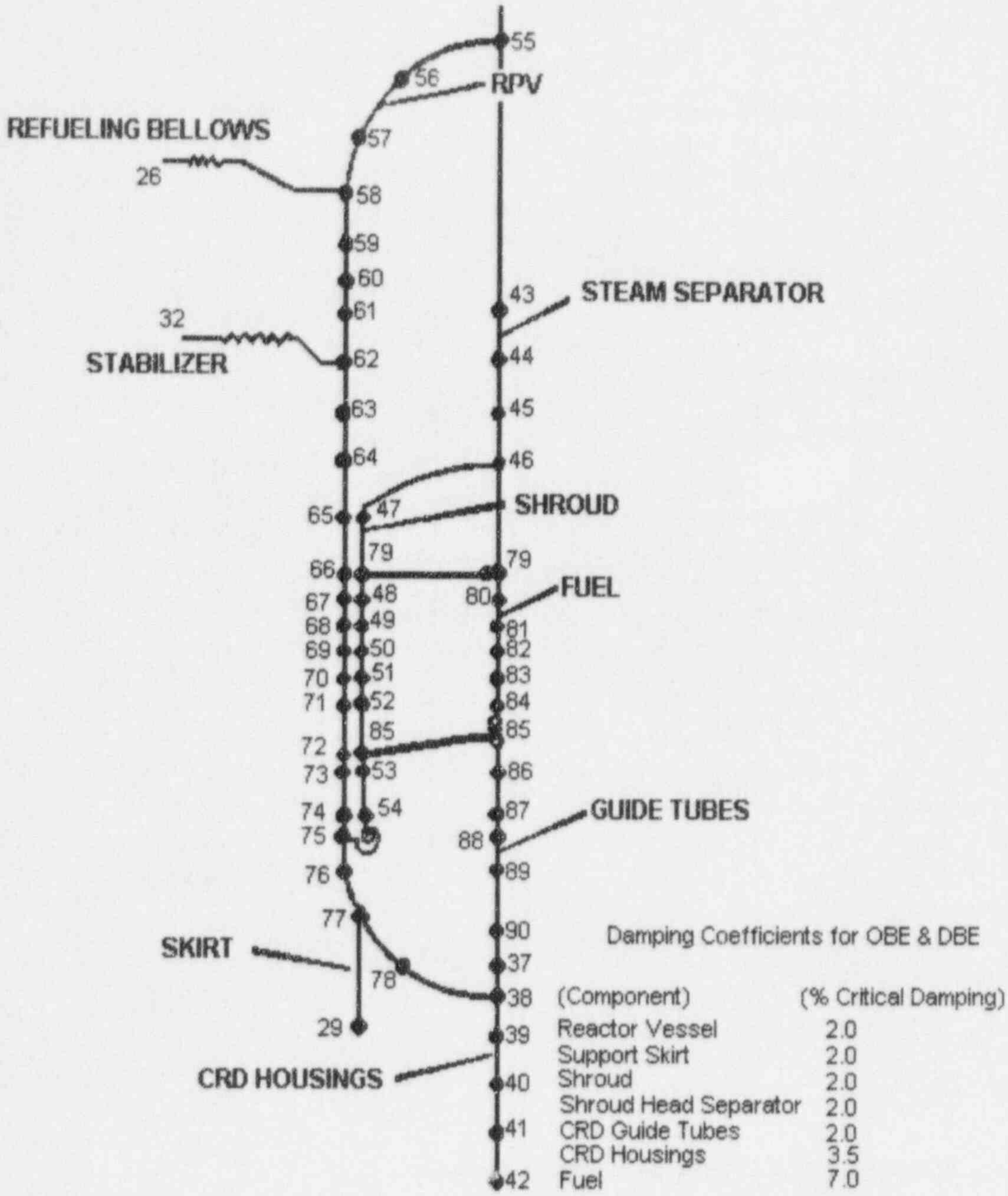


Figure 3-5 Reactor Pressure Vessel and Internals Dynamic Model

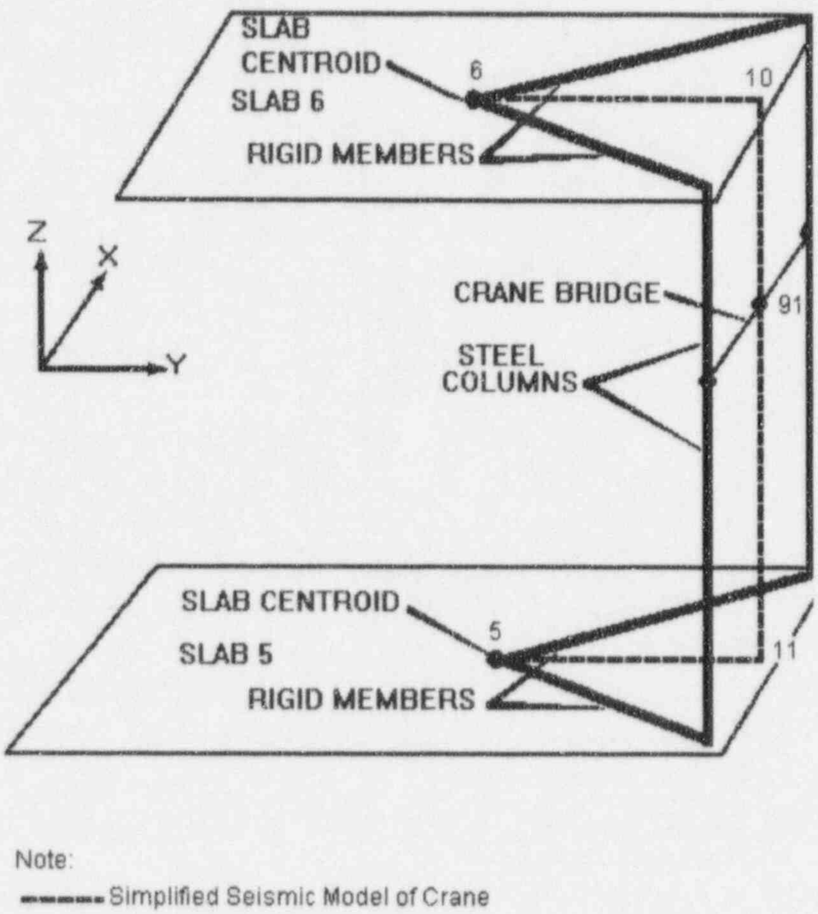


Figure 3-6 Reactor/Auxiliary Building Crane Dynamic Model

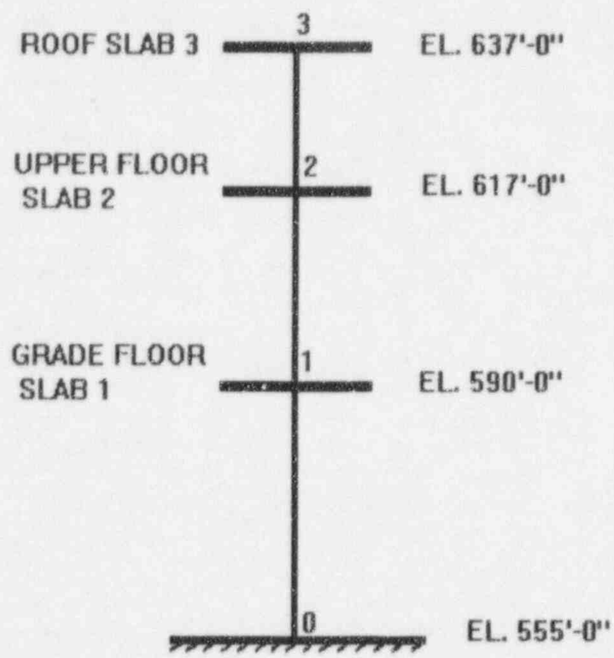


Figure 3-7 RHR Building Horizontal Dynamic Model

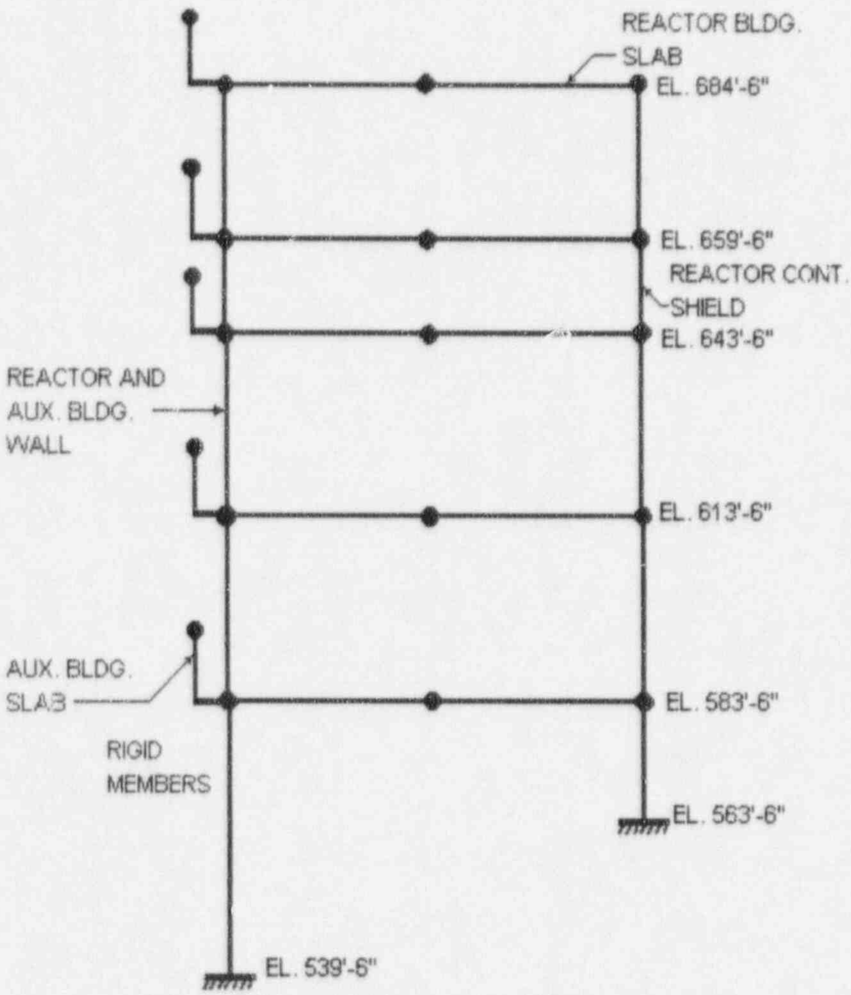


Figure 3-8 Reactor/Auxiliary Building Vertical Dynamic Model

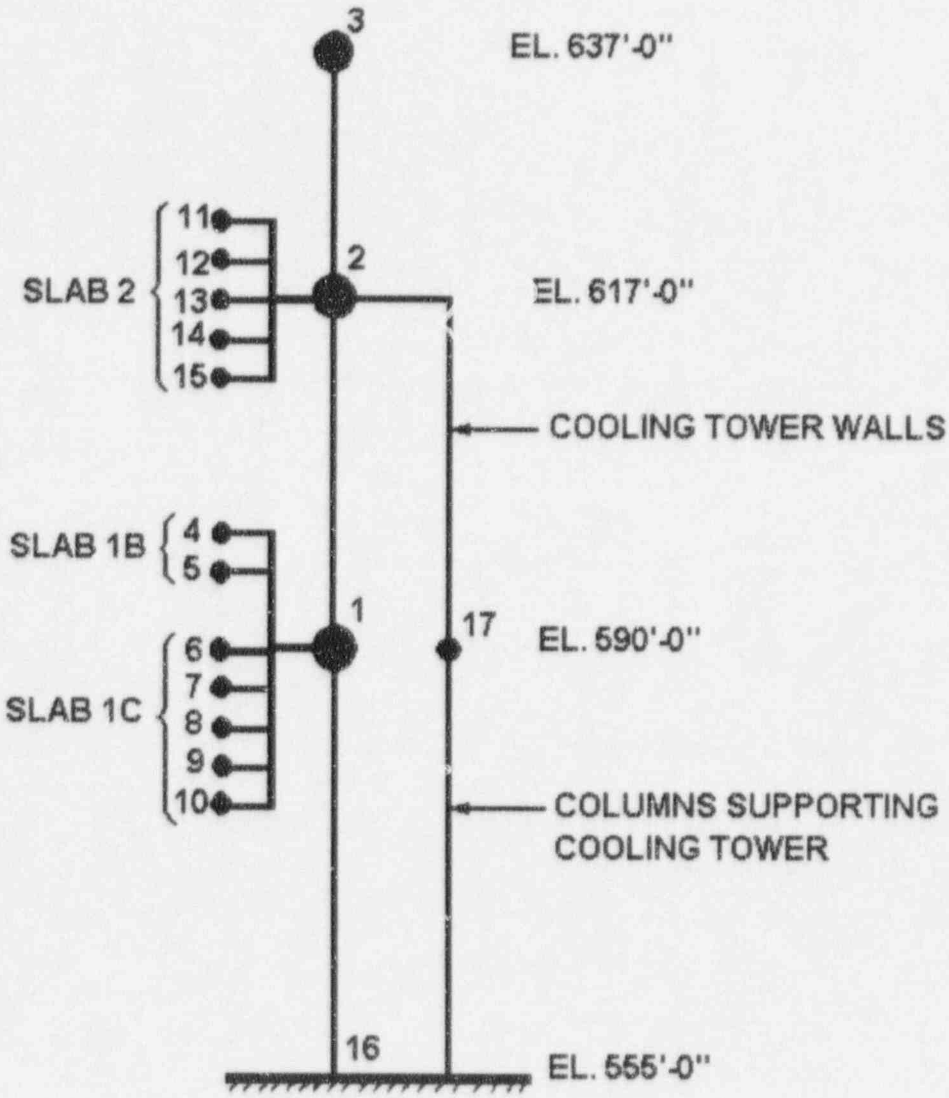


Figure 3-9 RHR Building Vertical Dynamic Model

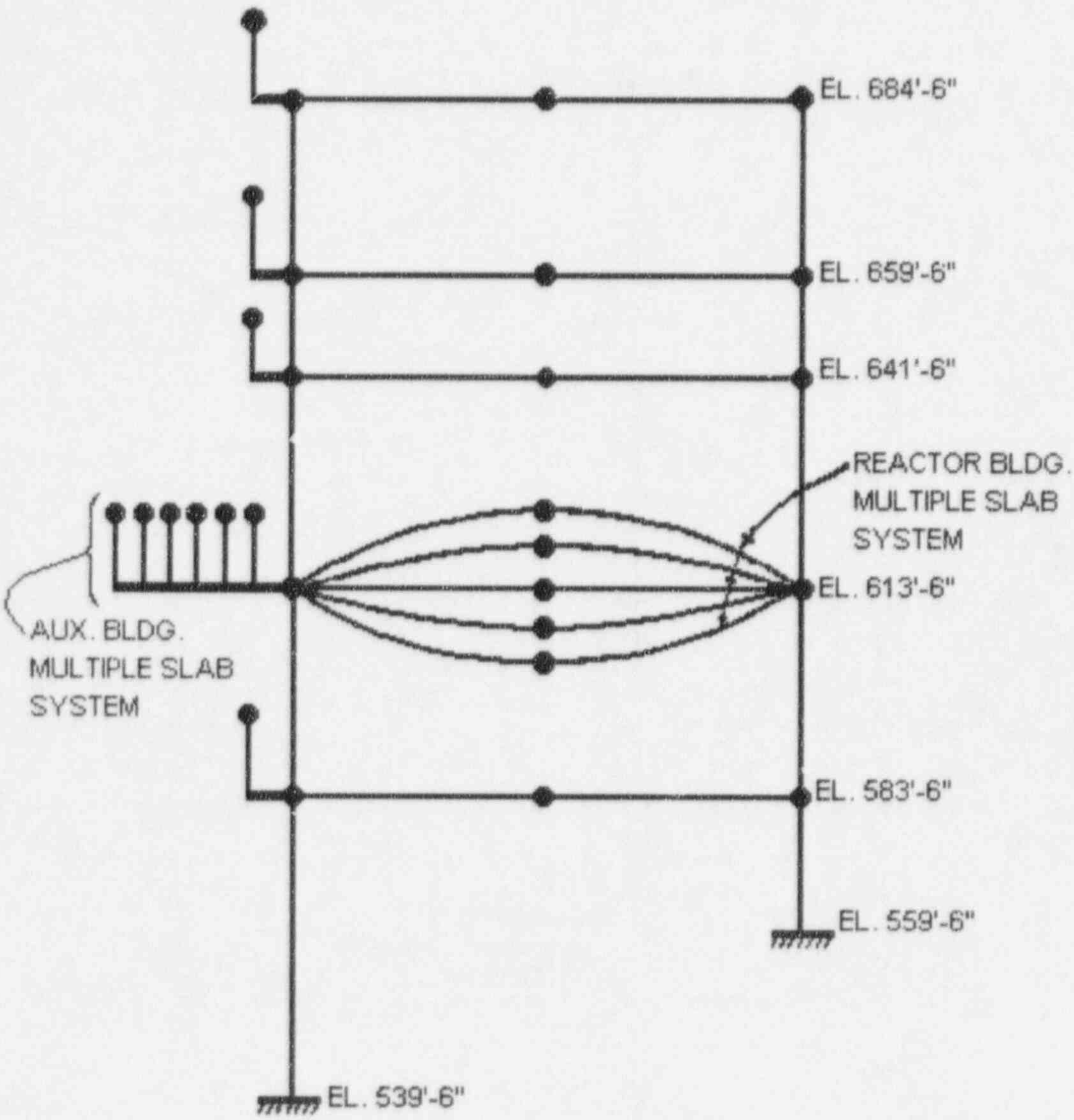


Figure 3-10 Reactor/Auxiliary Building Modified Dynamic Model at Elevation 613'-6"

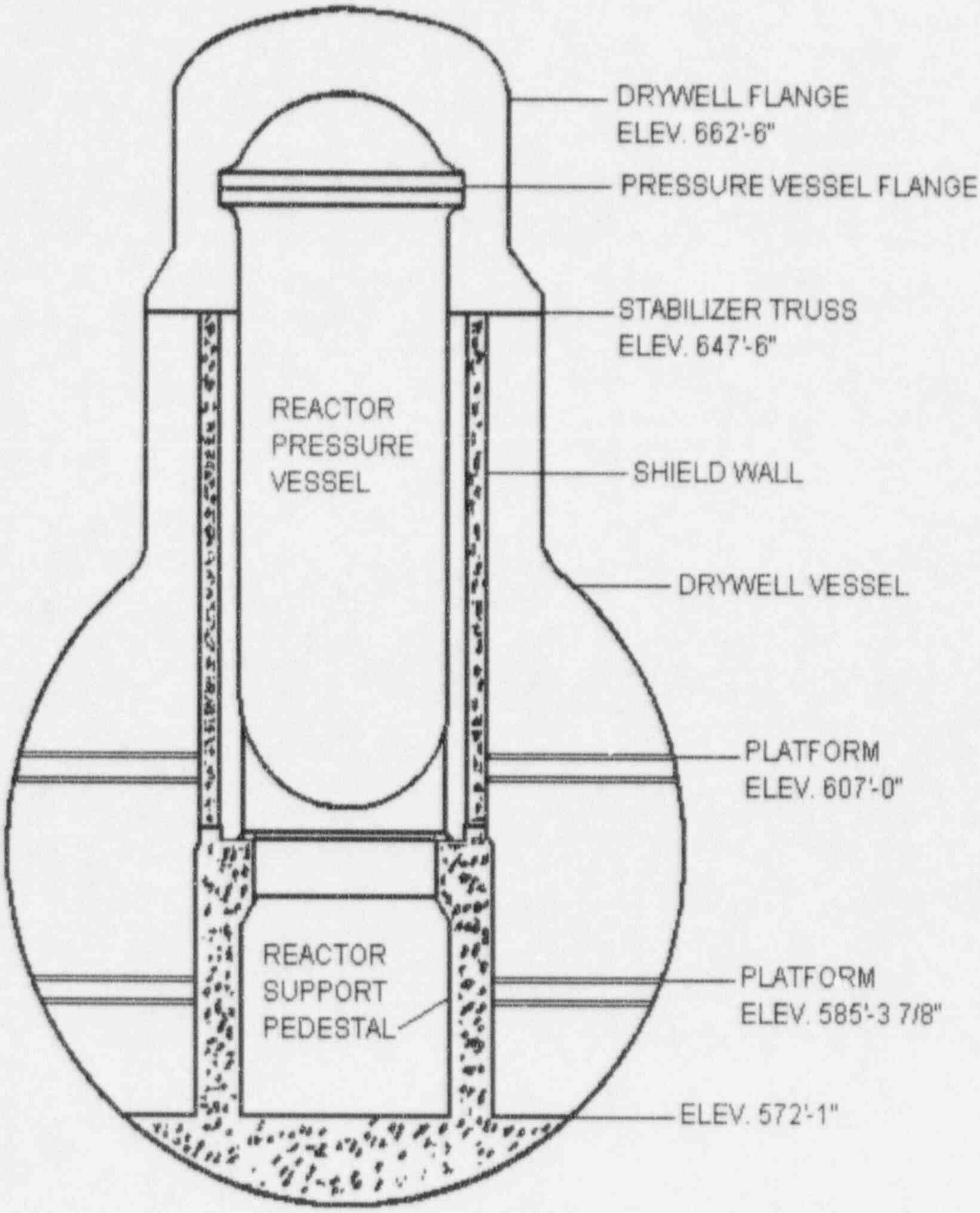


Figure 3-11 Section Through Centerline of Primary Containment

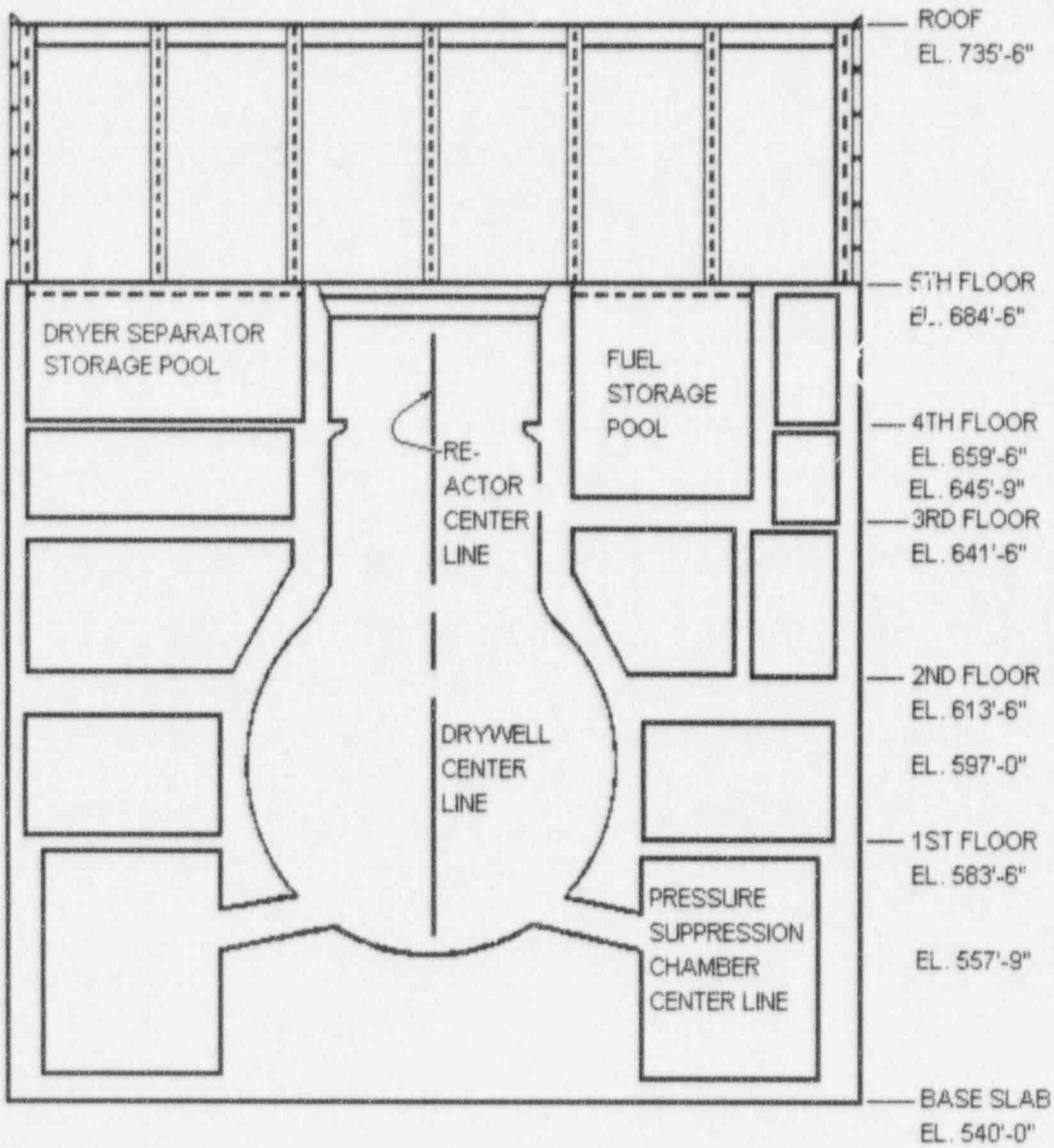


Figure 3-12 Reactor/Auxiliary Building North-South Section Through Centerline Looking West

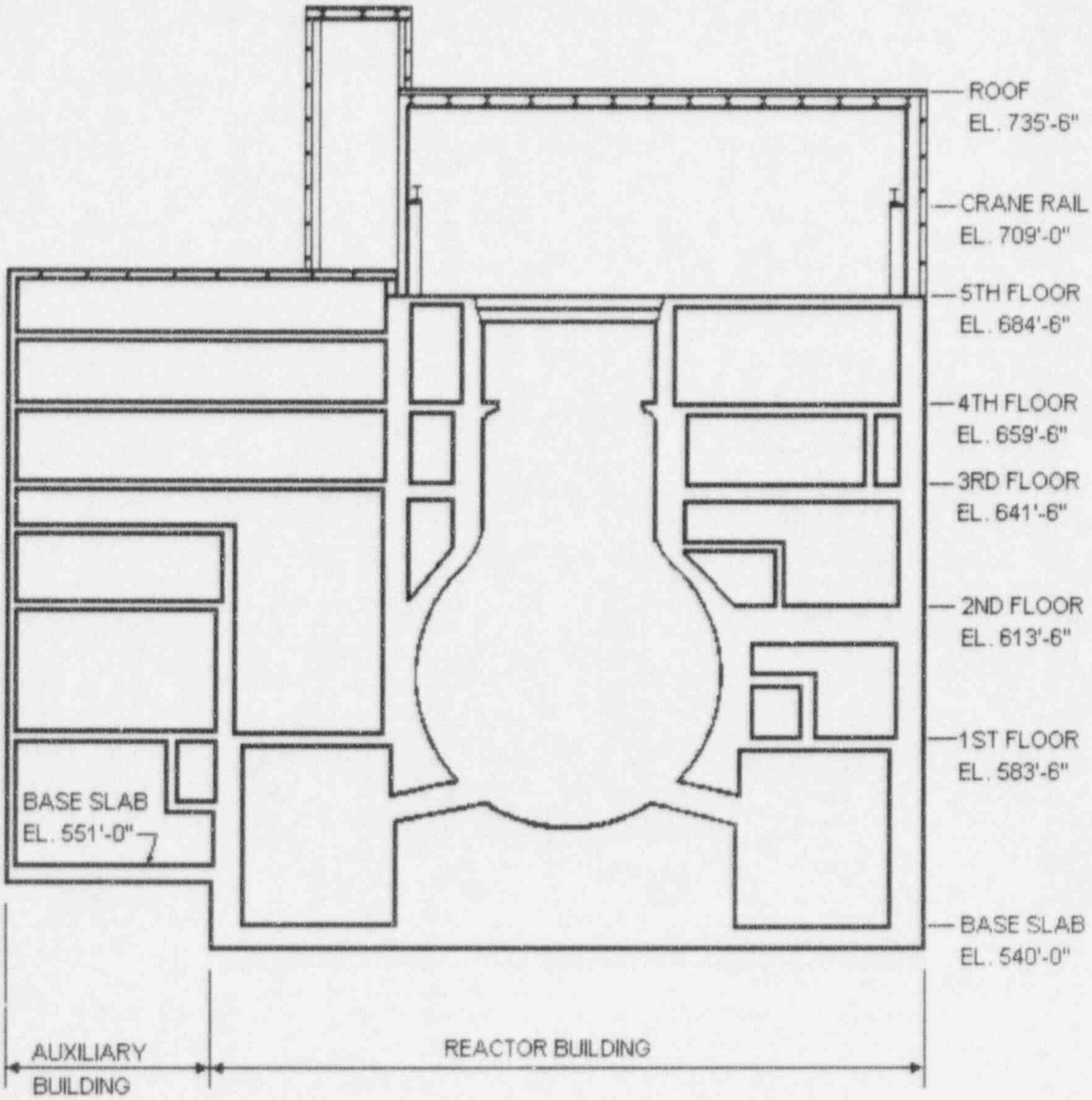


Figure 3-13 Reactor/Auxiliary Building East-West Section Through Centerline Looking South

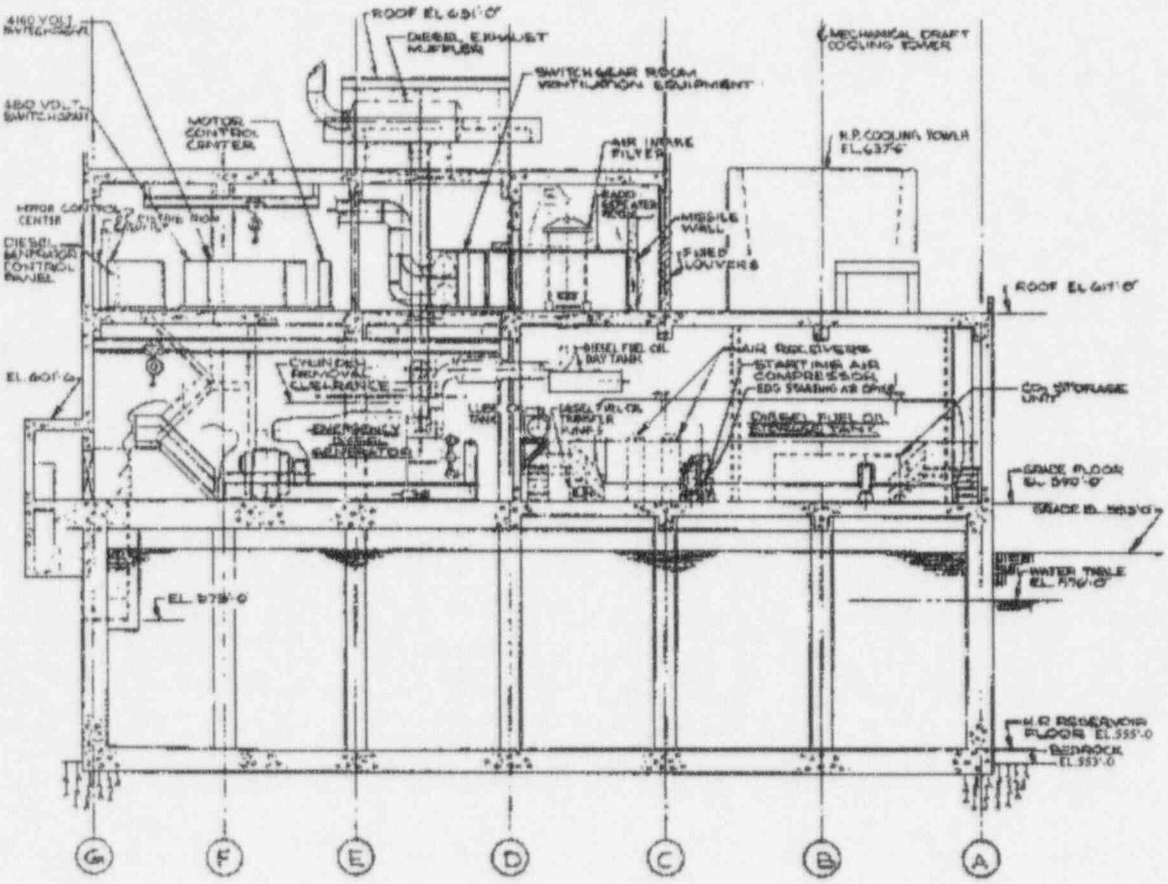


Figure 3-14 RHR Building Section Through EDG and Tank Rooms Looking South

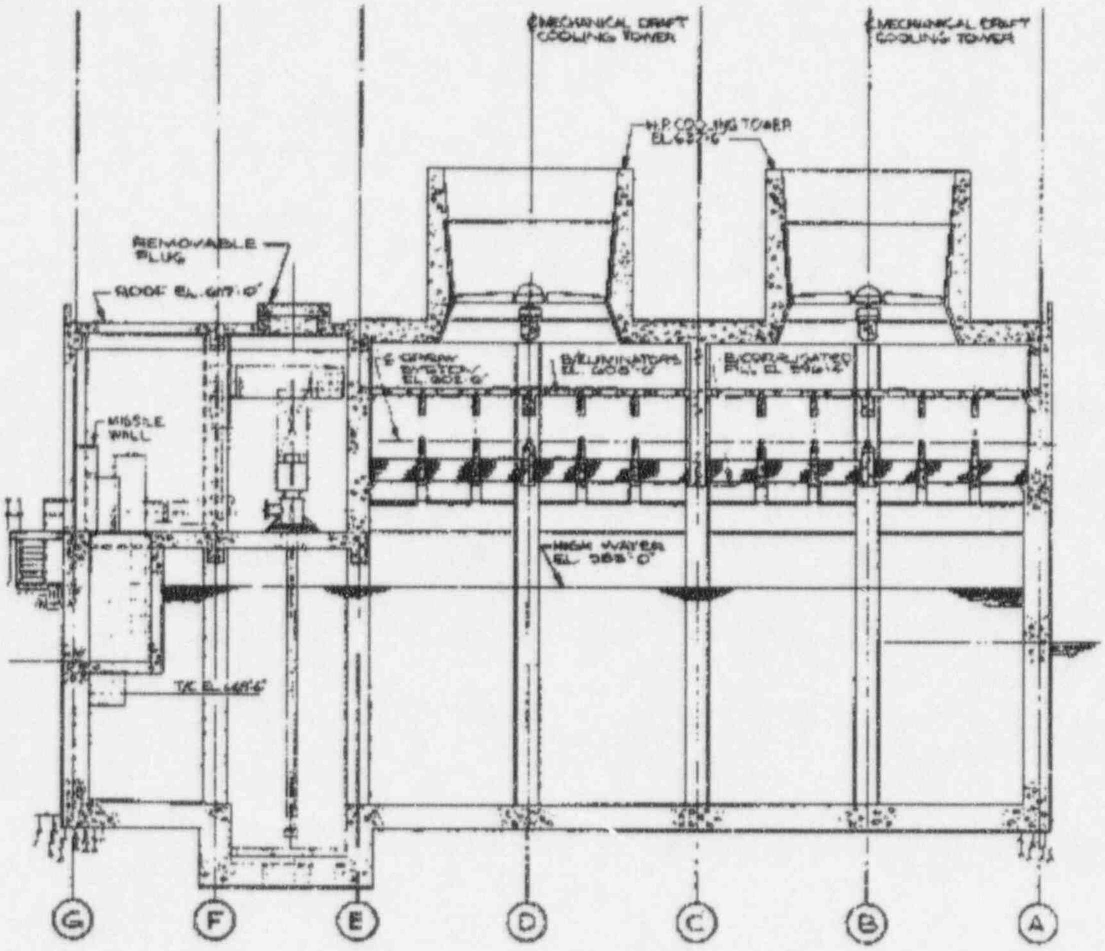


Figure 3-15 RHR Building Section Through Cooling Towers Looking South

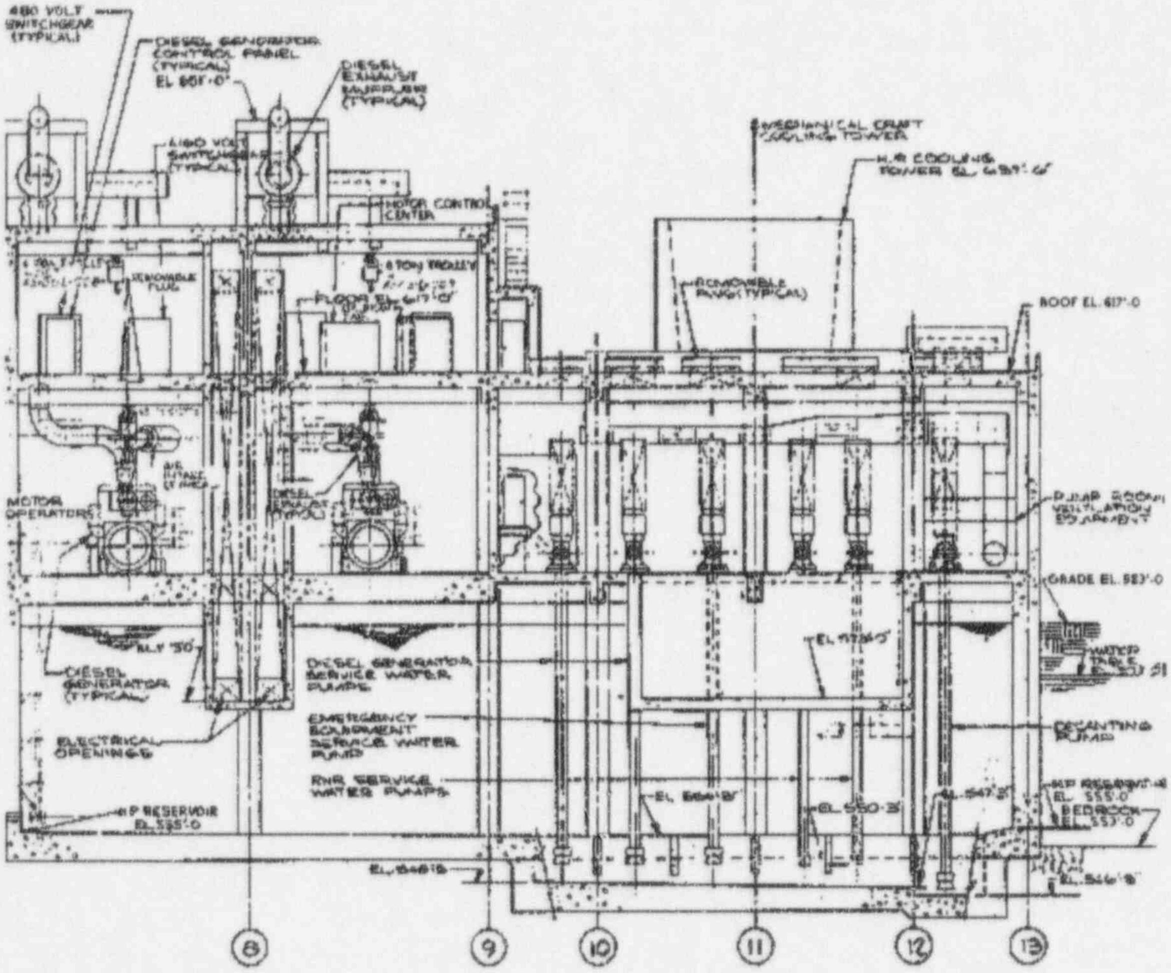


Figure 3-16 RHR Building Section Through EDG Rooms Looking West

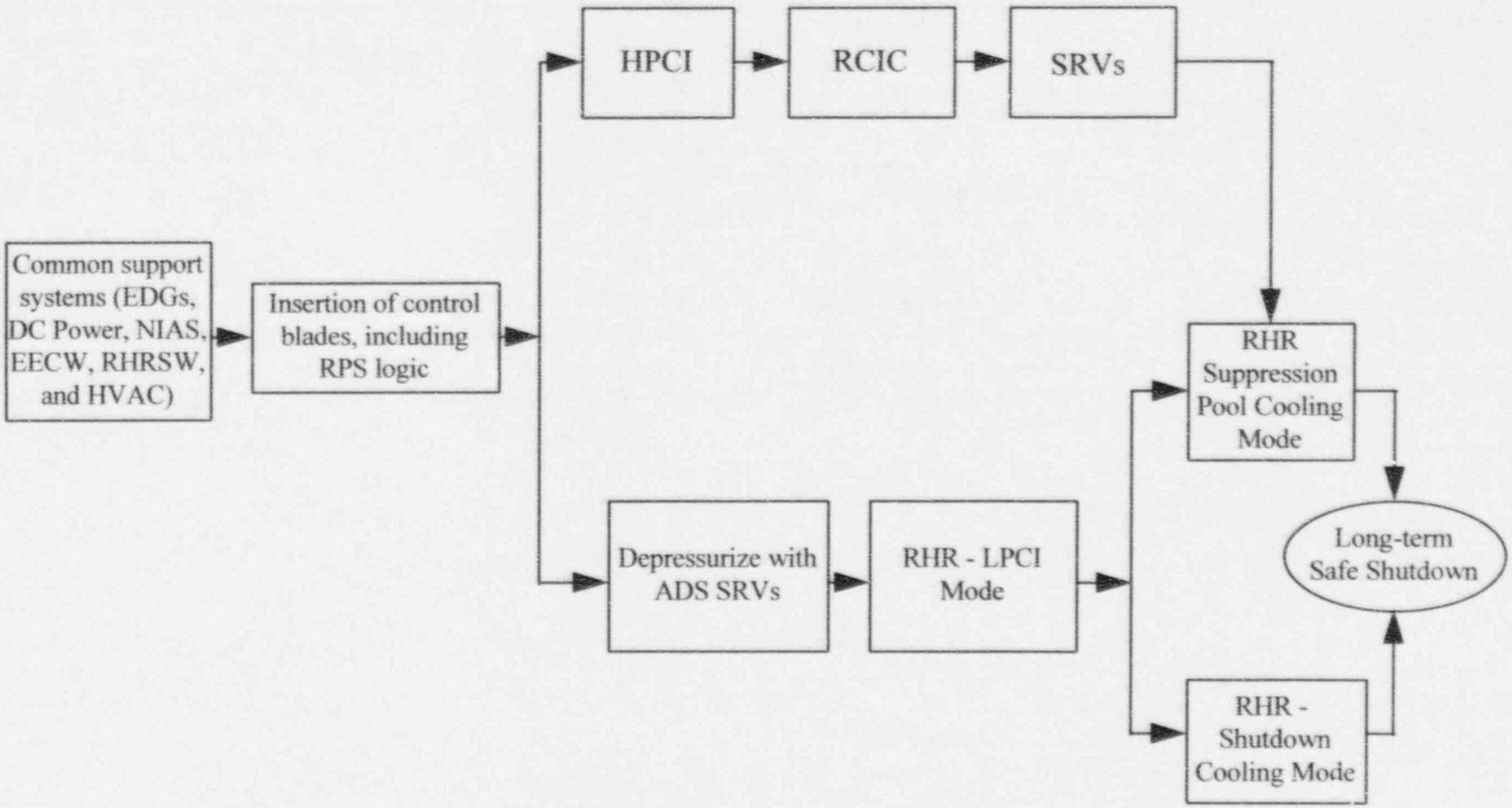


Figure 3-17 Finalized Success Path Logic Diagram

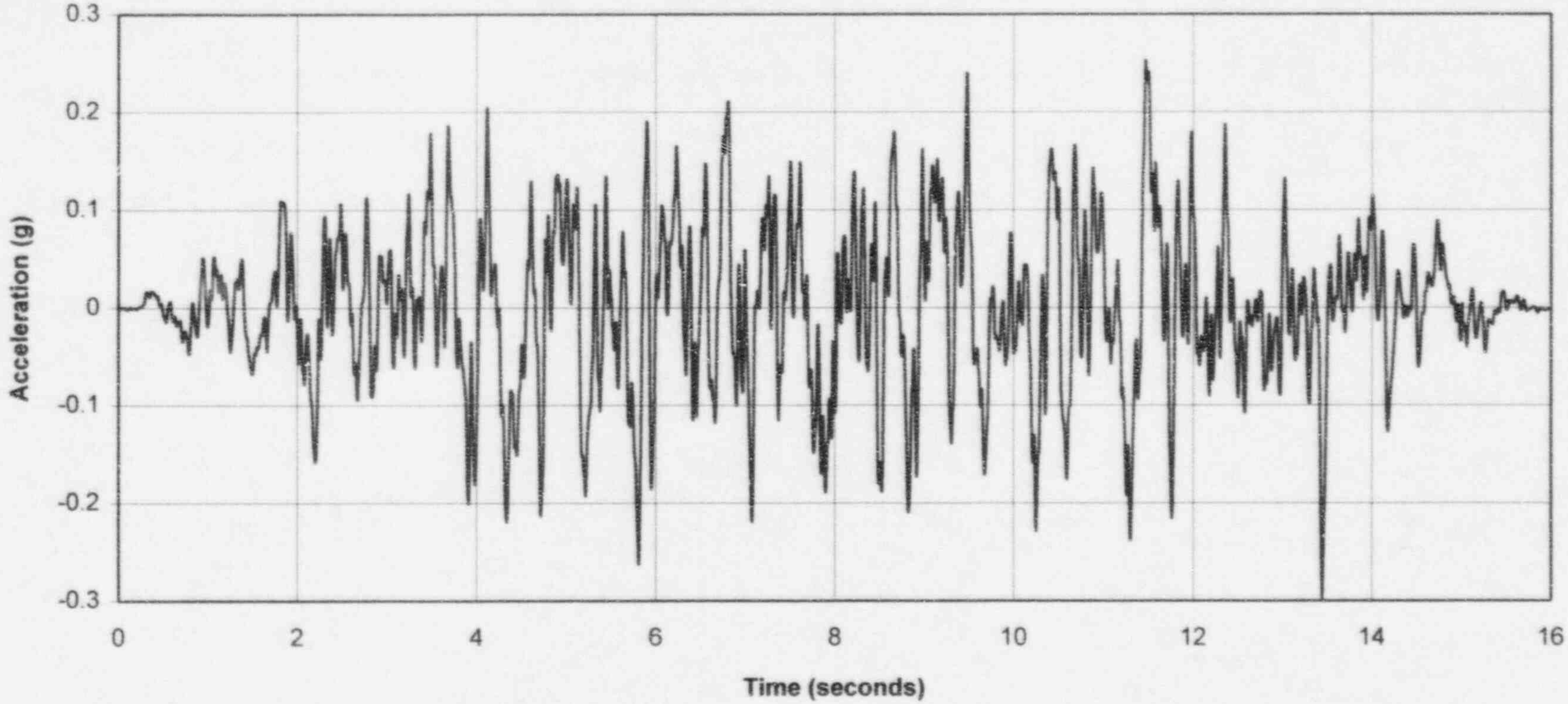
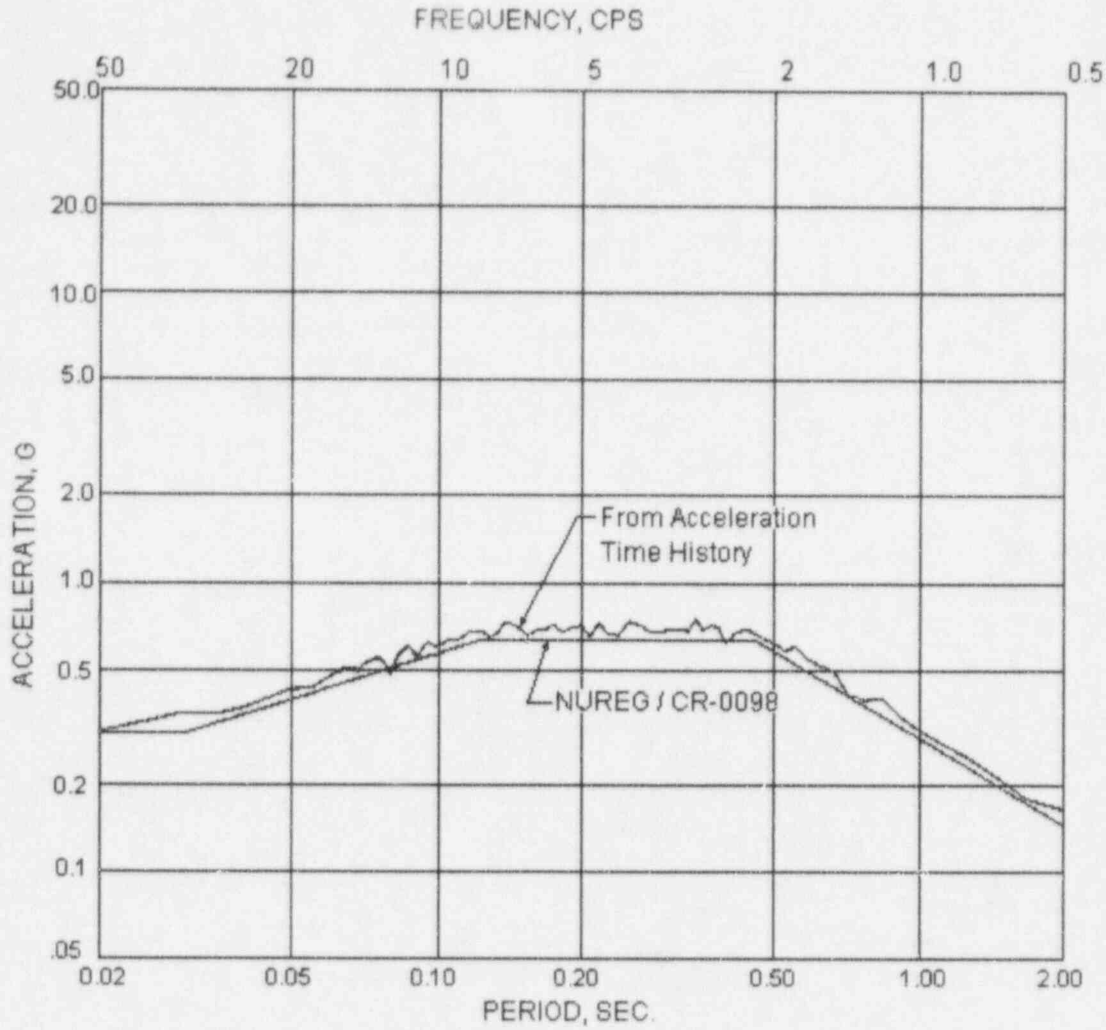


Figure 3-18 Time History Used To Develop In-Structure RLE Response Spectra



DAMPING = 0.05

Figure 3-19 Comparison of Time History Response Spectrum with NUREG/CR-0098 Response Spectrum

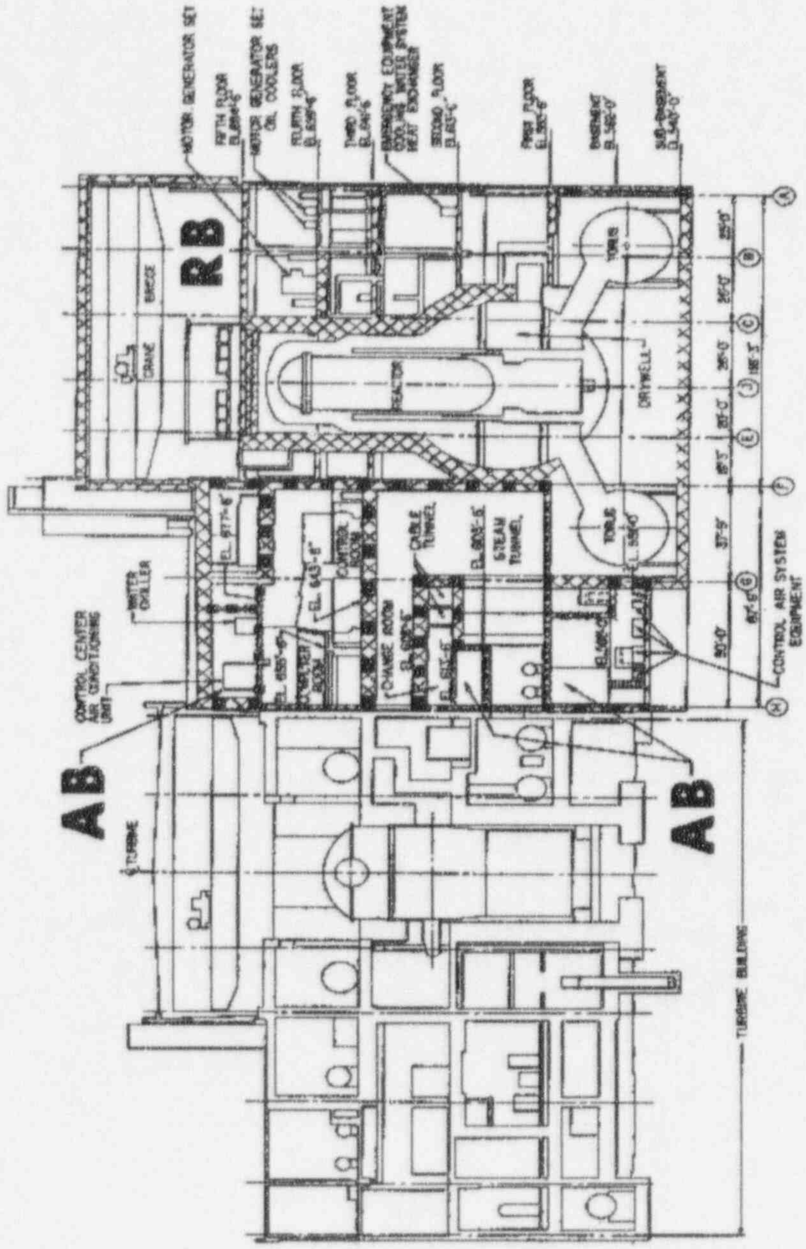


Figure 3-20 General Arrangement Transverse Section Through Turbine and Reactor/Auxiliary Buildings

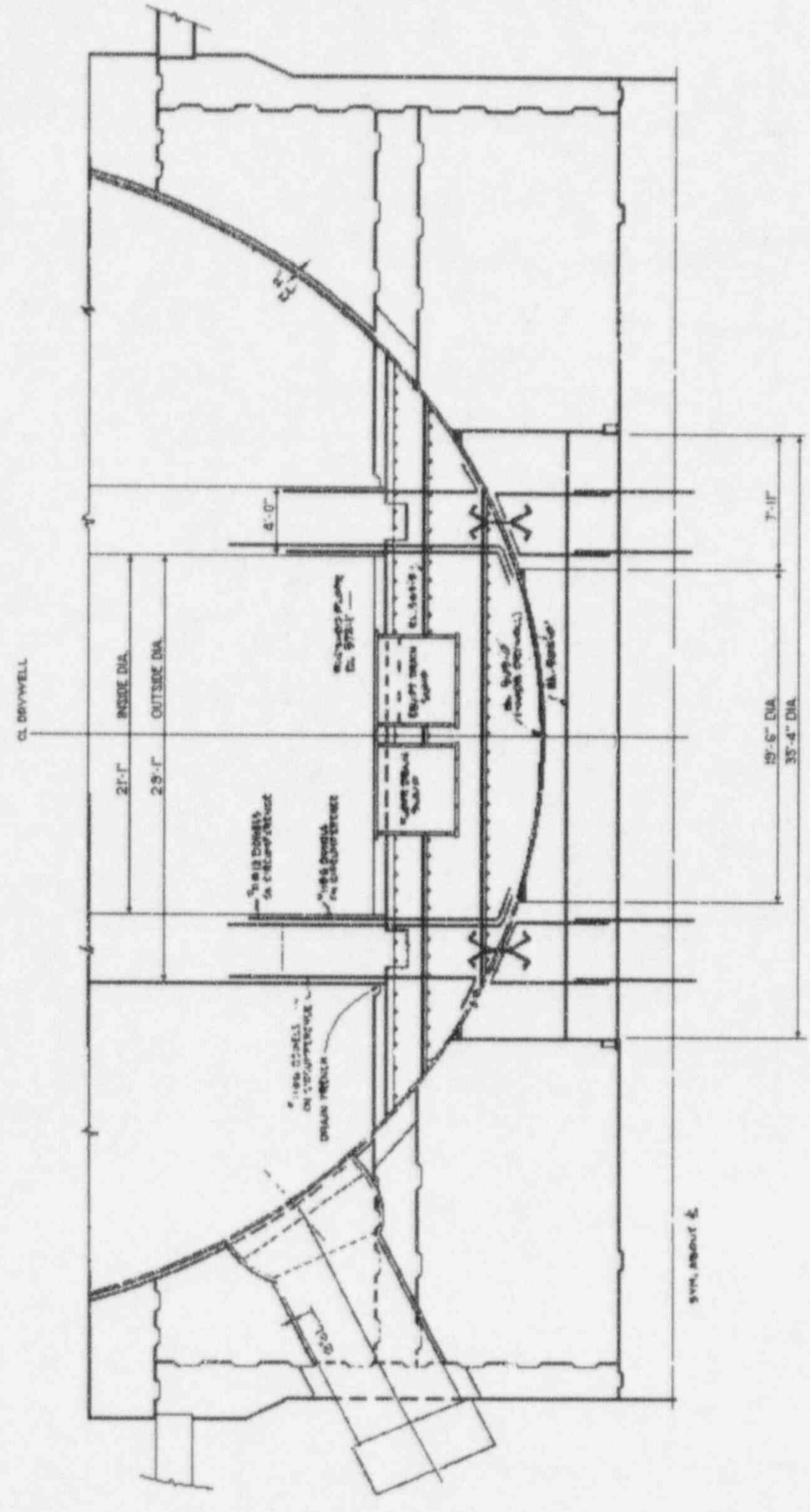


Figure 3-21 Attachment of the Drywell to the Drywell Pedestal

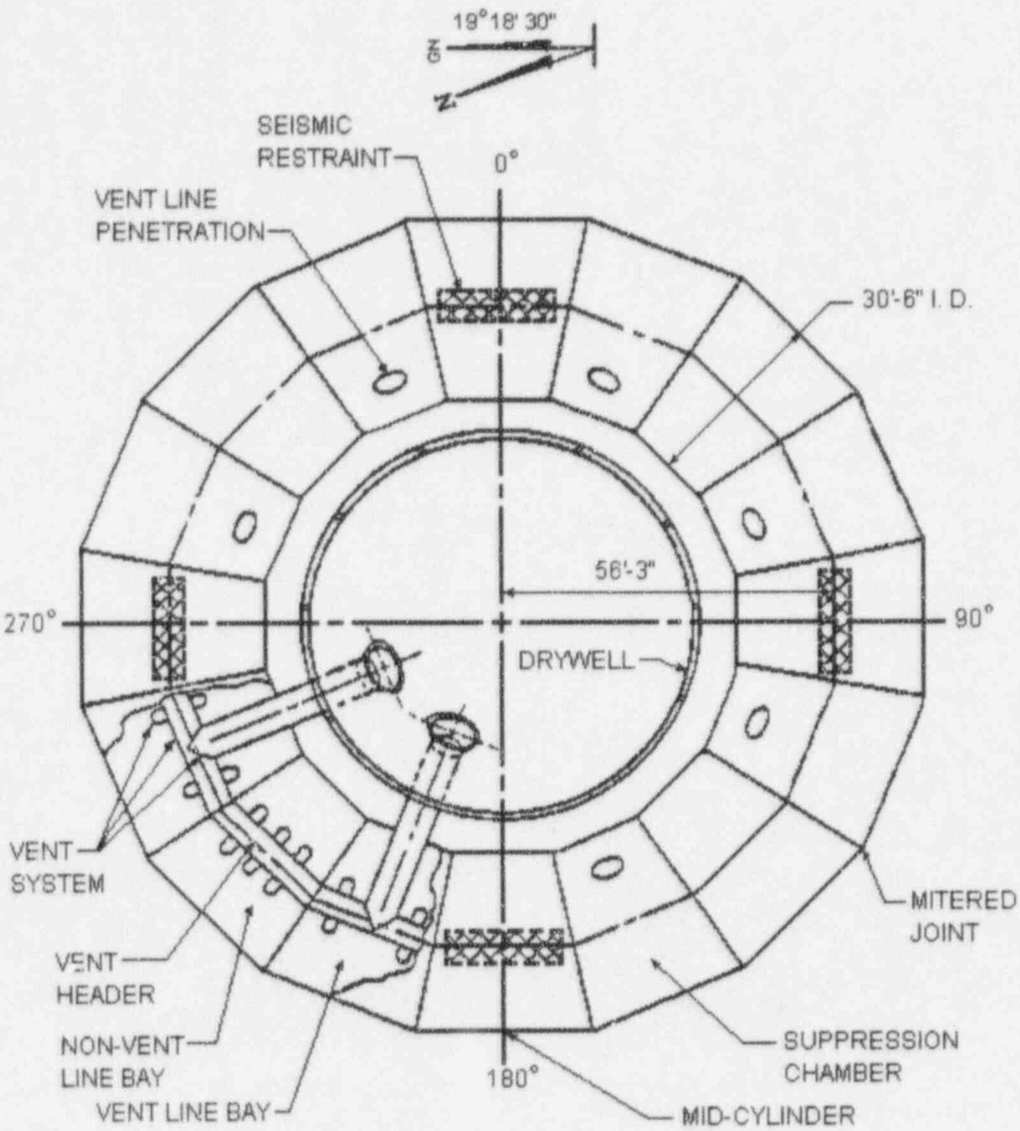


Figure 3-22 Suppression Chamber Plan View

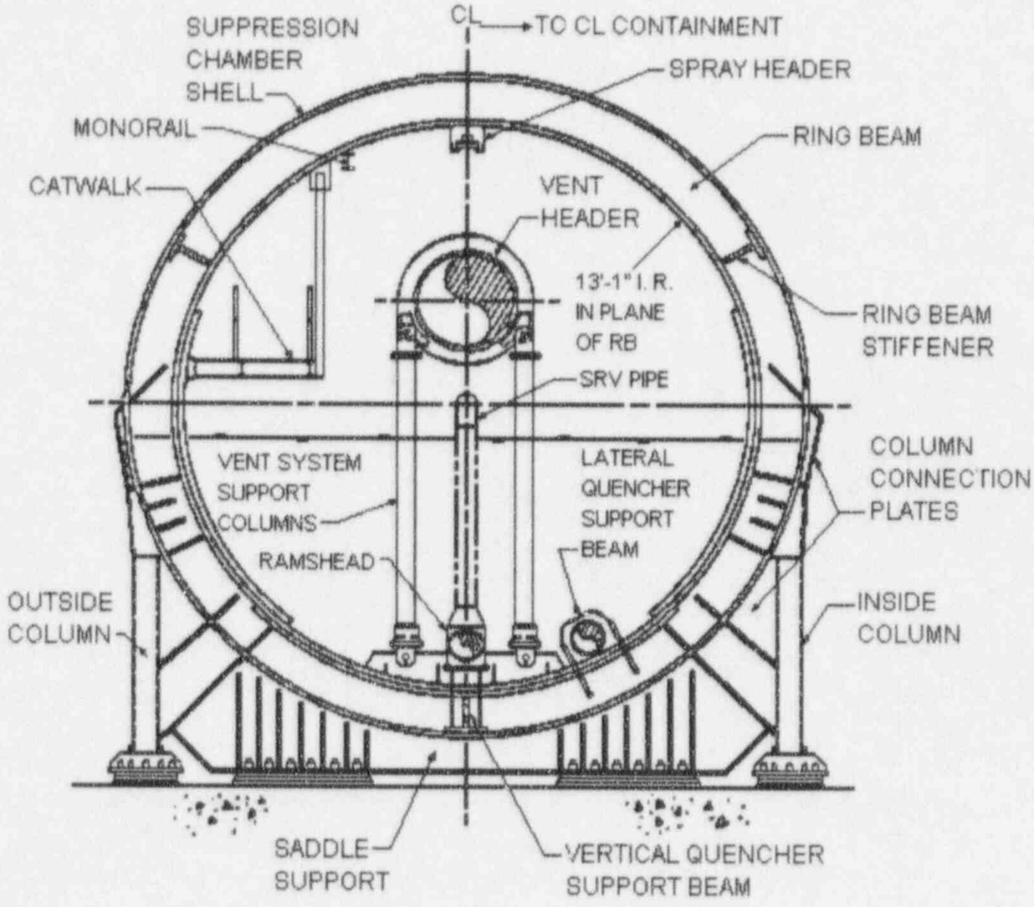


Figure 3-23 Suppression Chamber Support Details

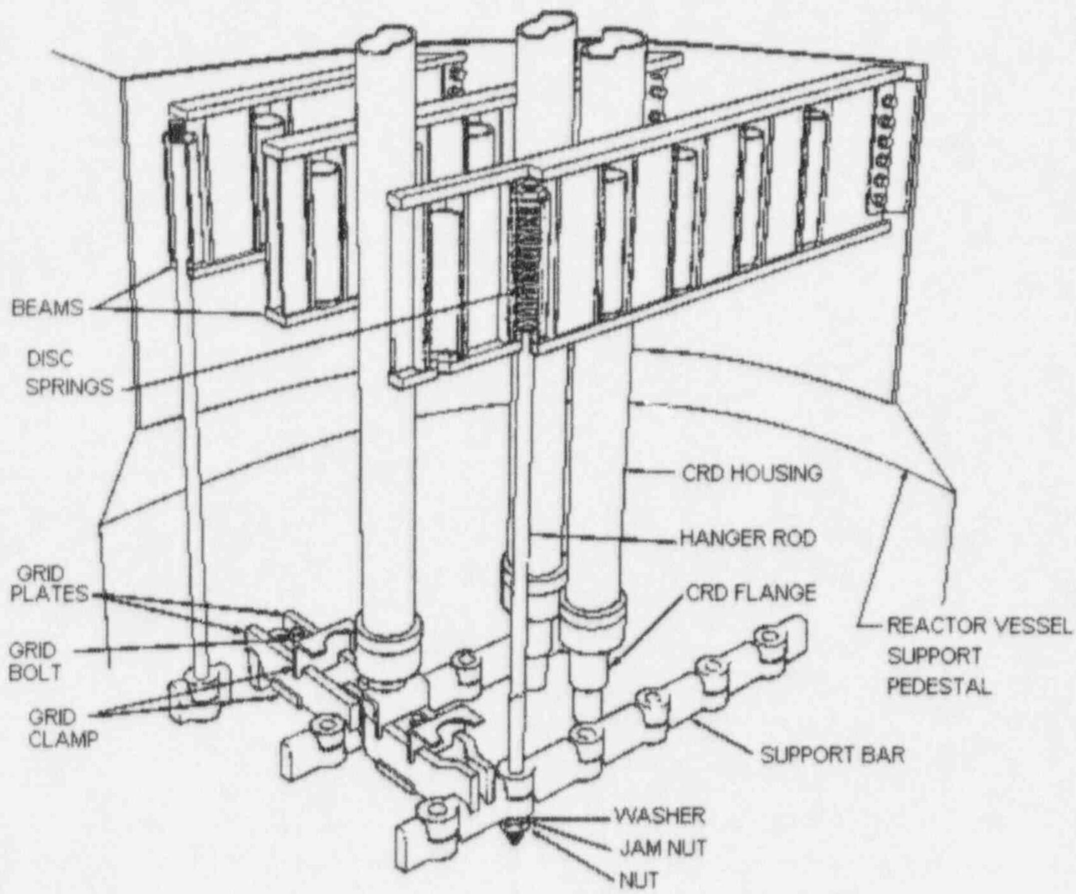


Figure 3-24 Control Rod Drive Housing Support

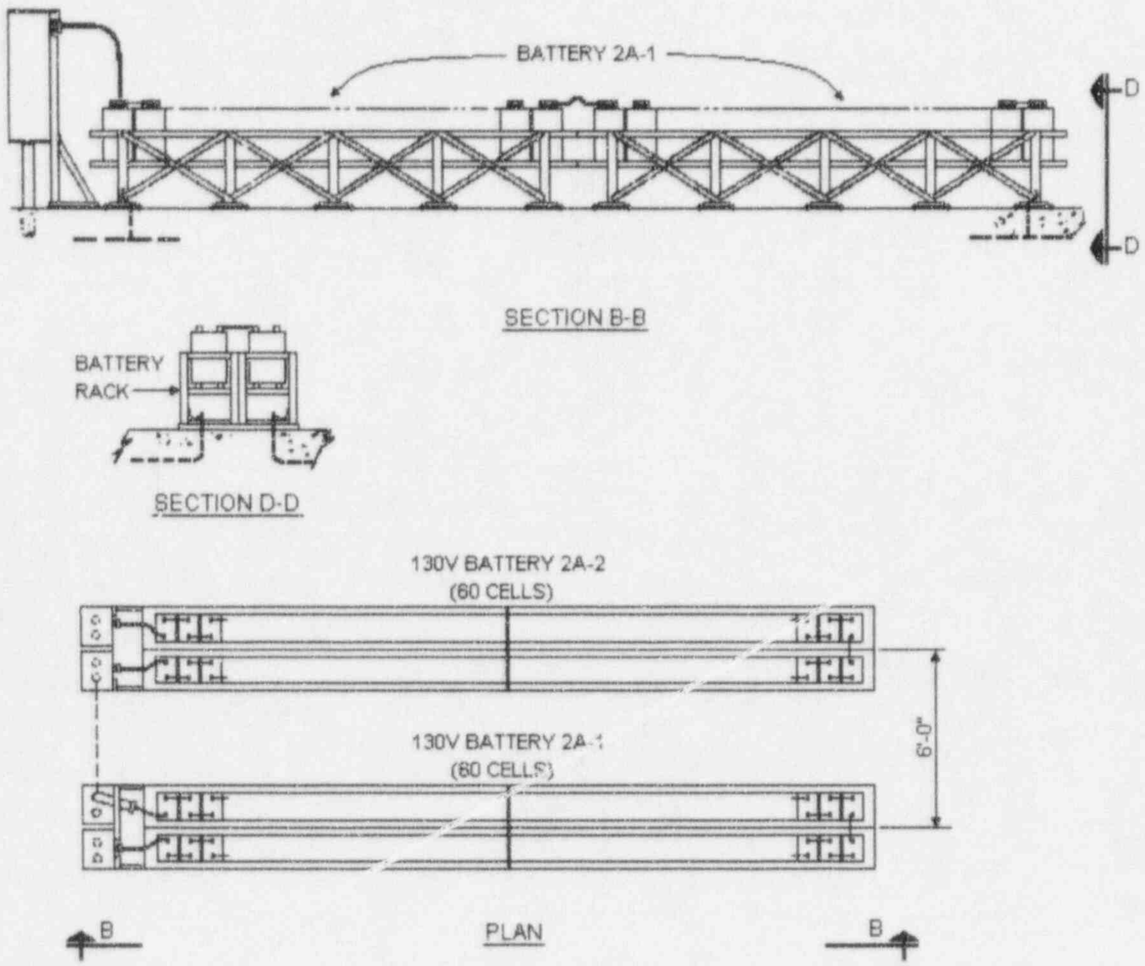


Figure 3-25 Battery Rack Arrangement

SECTION 4

INTERNAL FIRES ANALYSIS

4.0 GENERAL METHODOLOGY AND CONTRACTOR INTERFACE

4.0.1 Methodology Selection

The Fire-Induced Vulnerability Evaluation (FIVE) methodology **[4.2]** was selected as the method to satisfy the NRC request described in Generic Letter 88-20, Supplement 4. The NRC has reviewed the EPRI developed FIVE methodology and has determined that it provides a comprehensive approach for screening plant areas for fire risk and is an acceptable method for meeting GL 88-20 requirements.**[4.2, 4.25]** The FIVE methodology was used to identify fire areas of potential risk significance, calculate area fire ignition frequencies, and provide hazards analysis for resulting critical areas. This methodology provides a conservative analysis process that utilizes existing plant analyses, such as the Fermi 2 Fire Hazards Analysis (FHA) and the Fermi 2 Level 1 probabilistic safety assessment (PSA).

4.0.2 Description of FIVE Methodology

The FIVE methodology is a progressive screening technique based on conservative assumptions using industrial and plant specific data bases for evaluating fire event sequences. The overall objective is to determine the availability, in the event of fire, of plant equipment, cabling, and components necessary to achieve and maintain safe shutdown.

The methodology utilized in this analysis for evaluating the potential fire risk of the subject fire compartments is based upon the guidelines presented for Phase I, II and III of the Fire Induced Vulnerability Evaluation (FIVE) Methodology. Per the FIVE Methodology, a fire area containing no risk significant circuits or equipment, or with a calculated Core damage frequency (CDF) less than or equal to 1E-6 events per reactor year is considered insignificant to fire risk and may be screened from further analysis.

The FIVE methodology and this evaluation consist of three phases:

- Phase I: Fire Area Screen (Qualitative Analysis)
- Phase II: Critical Fire Compartment Screen (Quantitative Analysis)
- Phase III: Plant Walkdown/Verification and Documentation

The three phase process of the FIVE methodology is shown in Figure 4-1 and is described below.

4.0.2.1 FIVE Phase I (Qualitative Analysis)

The EPRI FIVE Phase I methodology is used to perform an initial screening of fire areas. This screening involves the identification of plant fire areas and a qualitative assessment of the consequences of a fire in these areas. The overall Phase I effort consists of the following tasks:

- Identify Appendix R safe shutdown systems
- Identify fire areas and associated compartments
- Identify Appendix R safe shutdown equipment by compartment
- Perform fire area vs. Appendix R safe shutdown system screen
- Perform fire area vs. Appendix R safe shutdown function evaluation
- Perform Fire Compartment Interaction Analysis (FCIA)

Each of the these tasks is described briefly below. More detailed discussions can be found in the EPRI FIVE document.**[4.2]**

4.0.2.1.1 Identify Appendix R Safe Shutdown Systems

This step involves identifying all the safe shutdown systems credited in the plant 10CFR50 Appendix R analysis. Note that Phase I considers only Appendix R safe shutdown equipment, as directed in the FIVE methodology.

4.0.2.1.2 Identify Fire Areas and Associated Compartments

The next step is to identify the distinct fire areas of the plant. This can be performed by review of plant general arrangement drawings and the Fermi 2 Fire Protection Analysis.**[4.3]** FIVE states that fire areas with numerous small rooms can either: (1) be treated together as one room; or, (2) represented by individual fire compartments. The use of individual fire compartments allows a more precise accounting for the location of equipment. The FIVE methodology recommends using a numbering scheme that relates to the Appendix R numbering scheme. Credit can only be taken for fire area barriers that are included in the plant inspection, testing, and maintenance program.

4.0.2.1.3 Identify Appendix R Safe Shutdown Equipment by Compartment

Once the fire compartments and safe shutdown systems are identified, the next step is to list safe shutdown equipment by compartment. This categorization lists the safe shutdown equipment expected to be impacted by a compartment fire and the safe shutdown equipment credited as available following a compartment fire.

4.0.2.1.4 Perform Fire Area vs. Appendix R Safe Shutdown System Screen

The next step is to review the safe shutdown and fire compartment information and to perform an initial screen of fire areas unimportant to risk associated with a fire in that area (e.g., Onsite Storage Bldg.). If a fire area and all its associated fire compartments do not contain safe shutdown equipment <u>and</u> a fire does not result in the demand for a plant shutdown, the entire fire area can be screened from further analysis.

4.0.2.1.5 Perform Fire Area vs. Appendix R Safe Shutdown Function Evaluation

For the remaining unscreened fire areas, the next step is to evaluate each fire compartment within the area assuming all safe shutdown equipment in each fire compartment is damaged and that the normal alternate shutdown path is unavailable. If, under this scenario, a fire in each compartment within the fire area damages safe shutdown equipment but does not cause a demand for safe shutdown functions in that fire area then the entire fire area can be screened from further analysis. If there is doubt whether the plant would shut down for a fire in a given area, FIVE directs to assume that plant shutdown would occur and not to screen out the area.

Note that in these first five steps of Phase I a fire is assumed to damage everything in the compartment, and either all fire compartments of a fire area screen out or the entire fire area must be retained for further evaluation.

4.0.2.1.6 Perform Fire Compartment Interaction Analysis (FCIA)

Up to this point all screened areas are entire fire areas and all their associated fire compartments. The remaining unscreened fire areas may contain numerous associated fire compartments, some of which may be unimportant with respect to plant fire risk. The purpose of this last step of FIVE Phase I is to identify these unimportant fire compartments and screen them from the detailed analysis to follow in Phase II.

Those fire areas not screened out in the prior steps are reviewed on a compartment basis to establish the adequacy of compartment boundaries. The information required to complete this review includes the fire rating of the compartment boundaries and the combustible loading in the compartment. The criteria used to determine whether a particular boundary is adequate with respect to the FIVE methodology is provided in the EPRI FIVE document and is as follows:

1. Boundaries between two compartments, neither of which contain safe shutdown components nor plant trip initiators, on the basis that fire involving both compartments would have no adverse affect on safe shutdown capability.
2. Boundaries that consist of a 2-hour or 3-hour rated fire barrier on the basis of barrier effectiveness.
3. Boundaries that consist of a 1-hour rated fire barrier with a combustible loading in the exposing compartment <80,000 Btu per sq. ft. on the basis of barrier effectiveness and combustible loading.
4. Boundaries where the exposing compartment has a very low combustible loading <20,000 Btu per sq. ft. and automatic fire detection on the basis that manual suppression will prevent fire spread to the adjacent compartment.
5. Boundaries where both the exposing and exposed compartments have a very low combustible loading <20,000 Btu per sq. ft. on the basis that a significant fire cannot develop in the area.
6. Boundaries where automatic fire suppression is installed over combustibles in the exposing compartment on the basis that this will prevent fire spread to the adjacent compartment.

A FIVE Fire Compartment Interaction Analysis (FCIA) form is completed for each of the unscreened fire zones. The FCIA form organizes the relevant data to facilitate the determination of the potential for fire spread beyond a single compartment.

If all boundaries of a compartment screen out and:

- the compartment contains no safe shutdown equipment, or
- following a fire, there is no demand for safe shutdown functions,

then the compartment can be screened from further analysis. Adjacent fire compartments with unscreened boundaries (i.e., a fire may spread from one compartment to the next) are combined into a single compartment.

The unscreened fire compartments at the end of Phase I are considered critical fire compartments that require further detailed analysis in Phase II.

4.0.2.2 FIVE Phase II (Quantitative Analysis)

Phase II of the FIVE methodology is a progressive probabilistic evaluation that considers the sequence of events which must occur to create the loss of safe shutdown. Phase II allows fire areas to be screened from further analysis once the frequency of fire initiated core damage accident sequences drops below 1E-6/yr. Phase II analysis consists of the following three tasks:

- Calculate compartment fire initiation frequency
- Calculate safe shutdown failure probability given the fire initiating event
- Calculate and/or evaluate fire propagation, damage, and suppression system effectiveness if required.

Each of the these tasks is described briefly below. More detailed discussions can be found in the EPRI FIVE document.**[4.2]**

4.0.2.2.1 Compartment Fire Ignition Frequency

This step estimates the fire ignition frequency for each Phase I unscreened fire compartment. The calculation of a compartment Fire Ignition Frequency requires data regarding the type and amount of equipment located in the compartment. The FIVE methodology uses a total of over 800 fire events spanning the period 1965-1988 to develop a generic data base used to estimate the Fire Ignition Frequency. This generic information is adapted to individual plants by counting and weighing the plant specific ignition sources and fuel sources.

If the Fire Ignition Frequency, F_i, is less than 1.0E-06/yr, then the fire compartment is screened from further analysis.

4.0.2.2.2 Safe Shutdown Failure Probability

If F_i is greater than 1.0E-06/yr, then further analysis is required. Redundant systems (i.e., either redundant Appendix R safe shutdown equipment or alternate safe shutdown equipment) which can perform the functions of the safe shutdown equipment in the area must then be identified.

All available shutdown paths, in addition to those considered in the Appendix R analyses, may be considered. The Fire Protection Analysis, the Fermi 2 equipment database (CECO) and the Fermi 2 cable database provide reference sources for identifying the specific equipment and cables in each fire compartment. The information from these databases is cross-referenced to Fermi 2 drawings in order to accurately determine equipment location. The Fermi 2 PSA model is modified to accont for fire damaged equipment and the model run to determine the conditional safe shutdown failure probability.

If the product of the fire ignition frequency and the conditional safe shutdown failure probability is less than 1.0E-06/yr, then the fire compartment is screened from further analysis.

4.0.2.2.3 Fire Damage Modeling and Suppression System Effectiveness

If the product of the Fire Ignition Frequency and the redundant system failure probability is greater than 1.0E-6/yr, then further analysis is required. Evaluation of each unscreened fire compartment continues to relax conservative assumptions in the analysis and further examine the events in the fire sequences. This evaluation includes consideration of combustibles in the area, vulnerability of safe shutdown equipment and alternate safe shutdown equipment to radiant heat exposure and hot gases, fire detection and suppression available in the area, and fire modeling to characterize the worst case fire scenario. Accepted fire modeling techniques are provided in the FIVE Methodology manual **[4.2]**. This step can lead to an iterative process requiring reruns of the PSA model described above with modifications to the extent of fire damaged equipment.

4.0.2.3 FIVE Phase III (Walkdown and Verification)

Phase III is a walkdown and verification process to determine whether or not the assumptions and calculations of the evaluation are supported by the physical conditions of the plant. This process can be performed during or after the Phase I and Phase II analyses. In practice, such as in this evaluation, it often becomes necessary to also perform verification walkdowns during the first two phases.

4.0.3 Key Assumptions

4.0.3.1 General Discussion of Assumptions

For the purpose of this analysis, the phrases 'Safe Shutdown equipment' and 'Safe Shutdown circuits' refer to that set of equipment identified as being required to achieve and maintain safe shutdown. This equipment is the same as that credited in the Fire Hazards Analysis **[4.3]**. 'Safe Shutdown equipment' and 'Safe Shutdown circuits' are identified in the Fermi 2 database 'Combine1.db'.

For the purpose of this analysis, the phrases 'Balance of Plant equipment' and 'Balance of Plant circuits' refer to that set of equipment identified during the performance of the FIVE analysis as being available to assist in the removal of decay heat or to provide makeup water to the reactor. This equipment is not required for the Appendix R safe shutdown of the plant and is not credited in the Fire Hazards Analysis **[4.3]**. 'Balance of Plant equipment' and 'Balance of Plant circuits' are identified in the Fermi 2 database 'Combop.db'.

Routing information for the Reactor Protection System (RPS) is not required for the normal Appendix R reviews and was not available for this analysis. Therefore in most cases this analysis assumed that RPS existed in the compartment of interest and an automatic trip occurred resulting in the loss of the condenser.

The cabling at Fermi 2 has been evaluated by Detroit Edison and has been found to be equivalent to IEEE 383 cable in regard to self ignition and propagation of flame **[4.3]**. Cable runs, junction boxes and cable splices are not considered as ignition sources in the subject fire areas.

4.0.4 Software for FIVE Evaluation

Detroit Edison (DECo) was not a member of EPRI when the FIVE analysis was being performed. In order to maintain consistency between the two contractors chosen to assist in the preparation of the report, DECo had ERIN Engineering develop a software program, to be used by both contractors, to perform the automated portions of the analysis. This software is identified throughout this report as QUICK FIVE **[4.7]**. QUICK FIVE is a software program that emulates the analyses performed in the automated FIVE program that EPRI provides.

QUICK FIVE is organized into two separate functional system program elements.

> The Phase I program element allows data to be imported from the plant specific databases and automatically prepares the Safe Shutdown System versus Fire Area/Compartment Matrices based on the imported data and prepares the Fire Compartment Interaction Analysis (FCIA) data sheets.
>
> The Phase II program element consists of two Excel workbooks and a macro. The INPUT.XLW workbook is completed by the user and provides generic plant data to be used by the individual compartment workbooks. The PHASE_II.XLW workbook is a template for the compartment workbook. The macro is provided to assist the user in making copies of the template for each of the fire compartments and tracking the status of the Phase II analysis.

The combination of these two QUICK FIVE program elements allows the user to quickly and efficiently implement the FIVE methodology.

Detroit Edison also provided both contractors with copies of databases developed internally which contained:

- a listing of the equipment credited in the Appendix R analysis (COMBINE1.DB)
- a listing of the non Appendix R equipment that had been routed and therefore could be credited (COMBOP.DB)
- a listing of ignition sources, located by fire area/compartment (COMBINE2.DB)

These databases were developed and maintained by DECo and provided to the contractors to maintain consistency in the equipment being used in their individual analyses.

4.0.5 Contractor Interface

Detroit Edison (DECo) used in-house resources to identify the equipment to be credited in the analysis, provide routing information, perform the PSA analysis, perform the verification walkdown and perform the peer review of the overall FIVE report.

VECTRA Technologies, Inc., performed the FIVE analysis for the Reactor Building, the Turbine and Radwaste Buildings and the balance of the non-safety related buildings and yard structures. This analysis is documented in Section 4A.

ERIN Engineering and Research, Inc., performed the FIVE analysis for the Auxiliary Building. This analysis is documented in Section 4B.

ERIN and VECTRA were used due to their experience in similar analyses at other utilities and because DECo staff was involved in extensive efforts to restart the plant following the 1994 turbine failure.

DECo staff interfaced with ERIN and VECTRA throughout their analysis. DECo reviewed and accepted the assumptions and provided coordination between the two contractors in order to ensure that the final report from both contractors was as similar as possible. Several discussions were held among the three parties in order to ensure that all assumptions and fire modeling parameters were similar and reasonable. DECo also visited the contractors offices during the preparation of the report in order to review the in-progress work.

Both contractors toured Fermi and performed walkdowns to confirm assumptions. The number of the walkdowns performed by the contractors varied because of the buildings analyzed and type of information required. DECo staff escorted the contractors and participated in all walkdowns providing utility expertise and knowledge of the plant structures and systems. DECo engineers performed a final verification walkdown after receipt of the analyses in order to satisfy themselves that the analyses correctly addressed the hazards for each fire compartment.

4.1 REVIEW OF PLANT INFORMATION AND WALKDOWNS

A variety of information sources were reviewed and a number of walkdowns were performed in support of the internal fire analysis. These are discussed below.

4.1.1 Review of Plant Information

The following are the primary information sources reviewed as part of the internal fire analysis:

- Fire Protection Analysis
- Fermi 2 CECO database
- Fermi 2 cable database
- Plant arrangement drawings
- FHA drawings
- Electrical drawings
- Fermi 2 PSA REBECA **[4.68]** documentation
- Plant procedures
- Design Calculations

The Fire Protection Analysis is a detailed information source that provided numerous inputs into the fire analysis, such as:

- Equipment in each fire zone
- Combustible loading in each fire zone (type and quantity)
- Fire protection equipment in each fire zone
- Consequences of design basis fire in each fire zone
- Consequences of fire suppression system actuation in each fire zone
- Design basis information concerning fire suppression systems
- Design information concerning fire barriers
- Design information concerning alternate shutdown capability

The accuracy of the cable routing information is of paramount importance since it is this that provides much of the basis for the fire analysis. For the Appendix R equipment, the associated raceways and their location within specific fire areas were validated as part of the Appendix R program. Summaries of this information are contained in the Fire Protection Analysis in the form of drawings and text. More detailed information, such as conduit, cable tray, and cable numbers were obtained from the DECo cable database.

Plant arrangement drawings were used, as necessary, in conjunction with CECO database and DECo cable database information to locate equipment credited in the fire analysis. The arrangement drawings were also reviewed throughout the internal fire assessment, as necessary, to locate equipment and to assess the potential for fire spread.

The FHA drawings were also a well used source that aided in the determination of the potential for fire spread. The FHA drawings indicated the following information directly on the drawings:

- Fire zone boundaries
- Fire doors
- Fire dampers
- Raceway fire wrap
- Fire detection equipment
- Suppression systems and coverage areas

Conduit and cable tray drawings were used when it was necessary to accurately locate specific cabling (either for fire damage modeling or due to some ambiguity in another information source). one-line electrical drawings were used as a reference source in the determination of power supply vulnerabilities.

The Fermi 2 PSA REBECA documentation was used as an information source in the determination of equipment vulnerabilities for the auxiliary building evaluation. The REBECA fault trees provided a quick reference source for determining the impact on systems modeled in the fire analysis due to fire induced damage of individual components or subsystems. This was necessary to support the modeling of system train failures in the RISKMAN PSA fire model.

Plant procedures were reviewed as part of the assessment to: 1) verify that all fire barriers credited in the fire analysis are covered under a surveillance program, 2) verify that ignition sources and combustibles are covered by procedural requirements, and 3) determine whether a fire initiated event occurs for a fire in a given area. The procedures reviewed included the following:

- Abnormal Operating Procedures (AOPs)
- Administrative Procedures
- Fire Protection Procedures
- Fire Pre-Plan Procedures

A number of Fermi 2 design calculations were used to support the determination of equipment vulnerabilities:

- DC-5024, "Load List and Loss of Power Impact - MPU #1"
- DC-5025, "Loss of Power Impact on MPU #2"
- DC-5026, "Loss of Power Impact for MPU #3"
- DC-5027, "Loss of Power Impact for MPU #4"

- DC-5028, "Loss of Power Impact for MPU #5"
- DC-5029, "Loss of Power Impact for MPU #6"
- DC-5702, "Fire Loading Calculation"

In addition, the Fermi 2 "Cable Routing Equipment - From and To By Equipment" database was used to support the determination of equipment vulnerabilities.

4.1.2 Walkdowns

Walkdowns were performed by ERIN and VECTRA during their analyses in order to verify:

- That the quantities and types of transient combustibles used in the analysis were suitable for the area being analyzed.
- That the equipment location and potential for fire damage used in the fire modeling was suitable for the area being analyzed.

The fire task manager who was familiar with the FIVE methodology and the material needing to be determined during these walkdowns escorted the contractor personnel and was involved in the collection of the material required for the analyses.

In addition to the walkdowns discussed above, DECo performed a final series of confirmatory walkdowns, following the receipt of the ERIN and VECTRA reports. This final series of walkdowns was used to verify that the assumptions used in the analysis were acceptable and to check on whether there were any significant ignition sources or equipment missing from a screened or unscreened compartment.

During this final walkdown several minor discrepancies were identified such as equipment being correctly located in a fire area but in an adjoining fire compartment. This type of discrepancy, because it identified only a small number of ignition sources, would have no significant effect on the results of the analysis.

In addition, since no fire areas screened in the Phase I portion of FIVE and fire modeling was used to screen the compartments in Phase II, the final results of the analysis would not change. Therefore the Phase I and initial portions of the Phase II analyses were not redone to reflect this type of change.

4.2 FIRE HAZARD ANALYSIS

4.2.1 Fire Hazards Methodology

4.2.1.1 Fire Zones

The FIVE Phase I analysis relies heavily on the existing Appendix R documentation, most notably, the Fermi 2 Fire Protection Analysis and the Fermi CECO and cable databases. The Fermi 2 fire zones are the starting point for identifying FIVE fire compartments. A list of safe shutdown systems that could be disabled by a fire was obtained from the Fire Protection Analysis for each of the fire zones. Fire zones that do not contain safe shutdown equipment were examined for the potential for a Fire Initiated Event (FIE).

4.2.1.2 Safe Shutdown Equipment

The identification of which Safe Shutdown equipment would be impacted by a fire event in the fire areas was determined by Detroit Edison personnel and provided to ERIN and VECTRA in the database 'Combine1.db'. This identification included the safe shutdown cables identified in the Appendix R calculation **[4.6]**. This allowed the analyst to determine what equipment outside the fire area or compartment would be affected by fire in the area of interest. This location and routing of this equipment has been evaluated in the Appendix R analysis **[4.6]** and it meets the separation and protection requirements of Appendix R.

The FIVE analysis did not arbitrarily assume loss of offsite power (LOSP). The routing of the offsite power cables and the evaluation of the potential for LOSP was evaluated for each fire area. LOSP was assumed only when the offsite power feeds or equipment was affected by the fire in the area of concern.

4.2.1.3 Balance of Plant Equipment

In addition to Safe Shutdown equipment, the identification of Balance of Plant equipment impacted by a fire event in the fire areas was determined by DECo personnel and provided to ERIN and VECTRA in the database 'Combop.db'.

The BOP equipment credited in the analysis included only those cables and equipment that were specifically located or routed. The systems included in the BOP routing are identified in Table 4- 3. The choice of equipment was determined by reviewing the existing IPE analysis to determine the most significant systems, with respect to core damage, and then routing the cables and locating the equipment, by fire zone. By locating these components, the FIVE analysis was able to credit the ability to sustain decay heat removal and injection into the reactor to limit the potential for core damage.

4.2.1.4 Plant Wide Ignition Sources

The number and location of plant wide ignition sources were determined by DECo. This information was provided to ERIN and VECTRA in database 'Combine2.db'. This database was developed by reviewing plant drawings to determine the equipment locations and by using the Central Component Database (CECO) to identify the potential ignition sources.

4.2.1.5 Fixed Ignition Sources:

- Motors:

  The assumption was made that motors less than 25 horse power were not significant ignition sources. A fire in small motors will consist of the insulation and small quantities of grease inside the motor and the damage would be limited to the motor. The heat developed by a fire in a small motor was considered insufficient to damage components outside the motor itself.

- Pumps:

  The assumption was made that pumps less than 25 horse power were not significant ignition sources. A fire in small pumps will consist of the small quantities of grease or oil inside the pump and the damage would be limited to the pump. The heat developed by a fire in a small pump was considered insufficient to damage components outside the pump itself.

- Motor Control Centers
- Electrical Cabinets
- HVAC system fans and electric heaters
- Batteries and Battery Chargers
- Elevator Motors

Non-Qualified cable runs and splices in junction boxes are not considered probable ignition sources. All cables at Fermi are equivalent to IEEE 383 **[4.3]** and splices in junction boxes or pull boxes are controlled by site specifications **[4.8]** and do not degrade the IEEE 383 equivalency. In addition, it is also assumed that a fire in any junction box would be limited to the box and would not generate sufficient heat to damage equipment or cables outside the box.

4.2.1.6 Transient Ignition Sources:

The types and quantities of transient combustibles used in fire modeling varies between the Auxiliary and Reactor Building. This is based on a review of the plant combustible control procedure **[4.19]** and the types of transients reasonably expected to be present or used in the area of concern. Typical transient combustibles are listed below:

- Waste containers
- Welding cables
- Extension cords
- Heaters
- Overheating

- Hot pipes

4.2.1.7 PSA Modeling

The existing IPE PSA model was adjusted to account for fire damaged equipment which also includes any equipment (Safe Shutdown or Balance of Plant) for which routing information was not available. Section 4.6 addresses the methodology used to determine the Conditional Core Damage Frequency (or Conditional Core Damage Probability).

4.2.1.8 Screening

The Phase I Fire Compartment Interaction Analysis (FCIA) was performed using the QUICK FIVE program **[4.7]**. The purpose of the Fire Compartment Interaction Analysis (FCIA) is to evaluate the potential for fire spread across fire compartment boundaries within a fire area. The Fire Compartment Interaction Analysis (FCIA) also evaluated the potential for fire spread across barriers separating fire areas.

Phase I analyzed a total of 48 fire areas. These areas consisted of the fire zones identified in the Fire Hazard Analysis **[4.3]** and are listed in Table 4-1. No fire areas were screened in the FCIA due to lack of routing information for the Reactor Protection System. As a result, all fire areas remain unscreened in Phase I and were evaluated in the Phase II analysis.

The Phase II Fire Compartment Screening was performed using QUICK FIVE. The purpose of Phase II is to identify potential fire vulnerabilities to equipment, components and cables necessary to assure the capability for safe and stable plant shutdown conditions.

Phase II analyzed 54 fire compartments. These compartments are identified in Table 4- 2. In Section 4A VECTRA discusses a Phase II analysis and a Phase III analysis. The Phase III analysis is in reality a continuation of the Phase II analysis with additional modeling and definition of fire compartments. ERIN limited all discussion to a phase II analysis.

Both ERIN and VECTRA performed the Phase II analysis in two steps. The first step consisted of evaluating the fire compartments as a whole to determine whether they would screen out. The compartments that screened out in the initial cut are listed in Table 4- 6 and Table 4- 12. Sixteen of the fifty-four compartments screened out in the initial analysis and thirty-two of the remaining thirty-eight screened out in the second step of the analysis.

The six remaining unscreened compartments are listed in Table 4- 14. These unscreened compartments consist of the Control Room, Division 1 and 2 Switchgear Rooms, the Division 1 portion of the Miscellaneous Room on Elevation 643' 6", the Relay Room, and the Second Floor of the Reactor Building.

4.2.2 Fire Growth And Propagation

The treatment used in this analysis for fire growth and propagation follows the FIVE methodology. As part of the fire growth and propagation analysis, calculation spreadsheets were developed for different fire types using the QUICK FIVE computer code.

Consistent with the requests of NUREG-1407, the following issues are discussed below:

- Fire size and duration
- Cross-zone fire spread
- Spread of hot gases and smoke

4.2.2.1 Fire Size and Duration

The recommendations in the FIVE methodology are used to estimate the fire size and duration.

The fire size and duration depends on the type and amount of combustibles available. The FIVE methodology classifies combustibles into two types:

- Fixed combustibles
- Transient combustibles

Examples of fixed combustibles are cables, pump/motor lubricating oil, electrical cabinets, batteries and filtration media (e.g., carbon). Examples of transient combustibles are rags, anti-contamination clothing, cleaning solvents, and trash barrels. Per guidance in the FIVE methodology, transient combustibles that need not be considered include:

- Flammable and combustible liquids stored in approved containers
- Flammable and combustible liquids stored in approved storage cabinets
- Combustible liquids stored in sealed 55 gallon drums
- Clothing and other incidental combustibles kept in closed metal cabinets
- Clothing and trash kept in closed non-combustible containers

The duration and size of the fire is determined by the amount of combustibles in the area (determined from the Fermi 2 Fire Protection Analysis and walkdowns) and the heat release rate of the combustible type as defined in the FIVE methodology. Guidance provided in the FIVE methodology, in the form of examples and test data, was used to assign heat release rates and damage thresholds for equipment for which heat release rate and damage threshold data could not be located (e.g., switchgear fires).

4.2.2.2 Cross Zone Fire Spread

The assessment of the potential of fire spread from one compartment to adjacent compartments is performed in the Fire Compartment Interaction Analysis of Phase I. The FIVE methodology provides boundary criteria to be used in the determination of the potential for fire spread. If the boundary criteria are met, then the analysis assumes that any postulated fire in the area will remain within the confines of the fire compartment boundaries. If the boundary criteria are not met for an adjacent compartment then the analysis assumes that fire may spread to that adjacent compartment. As such, adjacent fire compartments with unscreened fire barriers are combined into single fire compartments at the end of Phase I.

4.2.2.3 Spread of Hot Gases

The FIVE methodology conservatively assumes at the initial stage that any fire in a compartment will result in damage to all equipment within the compartment. For compartments that do not screen out with this assumption, the methodology provides a detailed process of fire damage analysis. This process evaluates targets in relation to the fire plume, hot gas layer, and thermal radiation. The modeling of fire growth and propagation uses the algorithms and look-up tables contained in the FIVE methodology.

4.2.3 Evaluation Of Component Fragilities And Failure Modes

4.2.3.1 Evaluation of Fire-Induced Failures

Information from the Fermi 2 CECO and cable databases and drawings (conduit, cable tray, isometric, and arrangement) was used to determine the component fragilities (vulnerabilities) in a compartment. Database searches by compartment provided location information for the following components:

- cable trays
- conduit
- cables
- equipment

For many compartments, assuming that all cable and equipment in a compartment resulted in failure of the associated systems would result in a core damage frequency many orders of magnitude above the FIVE screening criterion of 1E-6/yr. As such, the above information was reviewed in detail to determine realistic functional and systemic vulnerabilities in the compartment. This investigation typically involved noting whether a cable was a control, power, or instrumentation cable, and which equipment the cable connected. Using systems interaction knowledge, a determination was then made as to whether failure of a specific cable resulted in failure of a modeled function or system. Detailed electrical circuit analysis was not performed. Although this investigation was conservative (e.g., when a circuit function was unclear, it was assumed to cause system failure), this investigation did result in deleting a number of obvious non-system failures, that would have otherwise been considered failures, from the initial screening quantification runs.

This approach was used primarily in the initial screening quantifications of the Phase II analysis. Once detailed fire modeling began, the analysis focused on damage thresholds, critical fire distances, realistic fire scenarios, and fire suppression. When equipment was determined to be damaged due to fire, the information from the equipment vulnerability investigation was used to make new quantification runs or use existing runs.

4.2.3.2 Fire Damage Modeling Approach

The deterministic fire modeling performed for the compartments that did not screen in the initial screening quantifications began with a realistic review of the ignition sources in the compartment. This review eliminated ignition sources that were determined not to exist or not to apply to the compartment (e.g., heater in the cable spreading room). In addition, ignition sources such as emergency lights and fire protection panels were generally dismissed as credible fire sources that would result in a core damage accident.

Once a list of realistic ignition sources was determined, the fire damage modeling defined fire scenarios and important target sets. For every identified target set, it was necessary to determine the geometric relationship between potential targets and fire sources. Three general types of fire scenarios were considered.

Targets located in the plume, directly above the fire source.

Targets located in the hot gas layer (outside the plume, but possibly in the ceiling jet)

Targets exposed to heating by thermal radiation, located next to the fire source.

The FIVE methodology was used to evaluate fire growth and propagation. The target temperature rises determined from the FIVE algorithms and look-up tables were then compared with target damage threshold criteria (temperature or heat flux), and if the criteria were not exceeded, the specified fire was screened from further analysis. (Note that this was performed using the automated QUICK FIVE worksheets.) If the damage threshold was exceeded, then the target was assumed failed and a PSA quantification run was performed to determine the conditional safe shutdown probability given the associated component damage of this fire scenario.

This analysis required collection of data for the following parameters.

Location of targets relative to a potential fire source

Damage threshold criteria for targets

The exposure fire peak intensity and total energy content

The fire enclosure volume and heat loss fraction

FIVE fire location factor (4 for corner, 2 for against wall, and 1 for center of room)

Fire suppression was generally not credited. When fire suppression was credited, the fire damage modeling included conservative assessments of equipment failure due to initial fire damage. The failure probabilities for the various fire suppression systems were taken from the FIVE methodology.

Credit was conservatively not taken for fire brigades

4.2.3.3 Damage Threshold Criteria

This analysis used basic FIVE methodology damage threshold criteria. The key criteria are repeated below.

A temperature of 700°F was used, per the FIVE methodology, as the failure temperature criterion for IEEE-383 qualified cables. The FIVE methodology suggests a temperature of 425°F for non-qualified cable. This value was not used as all cabling considered in this analysis is equivalent to IEEE-383.

In the case of radiant heat flux, the FIVE methodology prescribes a representative value of 1.0 Btu/sec/ft^2 for qualified cables. A value of 0.5 Btu/sec/ft^2 is suggested in the FIVE methodology as a screening value for non-qualified cable. This screening value and an intermediate value of 0.75 were used for equipment other than cables (e.g., switchgear, MCCs).

4.2.3.4 Fermi 2 Fire Damage Modeling Cases

Fire damage modeling was performed for all areas that did not screen in the initial screening quantifications of Phase II. The details of these analyses are presented on a compartment basis in Sections 4A and 4B.

4.2.4 Fire Detection And Suppression

Fire detection and suppression at Fermi 2 consists of both automatic and manual systems that use thermal, ionization, infra-red or photoelectric detection devices and gaseous or water suppression systems that use either flow switches or other indication devices to alarm system actuation. Complete descriptions of these systems are found in Section 9.5 of the UFSAR **[4.3]** and in the Fire Protection/Detection Systems DBD **[4.4]**.

Fire detection systems and, in most cases, fire suppression systems are not credited in the FIVE analysis. Where suppression is credited a specific discussion is provided in Sections 4A and 4B of this submittal. It was felt that the detection only systems would require manual response by the fire brigade and that by the time this action was taken, the fire damage would already have occurred. A similar approach was taken towards detection systems that activated gaseous or water suppression systems where it was felt that the alarm and initiation of the suppression medium would not occur until after damage occurred. The same approach was also taken for the wet pipe sprinkler systems where it was felt that prior to the heat in the compartment causing a fusible link to melt, a significant amount of damage would have occurred. In addition, no credit is taken for the fire brigade, or other plant personnel, detecting the incipient fire or extinguishing the fire prior to damage occurring.

Alarms for all safety related areas are received in the Control Room. Alarms for non-safety related areas are received either in the Control Room or in the Primary Access Portal (PAP). Alarms received in the Control Room are responded to in accordance with Abnormal Operating Procedure, "Plant Fires" **[4.5]**. Upon receipt of an alarm in the Control Room, the site fire alarm is sounded, the alarm is announced over the Hi Com System, the Fire Brigade Leader is notified and the fire brigade is activated. If there is no confirmation of a fire, the fire brigade stands by in the dress out area until either visual confirmation of a fire is received or they are directed to stand down. Alarms received in the PAP are the responsibility of Nuclear Security which will, if necessary, call for offsite assistance via the Control Room. In all cases a Nuclear Supervising Operator shall be sent to the scene **[4.5]**.

The following discussions about detection and suppression system are provided for information.

4.2.4.1 Fire Detection

Automatic fire detection systems at Fermi 2 perform two different functions. One type of detection system provides an alarm only, while the other provides both an alarm and initiation of suppression systems.

4.2.4.1.1 Alarm Only Function

Alarm only detection systems are installed in the Reactor Building, Auxiliary Building, Turbine Building, Radwaste Building, Office Service Building, General Service Water Pumphouse, Circulating Water Pump House, Auxiliary Boiler House, and the Residual Heat Removal (RHR) Complex. These detection systems consist of Ionization, Thermal, Photoelectric and Infrared detectors **[4.3, 4.4]**. The type of detection installed in an area is based upon the type of combustibles present in the area and the importance to safety. The detection systems were designed and installed using various NFPA codes as guidelines **[4.3]**. The detection systems have surveillances performed in accordance with plant procedures. The requirements for operability of the detection systems is prescribed by Section 9A.6 of the UFSAR **[4.3]**.

4.2.4.1.2 Alarm and Initiation Function

Alarm and Initiation detection systems are installed in the Auxiliary Building, RHR Complex, Radwaste, Office Service Building, Office Building Annex the PAP around the Main and Station Service Transformers. These detection systems consists of Thermal and Ionization detectors **[4.3 4.4**. The type of detection installed in an area is based upon the type of combustibles present in the area and the importance to safety. The detection systems were designed and installed using various NFPA codes as guidelines **[4.4]**. The detection systems have surveillances performed in accordance with plant procedures. The requirements for operability of the detection systems is proscribed by Section 9A.6 of the UFSAR **[4.4]**.

4.2.4.1.3 Loss of Fire Detection

Fire detection is not considered for the FIVE analysis. Therefore loss of fire detection would have no impact on the calculated results. Loss of the fire detection system due to loss of power or circuit failures will result in a trouble alarm coming in to the Control Room. The compensatory measures for inoperable fire detection systems are provided in UFSAR Section 9A.6 **[4.3]**.

4.2.4.2 Fire Suppression

Fire suppression systems consist of wet pipe and pre-action sprinkler systems, deluge systems, automatic CO_2 systems, automatic Halon systems, manual CO_2 hose stations, manual water hose stations, fire hydrants.

There are manual fire extinguishers: CO_2, halon, dry chemical, water pressure etc. located throughout the buildings suitable for the hazard in the area.

Fire brigade members, operations personnel and fire watches are trained in the proper operation of the suppression equipment.

4.2.4.2.1 Automatic Suppression

This analysis generally took no credit for automatic suppression system actuation. The only compartment where a suppression system is credited is the Auxiliary Building Basement (01AB), and the analysis specifically addressed the effect of fire suppression on mitigation of the damage due to a fire and the resultant effect on the core damage frequency.

Three types of automatic suppression are installed at Fermi: carbon dioxide (CO_2), halon, and water. These systems are installed in accordance with guidance provided by NFPA codes **[4.4]**.

Automatic CO_2 systems are installed in the Auxiliary Building and RHR Complex **[4.3]**. These systems are actuated either by manual actions, locally or from the Main Control Room, or by the associated fire detection system. They are designed to provide more than a single discharge into the affected compartment in order to provide the proper concentration and/or soak time for the suppression medium. The compartments with automatic CO_2 systems in the Auxiliary Building are the Cable Tunnel (05AB), the Cable Tray Area (08AB), Miscellaneous Rooms (11AB), and the Standby Gas Treatment Rooms (14AB) in the standby gas treatment system charcoal filters units.

Automatic Halon systems are installed in the Auxiliary Building, Office Service Building, Office Building Annex and the PAP **[4.3]**. These systems are activated either by manual actions, locally or from the Main Control Room, or by the associated fire detection system. They are designed to provide more than a single discharge into the affected compartment in order to achieve the proper concentration for the suppression medium. The areas with halon systems in the Auxiliary Building are the Relay Room (03AB), Cable Spreading Room (07AB), and the Computer Room above the Control Room (09AB). The halon systems in the other buildings do not protect safety related equipment and have no impact on the calculated core damage frequency.

Automatic water suppression systems are installed in the Reactor Building, Auxiliary Building, Turbine Building, Radwaste Building, RHR Complex, Office and Service Building, General Service Water Pump House, and the Onsite Storage Facility **[4.3]**. The Reactor Building, Auxiliary Building, and RHR Complex consist of wet pipe sprinklers systems. The other buildings have a combination of wet pipe, deluge and pre-action systems. The water for these systems is provided by the fire water suppression system. This system is primarily supplied by the general service water pumps with an electric fire pump and a diesel fire pump in standby. The electric fire pump and the diesel fire pump will start when the pressure in the system drops to a preset value. The diesel fire pump will also start if there is a loss of power thus ensuring a supply of suppression water.

4.2.4.2.2 Manual Suppression

Manual suppression of fires will be achieved by use of the fire extinguishers located throughout the plant, use of the manual hose station hoses or manual actuation of existing gaseous or water based suppression systems.

The fire brigade responds to all fires inside the plant and provides support, as required for all fires outside the protected area. Frenchtown fire department is responsible for fires outside the protected area and provides support for fires inside the protected area.

Manual suppression is not taken credit for in this analysis. The fire fighting training provided to the fire brigade, operations personnel, and the general awareness of fire hazards, addressed in the general employee training, along with the fire detection systems in all safety related areas will result in limiting the damage due to a fire and preventing its spread beyond the analyzed compartment boundaries.

4A FIRE HAZARDS ANALYSIS - ALL AREAS EXCEPT AUXILIARY BUILDING

This section provides a description of the overall fire hazards analysis for all plant areas except the Auxiliary Building.

4A.0 FIRE HAZARDS METHODOLOGY

The FIVE methodology and this evaluation consist of three phases:

Phase I: Fire Area Screen (Qualitative Analysis)

Phase II: Critical Fire Compartment Screen (Quantitative Analysis)

> NOTE: For this portion of the analysis, Section 4A, the FIVE terminology is not strictly followed. Phase II is treated in two steps. The first is identified as Phase II and an initial screening analysis is performed. Compartments that did not screen were then reanalyzed in what was called Phase III. Phase III is in actuality a continuation of the FIVE Phase II analysis using more refined techniques to achieve the desired screening result of <1E-6/year

Phase III: Plant Walkdown/Verification and Documentation

The three phase process of the FIVE methodology is shown in Figure 4-1.

4A.0.1 Fire Zones

Several areas (e.g. first floor, second floor, third floor and fourth floor) within the Reactor Building are open to each other due to the open stairwells and unsealed penetrations in the floors and ceilings. These barriers do not meet the requirements defined by Section 2.2 of the FIVE Methodology for fire area barriers. However, in Phase III these areas have been individually evaluated as separate fire areas (e.g. RB05, RB06, RB07 and RB08), as products of combustion are not considered to propagate between fire areas. This approach is based on the Fire Hazards Analysis of the Reactor Building in the Fermi 2 UFSAR.

The RHR Building is separated into two fire areas: RHR1 and RHR2. The division of RHR1 and RHR2 is established at the three hour fire rated common barrier at column line seven (7). This division is based upon the requirements defined by Section 2.2 of the FIVE Methodology for fire areas.

In Phase III, fire areas TB, RHR1 and RHR2 are further subdivided into smaller fire compartments. The compartmentalization is discussed in greater detail in the appropriate sections for Phase III.

4A.0.2 Safe Shutdown Equipment

The identification of which Safe Shutdown equipment would be impacted by a fire event in the fire areas were determined by the Fermi 2 database 'Combine1.db'. In addition to Safe Shutdown equipment, the identification of a set of Balance of Plant equipment impacted by a fire event in the fire areas were determined by the Fermi 2 database 'Combop.db'. The PSA model was adjusted so as not to credit any equipment (Safe Shutdown or Balance of Plant) for which routing information was not available.

4A.1 PHASE I SCREENING

The Phase I Fire Compartment Interaction Analysis (FCIA) was performed using the QUICK FIVE program. The purpose of the Fire Compartment Interaction Analysis (FCIA) is to evaluate the potential for fire spread across fire compartment boundaries within a fire area. The Fire Compartment Interaction Analysis (FCIA) also evaluated the potential for fire spread across barriers separating fire areas. The results of the Phase I FCIA is contained in Appendix A.1.3 of reference 4.36.

No fire areas were screened in the FCIA due to lack of routing information for the Reactor Protection System. Thus all fire are assumed to cause a SCRAM. As a result, all fire areas remain unscreened in Phase I and were evaluated in the Phase II analysis.

4A.2 PHASE II FIRE HAZARDS RESULTS

4A.2.1 Screening Results

The results of the Phase II analysis is contained in Table 4-5 . The Ignition Source Data Sheet (ISDS) for each fire area is contained in Appendix A.2.1 of reference 4.36. Several fire areas screened out in Phase II when the CDF value was less than 1.0E-6 events per reactor year. All remaining unscreened fire areas are evaluated in the Phase III analysis.

4A.2.2 Fire Initiation Frequency

The compartment fire frequency (F_1) for each fire area is calculated from the Ignition Source Data Sheet (ISDS) of the QUICK FIVE program for each fire area. The ignition sources identified in each fire area were obtained from the Fermi 2 database 'Comi. 'ie2.db'.

Several fire areas (ABFST, CST, HSF, OBA&TSC, ONSB and OSB) contained neither Safe Shutdown or Balance of Plant equipment or circuits. These fire areas were candidates for screening in the Phase I FCIA analysis, had routing information for the Reactor Protection System been available. These fire areas did not have an individual ignition frequency calculated since the calculation of an ignition frequency for these fire areas would have artificially lowered the ignition frequencies for other plant fire areas. These fire areas contained electrical cabinets, HVAC subsystems, a few pumps, etc. Therefore, the worst case ignition frequency, 5.68E-2/yr (of fire area TB), was conservatively applied to these fire areas. As the baseline CCDF (1.3E-5) is applied, these areas screen from further evaluation with a CDF of 7.38E-7/yr.

4A.2.3 Calculation of Conditional Core Damage Frequency (CCDF)

The CCDF (P_2 value) is calculated by considering the loss of all Safe Shutdown and Balance of Plant circuits and equipment located in each fire area. See Section 4.3. The identification of all Safe Shutdown and Balance of Plant circuits and equipment located in each fire area is obtained from the Fermi 2 databases 'Combine1.db' and 'Combop.db' respectively. The product CDF is calculated by multiplying the CCDF (P_2 value) and the compartment fire frequency (F_1) for each fire area. Values of CCDF aare summarized in Table 4-4.

The effectiveness of the automatic or manual fire detection and fire suppression systems was not analyzed within the fire areas. This results in a P_f of 1.0, representing that damage can occur from the fixed ignition sources. Considering a P_f of 1.0, P_{tc} is omitted in the calculation of P_3. Therefore, since $P_3 = P_f + P_{tc}$, P_3 is equal to 1.0 and no further analysis was performed in Phase II.

4A.3 FIRE GROWTH AND PROPAGATION -PHASE III

4A.3.1 Approach

The Phase III analysis utilizes simplified fire modeling (as described in the FIVE Methodology) to more accurately determine which Safe Shutdown equipment and circuits would be impacted by a fire event at the individual ignition sources within the compartment. The raceway routing of the Safe Shutdown equipment was obtained from the Fermi 2 spreadsheet 'Complcab.xls'. Due to the inconsistent amount of raceway routing for Balance of Plant equipment in 'Complcab.xls', Balance of Plant equipment was assumed lost in the Phase III analysis unless specifically determined to be free from damage.

A generic fire model was developed for the individual ignition source to accomplish this task. The generic model established an 'area of damage' around the individual ignition source. Safe Shutdown raceways are reviewed to determine if they are routed through the 'area of damage'. Safe Shutdown circuits routed through raceways within the 'area of damage' are considered damaged and the associated Safe Shutdown equipment lost.

4A.3.2 Fire Growth and Propagation Modeling

Sets of damaged equipment are developed for each individual ignition source. Conditional Core Damage Frequencies (CCDFs) are calculated based on the sets of damaged equipment. Each Conditional Core Damage Frequency (CCDF) is multiplied by the ignition frequency of the fire initiator (individual ignition source) to obtain an individual CDF for each scenario. The sum of all individual CDFs within the fire area provides a total CDF for the fire area. Fire areas with a total CDF less than 1.0E-6 events per reactor year are screened from further evaluation.

The Heat Release Rates used in the fire models for electrical cabinets in the Phase III analysis were developed from the Unit Heat Release Rate for XPE/Neoprene presented in Table 1E of the FIVE Methodology. The Unit Heat Release Rate was multiplied by a 'fire footprint' of two to four square feet, yielding Heat Release Rates for electrical cabinets of approximately sixty (60) to one-hundred (100) BTU/second. Additionally, Heat Release Rates were also considered from NUREG/CR-4527/1 of 2 **[4.17]**.

The Heat Release Rate used in the fire models for oil spills is 16,200 BTU/second. This Heat Release Rate is based on a unit Heat Release Rate of 135 BTU/second/square foot and a spill area of 120 square feet per gallon.

The ambient temperature used in the fire modeling for each fire area was obtained from the 'Normal Weighted Average Temperature' of Table 1 of Reference **[4.16]**.

For the Phase III analysis, a probability of transient exposure (P_{tc}) is applied to the transient ignition source. The probability of transient exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology as:

$$P_{tc} = (P_{fst})(u)(p)(x/2)\ln(1/x), \text{ where } x = F_{ccl}/F_w$$

Probability of suppression of a transient fire, P_{fst}, is set at 1.0 (no credit for automatic fire suppression). The probability of the transient located in a position to cause damage, u, is conservatively set at 1.0. The probability of the transient being exposed , p, is set at 0.10 based upon Fermi 2's transient control program **[4.19]** which meets the requirements defined in Section 6.3.7.2 of the FIVE Methodology. The number of findings per year, F_{ccl}, of the transient combustible found in violation of plant procedures is assumed to be 1.0. The frequency of combustible material inspections is twenty-six (26) per year based upon bi-weekly housekeeping inspections **[4.20, 4.21]** established at Fermi 2. Therefore, the probability of transient exposure (P_{tc}) is equal to 6.26E-3.

4A.3.3 Fire Propagation Analysis Assumptions

Due to the low amount of combustibles (cabling) associated with instrument racks, ignition of instrument racks are not considered to present a realistic fire hazard to other circuits and equipment. For the Phase III evaluation, ignition of instrument racks were not modeled unless Safe Shutdown circuits and equipment were located directly adjacent or above the instrument rack. Loss of any Safe Shutdown instrument racks would be accounted for in the PSA models.

Fire protection panels are not considered to propagate flame due to the relatively small, enclosed structure of the panel and the sealed conduits entering the panel. For the Phase III evaluation, ignition of fire protection panels was not modeled.

Electrical cabinet fires were modeled as propagating from the top of the cabinet unless information was available from plant personnel or walkdowns indicating that the cabinet was enclosed, covered with a drip pan, or vented elsewhere (e.g. side ventilation).

Safe Shutdown equipment is only considered lost 'downstream' of the damaged circuit. EXAMPLE: A circuit is damaged providing power from a MCC to a pump. The pump is considered lost. The MCC is not considered lost based on circuit protection from a 'hot short'.

For calculation of the Scenario specific Core Damage Frequencies (CDFs) in Phase III, Balance of Plant equipment located within the fire area was considered inoperable unless specifically routed, modeled and shown to be free from damage.

Neither automatic detection or automatic suppression were modeled to prevent damage to targets. Therefore the probabilities associated with these systems are not utilized in this evaluation.

4A.3.4 Fire Propagation Results

The results (Total CDF values) of the Phase III analysis is presented in Table 4-6. The Phase III analysis for the unscreened fire areas is described in detail in Section 4A.3.4.2 and 4A.3.4.3

4A.3.4.1 Results of FIVE Worksheet and COMPBRN IIIe Calculations

The fire modeling worksheets for the fire areas addressed in Phase III are presented in Appendix A.4 of the Vectra Report **[4.36]**. One fire area, RB06, remains unscreened at the conclusion of the Phase III analysis.

4A.3.4.2 Reactor Building Zones

4A.3.4.2.1 Phase III Analysis for Fire Area 01RB

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the Reactor Building Torus Room, ignition sources are identified which adversely impact Safe Shutdown circuits.

The ignition frequency for 01RB is 1.44E-3/yr. This frequency is comprised of six electrical cabinets (2.7E-4/yr) and the transient load (1.17E-3/yr) frequencies. Upon a review of plant documents, only three of the electrical cabinets are physically located in the Torus Room. The three electrical cabinets are a combination of one instrument rack and two 480 volt service switches. The instrument racks and service switches do not present a significant heat release rate and are not considered to have sufficient BTU content to damage any component other than themselves.

The loss of the three electrical cabinets have no impact on the PSA model , and the Base Model CCDF (1.3E-5) is applied to these ignition sources. The ignition frequency for this scenario is (3/6)(2.7E-4) = 1.35E-4/yr.

The CDF for this scenario is: (1.35E-4)(1.3E-5) = 1.8E-9/yr

Fire modeling demonstrated that a transient fire of a thirty-two gallon waste container would not develop a plume height sufficient to damage nor ignite cable trays in 01RB. Damage was recognized when the transient was placed against the Torus area walls. This damage and those circuits identified as safe from damage are represented in PSA Model RB01C.

The CDF for this scenario is: (1.17E-3)(6.26E-3)(6.3E-2) = 4.61E-7/yr

SCENARIO	**CDF**
Loss of the Electrical Cabinets	1.8E-9/yr
Transient Load	4.61E-7/yr
TOTAL:	4.6E-7/yr

The results of this analysis indicate that the Fire Area 01RB is not fire risk significant. The CDF for this fire area is 4.6E-7/yr, allowing the Fire Area to screen. No recommendations for further action are necessary.

4A.3.4.2.2 Phase III Analysis for Fire Area 03RB

As the Phase II CDF was 1.19E-6/yr, slightly over the 1E-6/yr screening criteria, the Phase III analysis for this fire area involves an examination of the ignition sources. The CDF calculated by PSA Model RB03 for loss of the entire area is 3.5E-4/yr. The CCDF calculated by PSA Model RB03 is conservatively applied to all the scenarios.

Ignition of Electrical Cabinets:

Three of the seven electrical cabinets are battery operated emergency lights and do not constitute a credible ignition source within the fire area. Therefore, the ignition frequency for the remaining electrical cabinets is (4/7)(3.1E-4) = 1.77E-4/yr. The CCDF calculated by PSA Model RB03, 3.5E-4, is applied to this scenario.

CDF for ignition of the electrical cabinets is: (1.77E-4)(3.5E-4) = 6.2E-8/yr

Ignition of Pumps:

The CCDF calculated by PSA Model RB03, 3.5E-4, is applied to this scenario. The ignition frequency for pumps is 1.9E-3/yr.

CDF for ignition of the pumps is: (1.9E-3)(3.5E-4) = 6.65E-7/yr

Ignition of HVAC:

The CCDF calculated by PSA Model RB03, 3.5E-4, is applied to this scenario. The ignition frequency for HVAC is 5.7E-5/yr.

CDF for ignition of the HVAC is: (5.7E-5)(3.5E-4) = 2.0E-8/yr

Ignition of Transient Combustibles:

The CCDF calculated by PSA Model RB03, 3.5E-4, is applied to this scenario. The ignition frequency for transient combustibles is 1.17E-3/yr. The probability of fire exposure from transient combustibles (P_{tc}) is evaluated. P_{tc} is calculated in Section 4A.3.1.2 to be 0.00626.

CDF for ignition of the transients is: (1.17E-3)(3.5E-4)(.00626) = 2.6E-9/yr

The individual CDFs for the above scenarios is added to give a total CDF for the fire area:

SCENARIO	**CDF**
Ignition of Electrical Cabinets	6.2E-8/yr
Ignition of Pumps	6.65E-7/yr
Ignition of HVAC	2.0E-8/yr
Ignition of Transients	2.6E-9/yr
TOTAL	7.5E-7/yr

The results of this analysis indicate that the Fire Area 03RB is not fire risk significant. The CDF for this fire area is 7.5E-7/yr, allowing the Fire Area to screen. No recommendations for further action are necessary.

4A.3.4.2.3 Phase III Analysis for Fire Area 04RB

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the reactor building corridor area ignition sources are identified which adversely impact Safe Shutdown circuits.

The frequencies for the ignition sources are:

Instrument Rack and Fire protection Panel:	5.6E-5/yr
Transients:	1.17E-3/yr
Total:	1.23E-3/yr

The CCDFs calculated for this area by PSA Models RB04D1 and RB04D2 are:

Loss of Division One CCDF: 4.2E-3

Loss of Division Two CCDF: 4.4E-3

Fixed Combustible Scenario:

Fire modeling has shown that considering the low BTU content of the Instrument Rack and Fire Protection Panel and the spatial separation of the divisions, damage will not occur to both divisions in 04RB, given ignition of either fixed sources. Therefore, the CCDF for loss of

Division Two (4.4E-3) is conservatively applied to the ignition frequency for the instrument rack and fire protection panel.

The CDF for the Fixed Combustible Scenario : (5.6E-5)(4.4E-3) = 2.46E-7/yr

Transient Combustible Scenario:

Fire modeling has also shown that a one quart oil spill (approximating a transient with high BTU content) would cause localized damage only in 04RB. The exposure in the southern portion of 04RB would damage Division Two components as well as the loss of E5150F022 (Division One). An exposure in the northern portion of 04RB would damage only Division One components. The PSA model demonstrated that the loss of E5150F022 would not impact the CCDF for loss of Division One, and therefore the CCDF for loss of Division Two (4.4E-3) is applied to the transient ignition frequency.

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. Therefore, utilizing P_{tc}, the CDF for the Transient Combustible Scenario is: (1.17E-3)(4.4E-3)(6.27E-3) = 3.2E-8/yr

The individual CDFs for the above scenarios are added to obtain the total CDF for the area: 2.46E-7 + 3.2E-8 = 2.78E-7/yr

The results of this analysis indicate that the Fire Area 04RB is not fire risk significant. The total CDF for this fire area is 2.78E-7/yr, allowing the Fire Area to screen. No recommendations for further action are necessary.

4A.3.4.2.4 Phase III Analysis for Fire Area 05RB

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the Reactor Building First Floor, ignition sources are identified which adversely impact Safe Shutdown circuits.

Ignition of Electrical Cabinet R1700S017B Scenario 1

Simplified fire modeling has shown that ignition of R1700S017B damages circuits for safe shutdown components H2100P037 and H2100P038. The CCDF calculated for this damage set enveloped by PSA Model RB05S1 is 1.2E-4. Since the ignition frequency is (1/246)(1.1E-2) = 4.5E-5/yr:

CDF for Ignition of R1700S017B: (4.5E-5)(1.2E-4) = 5.4E-9/yr.

Ignition of Electrical Cabinet H2100P475 Scenario 2

Simplified fire modeling has shown that ignition of H2100P475 does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set enveloped by PSA Model RB05S2 is 9.0E-5. Given that the ignition frequency is (1/246)(1.1E-2) = 4.5E-5/yr:

CDF for Ignition of H2100P475: (4.5E-5)(9.0E-5) = 4.0E-9/yr.

Ignition of Electrical Cabinet H2100P560 Scenario 3

Simplified fire modeling has shown that ignition of H2100P560 damages circuits for safe shutdown component H2100P475, as well as H2100P560. The CCDF calculated for this damage set enveloped by PSA Model RB05S2 is 9.0E-5. Given that the ignition frequency is (1/246)(1.1E-2) = 4.5E-5/yr:

CDF for Ignition of H2100P560: (4.5E-5)(9.0E-5) = 4.0E-9/yr.

Ignition of Electrical Cabinet H2100P009 or H2100P402D Scenario 4

Simplified fire modeling has shown that ignition of H2100P009 or H2100P402D damages circuits for safe shutdown component H2100P623. The CCDF calculated for this damage set by PSA Model RB05S7 is 2.3E-4. The ignition frequency for two cabinets is (2/246)(1.1E-2) = 8.9E-5/yr:

CDF for Ignition of H2100P009 or H2100P402D: (8.9E-5)(2.3E-4) = 2.05E-8/yr.

Ignition of Electrical Cabinet H2100P035 or H2100P402F Scenario 5

Simplified fire modeling has shown that ignition of H2100P035 or H2100P402F damages circuits for safe shutdown component H2100P035. The CCDF calculated for this damage set by PSA Model RB05S3 is 1.2E-4. Since the ignition frequency for two cabinets is (2/246)(1.1E-2) = 8.9E-5/yr:

CDF for Ignition of H2100P035 or H2100P402F: (8.9E-5)(1.2E-4) = 1.07E-8/yr.

Ignition of Electrical Cabinet H2100P015 Scenario 6

Simplified fire modeling has shown that ignition of H2100P015 damages circuits for safe shutdown components H2100P035 and H2100P623. The CCDF calculated for this damage set by PSA Model RB05S8 is 9.4E-4. Since the ignition frequency is (1/246)(1.1E-2) = 4.5E-5/yr:

CDF for Ignition of H2100P015: (4.5E-5)(9.4E-4) = 4.23E-8/yr.

Ignition of Electrical Cabinet H2100P474 Scenario 7

Simplified fire modeling has shown that ignition of H2100P474 does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set by PSA Model RB05S4 is 8.9E-5. Since the ignition frequency is (1/246)(1.1E-2) = 4.5E-5/yr:

CDF for Ignition of H2100P474: (4.5E-5)(8.9E-5) = 4.0E-9/yr.

Ignition of Electrical Cabinet R1600S002E Scenario 8

Simplified fire modeling has shown that ignition of R1600S002E does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set by PSA Model RB05S0 is 9.0E-5. Since the ignition frequency is (1/246)(1.1E-2) = 4.5E-5/yr:

CDF for Ignition of R1600S002E: (4.5E-5)(9.0E-5) = 4.0E-9/yr.

Ignition of Electrical Cabinet R1600S021 or R1600S092 Scenario 9

Simplified fire modeling has shown that ignition of R1600S021 or R1600S092 damages circuits for safe shutdown component H1100P823. The CCDF calculated for this damage set by PSA Model RB05S0 is 9.0E-5. Since the ignition frequency is (12/246)(1.1E-2) = 5.37E-4/yr:

CDF for Ignition of R1600S021 or R1600S092: (5.37E-4)(9.0E-5) = 4.83E-8/yr.

Ignition of Electrical Cabinet R1600S004B Scenario 10

Simplified fire modeling has shown that ignition of R1600S004B does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set by PSA Model RB05S5 is 1.2E-4. Since the ignition frequency is (40/246)(1.1E-2) = 1.79E-3/yr:

CDF for Ignition of R1600S004B: (1.79E-3)(1.2E-4) = 2.15E-7/yr.

Ignition of Electrical Cabinet R1600S002B Scenario 11

Walkdown of electrical cabinet R1600S002B indicated that the cabinet was totally enclosed, without any ventilation. The lack of ventilation or other openings prevents the propagation of flame or products of combustion from within the electrical cabinet. Ignition of cabinet R1600S002B will not damage other plant components or raceways. Therefore, the PSA model for this scenario credits all available plant equipment unless supported by cabinet R1600S002B. The CCDF calculated for this damage set by PSA Model S002B is 2.7E-6. The ignition frequency is (39/246)(1.1E-2) = 1.74E-3/yr:

CDF for Ignition of R1600S002B: (1.74E-3)(2.7E-6) = 4.7E-9/yr.

Ignition of Electrical Cabinet H2100P626 Scenario 12

Simplified fire modeling has shown that ignition of H2100P626 does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set by PSA Model RB05S9 is 3.7E-3. Since the ignition frequency is (1/246)(1.1E-2) = 4.5E-5/yr:

CDF for Ignition of H2100P626: (4.5E-5)(3.7E-3) = 1.65E-7/yr.

Ignition of Remaining Electrical Cabinets

The remaining 144 electrical cabinets consist of instrument racks, lighting panels, remote lamp panels and battery operated emergency lights. None of the remaining electrical cabinets are safe shutdown components. The base model CCDF of 1.3E-5 is applied to these components. The ignition frequency is (144/246)(1.1E-2) = 6.44E-3/yr:

CDF for Ignition of Remaining Electrical Cabinets: (6.44E-3)(1.3E-5) = 8.37E-8/yr.

Ignition of Fire Protection Panels

The existing fire protection panels are not safe shutdown components. The base model CCDF of 1.3E-5 is applied to these components. The ignition frequency is 1.6E-4/yr:

CDF for Ignition of Fire Protection Panels: (1.6E-4)(1.3E-5) = 2.1E-9/yr.

Ignition of Transformer R1700S016B Scenario 1

Simplified fire modeling has shown that ignition of R1700S016B damages circuits for safe shutdown components H2100P037 and H2100P038. The CCDF calculated for this damage set enveloped by PSA Model RB05S1 is 1.2E-4. Therefore, given that the ignition frequency is (1/5)(4.9E-4) = 9.8E-5/yr:

CDF for Ignition of R1700S016B: (9.8E-5)(1.2E-4) = 1.18E-8/yr.

Ignition of Remaining Transformers Scenario

Simplified fire modeling has shown that ignition of R1600S128B, R1700S008, R1700S016A and R3600S247 does not result in damage to circuits for other safe shutdown components. These transformers are not safe shutdown components. The base model CCDF is applied to these components. Since the ignition frequency is (4/5)(4.9E-4) = 3.92E-4/yr:

CDF for Ignition of Remaining Transformers: (3.92E-4)(1.3E-5) = 5.1E-9/yr.

Ignition of Fan T4100C007 Scenario

Simplified fire modeling has shown that ignition of this fan does not result in damage to circuits for any safe shutdown components. The base model CCDF is applied to this components. Since the ignition frequency is 2.9E-5/yr:

CDF for Ignition of T4100C007: (2.9E-5)(1.3E-5) = 3.8E-10/yr.

Ignition of Transient Combustible Scenario

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. As calculated earlier, P_{tc} is 0.00626. For conservatism the scenario providing the highest CCDF (calculated by PSA Model RB05S9 as 3.7E-3) is applied to the transient combustibles. The ignition frequency is (1.2E-4 + 7.2E-4 + 3.3E-4) = 1.17E-3/yr

CDF for Ignition of Transient Combustibles: (1.17E-3)(3.7E-3)(0.00626) = 2.7E-8/yr

The CDF for the above scenarios are added to obtain the total CDF for the area:

SCENARIO	CDF
Scenario 1	5.4E-9/yr
Scenario 2	4.0E-9/yr
Scenario 3	4.0E-9/yr
Scenario 4	2.05E-8/yr
Scenario 5	1.07E-8/yr
Scenario 6	4.23E-8/yr
Scenario 7	4.0E-9/yr
Scenario 8	4.0E-9/yr
Scenario 9	4.83E-8/yr
Scenario 10	2.15E-7/yr
Scenario 11	4.7E-9/yr
Scenario 12	1.65E-7/yr
Remaining Elect.	8.37E-8/yr
Fire Prot. Panels	2.1E-9/yr
Xmfrs Scenario 1	1.18E-8/yr
Rem. Transformers	5.1E-9/yr
HVAC Fan	3.8E-10/yr
Transient Scenario	2.71E-8/yr

TOTAL 6.6E-7/yr

The results of this analysis indicate that the Fire Area 05RB is not fire risk significant. The total CDF for this fire area is 6.6E-7/yr, allowing this fire area to screen. No recommendations for further action are necessary.

4A.3.4.2.5 Phase III Analysis for Fire Area 06RB

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the Reactor Building Second Floor, ignition sources are identified which adversely impact Safe Shutdown circuits.

Ignition of H2100P584G, P4400C001A and T4100B034 Scenario 1:

Simplified fire modeling has shown that ignition of H2100P584G would damage circuits supporting components P4400N401A and H2100P584G. Simplified fire modeling has shown that ignition of P4400C001A and T4100B034 would damage circuits supporting components H2100P472, H2100P590, P4400F602A, T4100B034, H1100P613 and P4400C001A. PSA Model RB6S01 calculated the CCDF for this damage set as 8.9E-5. Considering the ignition frequency of one electrical cabinet, one pump and one HVAC combined together is (1/147)(6.5E-3) + (1/5)(2.3E-3) + (1/6)(1.7E-4) = 5.33E-4/yr, the following can be calculated:

CDF for Ignition of H2100P584G and P4400C001A: (5.33E-4)(8.9E-5) = 4.74E-8/yr

Ignition of H2100P584I, T4100B035, H2100P448, H2100P473, & P4400C001B Scenario 2:

Simplified fire modeling has shown that ignition of H2100P584I would damage circuits supporting components P4400F401B and H2100P584I. Simplified fire modeling has shown that ignition of T4100B035 would damage circuits supporting components H1100P612 and T4100B035. Simplified fire modeling has shown that ignition of H2100P448 would damage circuits supporting components P4400F602B, H1100P612 and H2100P448. Simplified fire modeling has shown that ignition of H2100P473 would damage circuits supporting components H2100P584I and H2100P473. Simplified fire modeling has shown that ignition of P4400C001B would damage circuits supporting components P4400F602B, H1100P612, T4100B035, H2100P591, E1150F023 and P4400C001B. PSA Model RB6S02 calculated the CCDF for this damage set as 8.9E-5. The ignition frequency of three electrical cabinets, one HVAC unit and one pump combined together is (3/147)(6.5E-3) + (1/5)(2.3E-3) + (1/6)(1.7E-4) = 6.21E-4/yr.

CDF for Ignition of H2100P584I, T4100B035, H2100P448, H2100P473, & P4400C001B: (6.21E-4)(8.9E-5) = 5.53E-8/yr

Ignition of R1600S003B, H2100P625 & R1600S003H Scenario 3:

Walkdown of electrical cabinets R1600S003B, H2100P625 and R1600S003H indicated that the cabinets are totally enclosed, without ventilation. The lack of ventilation or other openings prevents the propagation of flame or products of combustion from within the electrical cabinets. Ignition of cabinets R1600S003B, H2100P625 and R1600S003H will not damage other plant components or raceways. Therefore, the PSA model for this scenario credits all available plant equipment unless supported by cabinets R1600S003B, H2100P625 and R1600S003H. PSA Model S003B calculated the CCDF for this damage set as 7.9E-5. The ignition frequency of forty-four electrical cabinets is (44/147)(6.5E-3) = 1.9E-3/yr.

CDF for Ignition of R1600S003B, H2100P625 & R1600S003H: (7.5E-5)(1.9E-3) = 1.43E-7/yr

Ignition of H2100P554, H2100P284, R1600S100, R1600S221, H2100P522, H2100P002, H2100P559, H2100P282, H2100P595A, R1600S099, H2100P591, H2100P590, H2100P584F & H2100P595B Scenario 4:

Simplified fire modeling has shown that ignition of H2100P554 would damage circuits supporting component H2100P284. Simplified fire modeling has shown that ignition of H2100P284 would damage circuits supporting component H2100P554. Simplified fire modeling has shown that ignition of R1600S100, R1600S221, and H2100P522 would damage circuits supporting component H2100P002. Simplified fire modeling has shown that ignition of H2100P002 would damage circuits supporting components R1600S100, R1600S221, and H2100P522. Simplified fire modeling has shown that ignition of H2100P559 would damage circuits supporting component H2100P559. Simplified fire modeling has shown that ignition of H2100P282 would damage circuits supporting component H2100P282. Simplified fire modeling has shown that ignition of H2100P595A would damage circuits supporting component H2100P595A. Simplified fire modeling has shown that ignition of R1600S099 would damage circuits supporting components R1600S005G and R1600S099. Simplified fire modeling has shown that ignition of H2100P591 would damage circuits supporting components T4100B035, H1100P612, H2100P584I and H2100P591. Simplified fire modeling has shown that ignition of H2100P590 would damage circuits supporting component H2100P590. Simplified fire modeling has shown that ignition of H2100P584F would damage circuits supporting component H2100P584F. Simplified fire modeling has shown that ignition of H2100P595B would damage circuits supporting component H2100P595B. PSA Model RB6S04 calculated the CCDF for this damage set as 8.9E-5. The ignition frequency of fourteen electrical cabinets combined together is (14/147)(6.5E-3) = 6.19E-4/yr.

CDF for Ignition of H2100P554, H2100P284, R1600S100, R1600S221, H2100P522, H2100P002, H2100P559, H2100P282, H2100P595A, R1600S099, H2100P591, H2100P590, H2100P584F & H2100P595B: (6.19E-4)(8.9E-5) = 5.51E-8/yr

Ignition of R1600S003J, H2100P627 & R1600S003D Scenario 5:

Simplified fire modeling has shown that ignition of H2100P627, R1600S003J and R1600S003D would damage circuits supporting components H2100P627, R1600S003J and R1600S003D. PSA Model RB6S05 calculated the CCDF for this damage set as 6.9E-4. The ignition frequency of sixteen electrical cabinets is (16/147)(6.5E-3) = 7.07E-4/yr.

CDF for Ignition of R1600S003J, H2100P627 & R1600S003D: (6.9E-4)(7.07E-4) = 4.88E-7/yr

Ignition of H2100P482 Scenario 6:

Simplified fire modeling has shown that ignition of H2100P482 would damage circuits supporting components H2100P590, H2100P584G, H2100P559, H1100P613 and H2100P482. PSA Model RB6S07 calculated the CCDF for this damage set as 3.2E-4. The ignition frequency of this electrical cabinet is (1/147)(6.5E-3) = 4.4E-5/yr.

CDF for Ignition of H2100P482: (3.2E-4)(4.4E-5) = 1.41E-8/yr

Ignition of H2100P004 Scenario 7:

Simplified fire modeling has shown that ignition of H2100P004 would damage circuits supporting components H2100P623, H2100P004, E2150F004A, and E2150F005A. PSA Model RB6S08 calculated the CCDF for this damage set as 1.5E-4. The ignition frequency of one electrical cabinet is (1/147)(6.5E-3) = 4.4E-5/yr.

CDF for Ignition of H2100P004: (1.5E-4)(4.4E-5) = 6.6E-9/yr

Ignition of H2100P402A Scenario 8:

Simplified fire modeling has shown that ignition of H2100P402A would damage circuits supporting components H2100P004. PSA Model RB6S09 calculated the CCDF for this damage set as 1.5E-4. The ignition frequency of one electrical cabinet is (1/147)(6.5E-3) = 4.4E-5/yr.

CDF for Ignition of H2100P402A: (1.5E-4)(4.4E-5) = 6.6E-9/yr

Ignition of H2100P005 Scenario 9:

Simplified fire modeling has shown that ignition of H2100P005 would damage circuits supporting components H2100P005. PSA Model RB6S10 calculated the CCDF for this damage set as 1.6E-4. The ignition frequency of one electrical cabinet is (1/147)(6.5E-3) = 4.4E-5/yr.

CDF for Ignition of H2100P005: (1.6E-4)(4.4E-5) = 7.1E-9/yr

Ignition of R1600S005C Scenario 10:

Walkdown of electrical cabinet R1600S005C indicates that the cabinet is totally enclosed, without ventilation. The lack of ventilation or other openings prevents the propagation of flame or products of combustion from within the electrical cabinet. Ignition of cabinet R1600S005C will not damage other plant components or raceways. Therefore, the PSA model for this scenario credits all available plant equipment unless supported by cabinet R1600S005C. PSA Model MS005C calculated the CCDF for this damage set as 6.9E-5. The ignition frequency of forty-one electrical cabinets is (41/147)(6.5E-3) = 1.81E-3/yr.

CDF for Ignition of R1600S005C: (6.9E-5)(1.81E-3) = 1.25E-7/yr

Ignition of Remaining Electrical Cabinets Scenario 11:

Simplified fire modeling for the remaining electrical cabinets (23) showed there would be no damage to any other SSD circuits. Therefore, the base model CCDF is applied to the remaining ignition sources. The ignition frequency for the remaining electrical cabinets is (23/147)(6.5E-3) =1.02E-3/yr.

CDF for Ignition of the remaining electrical cabinets: (1.02E-3)(1.3E-5)= 1.32E-8/yr

Ignition of Remaining Pumps Scenario 12:

Simplified fire modeling for the remaining pumps (3) showed there would be no damage to any other SSD circuits. Therefore, the base model CCDF is applied to the remaining ignition sources. The ignition frequency is (3/5)(2.3E-5) = 1.38E-3/yr.

CDF for Ignition of the remaining pumps: (1.38E-3)(1.3E-5)= 1.79E-8/yr

Ignition of Fire Protection Panels Scenario 13:

Simplified fire modeling for the fire protection panels (4) showed there would be no damage to any other SSD circuits. Therefore, the base model CCDF is applied to the remaining ignition sources. The ignition frequency for the fire protection panels is 4.9E-5/yr.

CDF for Ignition of the fire protection panels: (4.9E-5)(1.3E-5)= 6.37E-10/yr

Ignition of Remaining HVAC Scenario 14:

Simplified fire modeling for the remaining HVAC (4) showed there would be no damage to any other SSD circuits. Therefore, the base model CCDF is applied to the remaining ignition sources. The ignition frequency for the remaining HVAC is (4/6)(1.7E-4) = 1.13E-4/yr.

CDF for Ignition of the remaining HVAC: (1.13E-4)(1.3E-5)= 1.47E-9/yr

Ignition of Transient Combustible Scenario 15:

The CCDF for the worst case scenario was applied to the transient combustibles. Model MS003B calculated a CCDF of 3.9E-3.

The frequency for transient combustibles is 1.17E-3/yr.

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. As calculated earlier in this report the P_{tc} is defined as .00626. Therefore, utilizing P_{tc}, the CDF for the Transient Combustible Scenario is: (1.17E-3)(.00626)(3.9E-3) = 2.86E-8/yr

The CDF values for the above scenarios are added to obtain the total CDF for the area:

SCENARIO	**CDF**
Scenario 1	4.74E-8/yr
Scenario 2	5.33E-8/yr
Scenario 3	1.43E-7/yr
Scenario 4	5.51E-8/yr
Scenario 5	4.88E-7/yr
Scenario 6	1.41E-8/yr
Scenario 7	6.6E-9/yr
Scenario 8	6.6E-9/yr
Scenario 9	7.1E-9/yr
Scenario 10	1.2E-7/yr
Scenario 11	1.32E-8/yr
Scenario 12	1.79E-8/yr
Scenario 13	6.37E-10/yr
Scenario 14	1.47E-9/yr
Scenario 15	2.86E-8/yr
TOTAL	1.00E-6/yr

The results of this analysis indicate that the fire area 06RB is fire risk significant. The total CDF for this fire area is 1.00E-6/yr, preventing the fire area from screening.

The scenario contributing the highest individual CDF (4.88E-7/yr) in the fire area is Scenario 5 (ignition of R1600S003J, H2100P627 & R1600S003D). A fire in any of these electrical cabinets is assumed to result in the loss of the entire cabinet and therefore the equipment fed from this cabinet is also lost. Loss of these cabinets will result in failure of the Division 1 and 2 RHR Low Pressure Coolant Injection (LPCI) isolation valves, the Reactor Recirculation (RR) Recirculation Pump "A" & "B" Discharge Valves and the RHR Crosstie Isolation Valve.

The scenario contributing the second highest individual CDF (1.43E-7/yr) in the fire area is Scenario 3 (ignition of R1600S003B, H2100P625 & R1600S003H). A fire in any of these electrical cabinets is assumed to result in the loss of the entire cabinet and therefore the equipment fed from this cabinet is also lost. Loss of these cabinets will result in failure of Division 1 RHR Suction Cooling Inboard Isolation Valve, HPCI Steam Supply Valve, EECW Supply and Return Isolation Valves, Core Spray Inboard Isolation Valve, and Drywell Cooling Fan #2 among other components.

The scenario contributing the third highest individual CDF (1.2E-7/yr) in the fire area is Scenario 10 (ignition of R1600S005C). A fire in this cabinet is assumed to result in the loss of the entire cabinet and therefore the equipment fed from this cabinet is also lost. Loss of this cabinet will result in failure of Division 2 RHR Inboard Suction Isolation Valve, EECW Drywell Outboard Return Valve, and RHR Outboard Containment Isolation Valve among other components.

The conservatism included in the analysis of these cabinets resulted in the compartment remaining above the screening criteria of 1.0E-6/yr. A further refinement to the analysis would most likely have resulted in screening out this compartment. This further analysis was not performed because the result was judged acceptable and no modifications, or compensatory measures are required.

4A.3.4.2.6 Phase III Analysis for Fire Area 07RB

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the Reactor Building Third Floor, ignition sources are identified which adversely impact Safe Shutdown circuits.

Ignition of T5101S003 Scenario 1:

Simplified fire modeling has shown that ignition of T5101S003 would damage circuits supporting components H2100P082, H2100P034 & H2100P016. PSA Model RB7S01 calculated the CCDF for this damage set as 2.2E-5. Considering that the ignition frequency of this transformer is 9.8E-5/yr, the following can be calculated:

CDF for ignition of T5101S003: (9.8E-5)(2.2E-5)= 2.2E-9/yr

Ignition of G4102C001A or G4102C001B Scenario 2:

Simplified fire modeling has shown that ignition of either G4102C001 A or B would damage circuits supporting components E1150F023, G4102C001A & G4102C001B. PSA Model RB7S02 calculated the CCDF for this damage set as 4.0E-5. Considering that the ignition frequency of the pumps is 9.3E-4/yr the following can be calculated:

CDF for Ignition of G4102C001A or G4102CC : (9.3E-4)(4.0E-5) = 3.72E-8/yr

Ignition of T4804Z001 or T4804Z002 Scenario 3:

Simplified fire modeling has shown that ignition of either T4804Z001 or T4804Z002 would damage circuits for only the Hydrogen Recombiners themselves and would not impact any other circuits/equipment located in the immediate area. Therefore, other equipment normally failed in this PSA model is credited in this scenario (based on no damage to any other circuits). PSA Model PSA95 calculated the CCDF for this damage set as 8.0E-6. Considering that the ignition frequency of the Recombiners is 8.6E-2/yr the following can be calculated:

CDF for Ignition of T4804Z001 or T4804Z002: (8.6E-2)(8.0E-6) = 6.88E-7/yr

Ignition of Electrical Cabinets Scenario 4:

Simplified fire modeling for the electrical cabinets showed there would be no damage to any other SSD circuits. Therefore, the base model CCDF (1.3E-5) is applied to these ignition sources. The ignition frequency for the electrical cabinets is 3.7E-3/yr.

CDF for Ignition of remaining electrical cabinets: (3.7E-3)(1.3E-5) = 4.81E-8/yr

Ignition of HVAC Scenario 5:

Simplified fire modeling for the HVAC units (3) showed there would be no damage to any SSD circuits. Therefore, the base model CCDF (1.3E-5) is applied to these ignition sources.

CDF for Ignition of the HVAC: (8.6E-5)(1.3E-5) = 1.12E-9/yr

Ignition of Transient Combustible Scenario 6:

The CCDF calculated for loss of all components and circuits in RB07 was applied to the transient combustibles. Model RB07 calculated the CCDF as 5.3E-3.

The ignition frequency for transient combustibles is 1.17E-3/yr.

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. As calculated earlier in this report the P_{tc} is defined as .00626. Therefore, utilizing P_{tc}, the CDF for ignition of Transient Combustible is: (1.17E-3)(.00626)(5.3E-3) = 3.88E-8/yr

The individual CDFs for the above scenarios are added to obtain the total CDF for the area:

SCENARIO	**CDF**
Scenario 1	2.2E-9/yr
Scenario 2	3.72E-8/yr
Scenario 3	6.88E-7/yr
Scenario 4	4.81E-8/yr
Scenario 5	1.12E-9/yr
Scenario 6	3.88E-8/yr
TOTAL	8.15E-7/yr

The results of this analysis indicate that the Fire Area 07RB is not fire risk significant. The total CDF for this fire area is 8.15E-7/yr, allowing the Fire Area to screen. No recommendations for further action are necessary.

4A.3.4.2.7 Phase III Analysis for Fire Area 09RB

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the Reactor Building Fifth Floor, ignition sources are identified which adversely impact Safe Shutdown circuits.

Fire Area 09RB consists of two distinct areas. One is located on elevation 684'-6" of the Reactor Bldg. The other area is a chase which extends from elevation 643'-6" up to 684'-6" in the southwest corner of the Auxiliary Building. All of the safe shutdown equipment in 09RB is located in the chase. There are no fixed ignition sources located in the chase.

The frequencies of the fixed ignition sources are:

Electrical Cabinets	6.2E-04/yr
Fire Protection Panels	2.4E-05/yr
Ventilation Subsystems	5.7E-05/yr
Elevator Motors	1.6E-03/yr
Total	2.3E-03/yr

The frequency for the transient ignition source is: 1.17E-03/yr

The CCDFs calculated for 09RB are:

Base Model:	1.3E-5
PSA Model RB09 calculated a CCDF of:	5.3E-4

Fixed Combustible Scenario:

The base model CCDF (1.3E-5) and the ignition frequency for fixed ignition sources were applied to the Reactor Building portion of 09RB.

CDF for Fixed Combustible Scenario: (1.3E-5)(2.3E-3) = 3.0E-8/yr

Transient Combustible :

The calculated CCDF for loss of all safe shutdown equipment and the ignition source frequency for all the transients were applied to the chase.

CDF for Transient Combustible Scenario: (5.3E-4)(1.17E-3) = 6.20E-7/yr

The individual CDFs for the above scenarios are added to obtain the total CDF for the area: 3.0E-8 + 6.20E-7 = 6.50E-7/yr

The results of this analysis indicate that the Fire Area 09RB is not fire risk significant. The total CDF for this fire area is 6.50E-7/yr, allowing the Fire Area to screen. No recommendations for further action are necessary.

4A.3.4.3 Other Zones

4A.3.4.3.1 Phase III Analysis of Fire Area EF1

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of spatial separation within EF1, ignition sources are identified which adversely impact Safe Shutdown circuits.

The Safe Shutdown circuits routed through EF1 are contained in underground ducts that extend from EF1 to the Fermi 2 Auxiliary and Turbine Building. The Ducts are accessible by several manhole structures.

The CCDF for loss of the Safe Shutdown circuits routed through EF1 calculated by PSA Model EF1 to be 2.5E-3.

The ignition sources for this area and the frequencies are:

Yard Transformers (Other):	6.2E-5/yr
Air Compressors:	5.2E-4/yr
Transients:	1.17E-3/yr

The CCDF for loss of the Safe Shutdown circuits routed through EF1 is applied to 'Yard Transformers (Other)' ignition source. The CDF for this scenario is (6.2E-5)(2.5E-3) = 1.55E-7/yr.

The ignition source 'Air Compressor' is not considered to damage any Safe Shutdown circuits. The base model CCDF is applied to the 'Air Compressor' ignition source. The CDF for this scenario is (5.2E-4)(1.3E-5) = 6.8E-9/yr

The CCDF for loss of the Safe Shutdown circuits routed through EF1 is applied to the 'Transients' ignition source. The probability of fire exposure from transient combustibles (P_{tc}) is calculated to be .00626. The CDF for the transient scenario is (1.17E-3)(0.00626)(2.5E-3) = 1.83E-8/yr.

The total CDF for EF1 is: 1.55E-7 + 6.8E-9 + 1.83E-8 = 1.80E-7/yr

The results of this analysis indicate that the Fire Area EF1 is not fire risk significant. The total CDF for this fire area is 1.80E-7/yr, allowing the Fire Area to screen. No recommendations for further action are necessary.

4A.3.4.3.2 Phase III Analysis for The RHR1 Complex

The approach utilized in the Phase III analysis of fire area RHR1 is to sub-compartmentalize the fire area. This approach is available based on the existence of internal barriers within the RHR Building. The two Switchgear Rooms on elevation 617'-00" are combined to form a separate fire area from the remaining portion of fire area RHR1. The new fire area for the combined Switchgear Rooms is designated as fire area RHR1SG.

4A.3.4.3.2.1 Phase III Analysis for Fire Area RHR1

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed and transient combustibles in fire area RHR1, ignition sources are identified which adversely impact Safe Shutdown circuits. These ignition sources are listed below with the Safe Shutdown components affected given ignition.

Ignition of Electrical Cabinet H2100P517 Scenario 18

Simplified fire modeling has shown that ignition of H2100P517 damaged circuits for safe shutdown component X4103N058A. The CCDF calculated for this damage set by PSA Model R1S01 is 1.3E-5. The ignition source frequency is (1/159)(2.4E-3) = 1.5E-5/yr.

The CDF for this scenario is: (1.5E-5)(1.3E-5) = 2E-10/yr

Ignition of Electrical Cabinet R1600S046 Scenario 27

Simplified fire modeling has shown that ignition of R1600S046 damaged circuits for safe shutdown component R3000S001. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (7/159)(2.4E-3) = 1.1E-4/yr.

The CDF for this scenario is: (1.1E-4)(1.3E-5) = 1.4E-9/yr

Ignition of Electrical Cabinet R1600S047 Scenario 28

Simplified fire modeling has shown that ignition of R1600S047 damaged circuits for safe shutdown component R3000S002. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (7/159)(2.4E-3) = 1.1E-4/yr.

The CDF for this scenario is: (1.1E-4)(1.3E-5) = 1.4E-9/yr

Ignition of Electrical Cabinet R1700S015A Scenario 44

Simplified fire modeling has shown that ignition of R1700S015A damaged circuits for safe shutdown component X4103F122. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/159)(2.4E-3) = 1.5E-5/yr.

The CDF for this scenario is: (1.5E-5)(1.3E-5) = 2E-10/yr

Ignition of Remaining Electrical Cabinets

The remaining 27 electrical cabinets consist of instrument racks, lighting panels, remote lamp panels and battery operated emergency lights. None of the remaining electrical cabinets are safe shutdown components. The base model CCDF was applied to these components. The ignition source frequency is (27/159)(2.4E-3) = 4.1E-4/yr.

The CDF for this scenario is: (4.1E-4)(1.3E-5) = 5.3E-9/yr

Ignition of Diesel Generator R3000S001 Scenario 50

Simplified fire modeling of a one pint oil spill has shown that ignition of R3000S001 damaged circuits for safe shutdown components R3000P312, and R1600S046. Fire modeling of 1 pint of oil was performed for the diesel generator. This quantity typifies the ignition of oily rags or oil leaking onto hot diesel generator components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/2)(2.6E-2) = 1.30E-2/yr.

The CDF for this scenario is: (1.30E-2)(1.3E-5) = 1.69E-7/yr

Ignition of Diesel Generator R3000S002 Scenario 51

Simplified fire modeling of a one pint oil spill has shown that ignition of R3000S002 damaged circuits for safe shutdown components R3000P322, and R1600S047. Fire modeling of 1 pint of oil was performed for the diesel generator. This quantity typifies the ignition of oily rags or oil leaking onto hot diesel generator components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/2)(2.6E-2) = 1.30E-2/yr.

The CDF for this scenario is: (1.30E-2)(1.3E-5) = 1.69E-7/yr

Ignition of Pump E1151C001A Scenario 12

Per plant personnel, a rupture of the oil reservoir on this pump is not credible. The model used (a one pint oil spill) approximated the accumulation of oil from a fuel sample line leak. The one pint oil spill is conservative since plant experience indicates that this type of leak involved only drops per day. Plant inspections occur at least once every 2 weeks. Therefore, the accumulated quantity of oil would not exceed 1 pint and, therefore, 1 pint is conservative. This simplified fire modeling has shown that ignition of E1151C001A damaged circuits for safe shutdown components E1151C001C and E1150F604A. The CCDF calculated for this damage set was enveloped by PSA Model R1S05, CCDF = 2.5E-4. The ignition source frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

The CDF for this scenario is: (4.6E-4)(2.5E-4) = 1.15E-7/yr

Ignition of Pump E1151C001C Scenario 13

Per plant personnel, a rupture of the oil reservoir on this pump is not credible. The model used (a one pint oil spill) approximated the accumulation of oil from a fuel sample line leak. The one pint oil spill is conservative since plant experience indicates that this type of leak involved only drops per day. Plant inspections occur at least once every 2 weeks. Therefore, the accumulated quantity of oil would not exceed 1 pint and, therefore, 1 pint is conservative. This simplified fire modeling has shown that ignition of E1151C001C damaged circuits for safe shutdown component E1151C001A. The CCDF calculated for this damage set was enveloped by PSA Model R1S05, CCDF = 2.5E-4. The ignition source frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

The CDF for this scenario is: (4.6E-4)(2.5E-4) = 1.15E-7/yr

Ignition of Pump P4500C002A Scenario 41

Simplified fire modeling has shown that ignition of P4500C002A does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S07, CCDF = 1.3E-5. The ignition source frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

The CDF for this scenario is: (4.6E-4)(1.3E-5) = 6.0E-9/yr

Ignition of Pump R3001C005 Scenario 42

Simplified fire modeling has shown that ignition of R3001C005 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

The CDF for this scenario is: (4.6E-4)(1.3E-5) = 6.0E-9/yr

Ignition of Pump R3001C006 Scenario 43

Simplified fire modeling has shown that ignition of R3001C006 does not result in damage to circuits for other safe shutdown components The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

The CDF for this scenario is: (4.6E-4)(1.3E-5) = 6.0E-9/yr

Ignition of Pump X4103C025 Scenario 54

Simplified fire modeling has shown that ignition of X4103C25 does not result in damage to circuits for safe shutdown components. This pump is not a safe shutdown component. The base model CCDF is applied to this component. The ignition source frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

The CDF for this scenario is: (4.6E-4)(1.3E-5) = 6.0E-9/yr

Ignition of Pumps R3000C002 and R300C004 Scenario 55

The fuel oil transfer pumps are located in the Fuel Oil Tank Room. Each pump is powered by a one horsepower motor. Ignition of either pump is considered to disable the other pump. The CCDF for loss of both fuel oil transfer pumps is conservatively enveloped by PSA Model R1S01 which represents loss of Diesel Generator R3000S002. The CCDF calculated is 1.3E-5. The ignition source frequency for the two pumps is (2/10)(4.6E-3) = 9.2E-4/yr.

The CDF for this scenario is: (9.2E-4)(1.3E-5) = 1.2E-8/yr

Ignition of Pumps R3000C001 and R300C003 Scenario 56

The fuel oil transfer pumps are located in the Fuel Oil Tank Room. Each pump is powered by a one horsepower motor. Ignition of either pump is considered to disable the other pump. The CCDF for loss of both fuel oil transfer pumps is conservatively enveloped by PSA Model R1S01 which represents loss of Diesel Generator R3000S001. The CCDF calculated is 1.3E-5. The ignition source frequency for the two pumps is (2/10)(4.6E-3) = 9.2E-4/yr.

The CDF for this scenario is: (9.2E-4)(1.3E-5) = 1.2E-8/yr

Ignition of Fire Protection Panels

The existing fire protection panels are not safe shutdown components, and will not affect safe shutdown circuits. The base model CCDF of 1.3E-5 is applied to these components. The ignition source frequency is 1.10E-4/yr.

The CDF for this scenario is: (1.10E-4)(1.3E-5) = 1.43E-9/yr.

Ignition of Heater X4103B213 Scenario 14

Simplified fire modeling has shown that ignition of X4103B213 damaged circuits for safe shutdown component X4103F104. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Heater X4103B234 Scenario 15

Simplified fire modeling has shown that ignition of X4103B234 damaged circuits for safe shutdown components E1150F603A and R3001C005. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Remaining Heaters Scenario

Simplified fire modeling has shown that ignition of the remaining 9 heaters does not result in damage to circuits for other safe shutdown components. These heaters are not safe shutdown components. The base model CCDF is applied to these components. The ignition source frequency is (9/27)(7.7E-4) = 2.57E-4/yr.

The CDF for this scenario is: (2.57E-4)(1.3E-5) = 3.3E-9/yr

Ignition of Fan E1156C001A Scenario 25

Simplified fire modeling has shown that ignition of E1156C001A results in damage to circuits for safe shutdown component E1156N130A. The CCDF calculated for this damage set was enveloped by the base model, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan E1156C001C Scenario 26

Simplified fire modeling has shown that ignition of E1156C001C results in damage to circuits for safe shutdown component E1156N130C The CCDF calculated for this damage set was enveloped by the base model, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C001 Scenario 29

Simplified fire modeling has shown that ignition of X4103C001 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C002 Scenario 30

Simplified fire modeling has shown that ignition of X4103C002 does not result in damage to circuits for other safe shutdown components The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C003 Scenario 31

Simplified fire modeling has shown that ignition of X4103C003 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C004 Scenario 32

Simplified fire modeling has shown that ignition of X4103C004 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C009 Scenario 33

Simplified fire modeling has shown that ignition of X4103C0009 does not result in damage to circuits for other safe shutdown components The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C010 Scenario 34

Simplified fire modeling has shown that ignition of X4103C010 does not result in damage to circuits for other safe shutdown components The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C011 Scenario 35

Simplified fire modeling has shown that ignition of X4103C011 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C012 Scenario 36

Simplified fire modeling has shown that ignition of X4103C012 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C017 Scenario 37

Simplified fire modeling has shown that ignition of X4103C017 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for the PSA base model, CCDF = 1.3E-5, is applied to this scenario. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C018 Scenario 38

Simplified fire modeling has shown that ignition of X4103C018 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for the PSA base model, CCDF = 1.3E-5, is applied to this scenario. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C021 Scenario 39

Simplified fire modeling has shown that ignition of X4103C021 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Fan X4103C022 Scenario 40

Simplified fire modeling has shown that ignition of X4103C022 does not result in damage to circuits for other safe shutdown components The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/27)(7.7E-4) = 2.85E-5/yr.

The CDF for this scenario is: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Transient Combustible Scenario

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. As calculated earlier, P_{tc} is 0.00626. For conservatism the whole area CCDF (calculated by PSA Model RHR1S1 as 4.3E-3) is applied to the transient combustibles. The ignition source frequency is (1.2E-4 + 7.2E-4 + 3.3E-4) = 1.17E-3/yr.

The CDF for this scenario is: (1.17E-3)(4.3E-3)(0.00626) = 3.15E-8/yr

The CDFs for the above scenarios are added to obtain the Total CDF for fire area RHR1:

Electrical Cabinet Scenario 18	2E-10/yr
Electrical Cabinet Scenario 27	1.4E-9/yr
Electrical Cabinet Scenario 28	1.4E-9/yr
Electrical Cabinet Scenario 44	2E-10/yr
Remaining Electrical Cabinets	5.3E-9/yr
Diesel Generator Scenario 50	1.69E-7/yr
Diesel Generator Scenario 51	1.69E-7/yr
Pump Scenario 12	1.15E-7/yr
Pump Scenario 13	1.15E-7/yr
Pump Scenario 41	6.0E-9/yr

Pump Scenario 42	6.0E-9/yr
Pump Scenario 43	6.0E-9/yr
Pump Scenario 54	6.0E-9/yr
Pump Scenario 55	1.2E-8/yr
Pump Scenario 56	1.2E-8/yr
Fire Protection Panels	1.43E-9/yr
Heater Scenario 14	3.71E-10/yr
Heater Scenario 15	3.71E-10/yr
Remaining Heaters	3.3E-9/yr
Fan Scenario 25	3.71E-10/yr
Fan Scenario 26	3.71E-10/yr
Fan Scenario 29	3.71E-10/yr
Fan Scenario 30	3.71E-10/yr
Fan Scenario 31	3.71E-10/yr
Fan Scenario 32	3.71E-10/yr
Fan Scenario 33	3.71E-10/yr
Fan Scenario 34	3.71E-10/yr
Fan Scenario 35	3.71E-10/yr
Fan Scenario 36	3.71E-10/yr
Fan Scenario 37	3.71E-10/yr
Fan Scenario 38	3.71E-10/yr
Fan Scenario 39	3.71E-10/yr
Fan Scenario 40	3.71E-10/yr
Transient Combustible	3.15E-8/yr
TOTAL	6.6E-7/yr

The results of this analysis indicate that the Fire Area RHR1 is not fire risk significant. The CDF for this scenario for this fire area is 6.6E-7/yr , allowing the Fire Area to screen.

4A.3.4.3.2.2 Phase III Analysis for Fire Area RHR1SG

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the RHR Building (Division 1), ignition sources are identified which adversely impact Safe Shutdown circuits.

The walls surrounding fire area RHR1SG are constructed of a minimal twelve (12) inch concrete wall. These walls are sufficient to withstand the hazards associated with the areas.

The penetrations within the RHR Building are not controlled and maintained as part of the Fermi 2 Penetration Seal Surveillance Program. During a walkdown of the RHR Building, observations of randomly selected electrical and mechanical penetrations were performed. These observations indicated that the electrical and mechanical penetrations were adequately sealed against the propagation of flame and products of combustion. Therefore, the electrical and mechanical penetrations through these Switchgear Room barriers are also considered to be adequately sealed against the propagation of flame and products of combustion.

Ignition of Electrical Cabinet R3000S006 Scenario 3

Simplified fire modeling has shown that ignition of R3000S006 damages circuits for safe shutdown components H1100P869, X4103F151, X4103F152, X4103N056C and X4103N057C, and the panel itself. The CCDF calculated for this damage set enveloped by PSA Model R1S01 is 1.3E-5. The ignition source frequency is (1/159)(2.4E-3) = 1.5E-5/yr.

The CDF for this scenario is: (1.5E-5)(1.3E-5) = 2E-10/yr.

Ignition of Electrical Cabinet R1400S036 Scenario 4

Simplified fire modeling has shown that ignition of R1400S036 damaged circuits for safe shutdown components X4103F149, X4103F150 and X4100N057A. The CCDF calculated for this damage set was enveloped by PSA Model R1S02, CCDF = 3.3E-4. The ignition source frequency is (7/159)(2.4E-3) = 1.1E-4/yr.

The CDF for this scenario is: (1.1E-4)(3.3E-4) = 3.63E-8/yr

Ignition of Electrical Cabinet R1400S002B Scenario 6

Simplified fire modeling has shown that ignition of R1400S002B damaged circuits for safe shutdown components X4103F115, X4103F116, and X4100N056C. The CCDF calculated for this damage set was enveloped by PSA Model R1S03, CCDF = 6.1E-4. The ignition source frequency is (4/159)(2.4E-3) = 6.0E-5/yr.

The CDF for this scenario is: (6.0E-5)(6.1E-4) = 3.66E-8/yr

Ignition of Electrical Cabinet R1400S002A Scenario 7

Simplified fire modeling has shown that ignition of R1400S002A damaged circuits for safe shutdown components X4100N056A, X4103F104, H1100P869 and X4103F103. The CCDF calculated for this damage set was enveloped by PSA Model R1S02, CCDF = 3.3E-4. The ignition source frequency is (4/159)(2.4E-3) = 6.0E-5/yr.

The CDF for this scenario is: (6.0E-5)(3.3E-4) = 1.98E-8/yr

Ignition of Electrical Cabinet R3000S005 Scenario 22

Simplified fire modeling has shown that ignition of R3000S005 damages circuits for safe shutdown component H1100P869 and the panel itself. The CCDF calculated for this damage set enveloped by PSA Model R1S01 is 1.3E-5. The ignition source frequency is (1/159)(2.4E-3) = 1.5E-5/yr.

The CDF for this scenario is: (1.5E-5)(1.3E-5) = 2E-10/yr.

Ignition of Electrical Cabinet R1600S016A Scenario 46

Simplified fire modeling has shown that ignition of R1600S016A does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S08, CCDF = 3.3E-4. The ignition source frequency is (23/159)(2.4E-3) = 3.5E-4/yr.

The CDF for this scenario is: (3.5E-4)(3.3E-4) = 1.15E-7/yr

Ignition of Electrical Cabinet R1600S016B Scenario 47

Simplified fire modeling has shown that ignition of R1600S016B does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S08, CCDF = 3.3E-4. The ignition source frequency is (23/159)(2.4E-3) = 3.5E-4/yr.

The CDF for this scenario is: (3.5E-4)(3.3E-4) = 1.15E-7/yr

Ignition of Electrical Cabinet R1600S017A Scenario 48

Simplified fire modeling has shown that ignition of R1600S017A does not result in damage to circuits for other safe shutdown components The CCDF calculated for this damage set was enveloped by PSA Model 3AB16, CCDF = 1.3E-4. The ignition source frequency is (24/159)(2.4E-3) = 3.6E-4/yr.

The CDF for this scenario is: (3.6E-4)(1.3E-4) = 4.68E-8/yr

Ignition of Electrical Cabinet R3200S063 Scenario 49

Simplified fire modeling has shown that ignition of R3200S063 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model 10AB1, CCDF = 4.0E-4. The ignition source frequency is (21/159)(2.4E-3) = 3.2E-4/yr.

The CDF for this scenario is:(3.2E-4)(4.0E-4) = 1.28E-7/yr

Ignition of Electrical Cabinet H2100P351 Scenario 52

Simplified fire modeling has shown that ignition of H2100P351 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/159)(2.4E-3) = 1.5E-5/yr.

The CDF for this scenario is: (1.5E-5)(1.3E-5) = 2E-10/yr

Ignition of Electrical Cabinet H2100P350 Scenario 53

Simplified fire modeling has shown that ignition of H2100P350 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set was enveloped by PSA Model R1S01, CCDF = 1.3E-5. The ignition source frequency is (1/159)(2.4E-3) = 1.5E-5/yr.

The CDF for this scenario is: (1.5E-5)(1.3E-5) = 2E-10/yr

Ignition of Electrical Cabinet R1400S037 Scenario 5

Simplified fire modeling has shown that ignition of R1400S037 damaged circuits for safe shutdown components X4103F151, X4103F152 and X4100N057C. The CCDF calculated for this damage set was enveloped by PSA Model R1S03, CCDF = 6.1E-4. The ignition source frequency is (6/159)(2.4E-3) = 9.1E-5/yr.

The CDF for this scenario is: (9.1E-5)(6.1E-4) = 5.55E-8/yr

Ignition of Heaters X4103B242 and X4103B243 Scenario 57

Simplified fire modeling has shown that ignition of the heaters X4103B242 and X4103B243 do not result in damage to circuits for other safe shutdown components. These heaters are not safe shutdown components. The base model CCDF is applied to these components. The ignition source frequency is (2/27)(7.7E-4) = 5.7E-5/yr.

The CDF for this scenario is: (5.7E-5)(1.3E-5) = 7.42E-10/yr

Ignition of Transformers R1700S013A, R1600S123A and R1600S125A Scenario 58

Simplified fire modeling has shown that ignition of R1700S013A, R1600S123A and R1600S125A does not result in damage to circuits for other safe shutdown components. These transformers are not safe shutdown components. The base model CCDF is applied to these components. The ignition source frequency is (3/5)(4.9E-4) = 2.94E-4/yr.

The CDF for this scenario is: (2.94E-4)(1.3E-5) = 3.82E-9/yr

Ignition of Transient Combustible Scenario

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. As calculated earlier, P_{tc} is 0.00626. For conservatism the whole area CCDF (calculated by PSA Model RHR1S1 as 4.3E-3) is applied to the transient combustibles. The ignition source frequency is (1.2E-4 + 7.2E-4 + 3.3E-4) = 1.17E-3/yr.

The CDF for this scenario is: (1.17E-3)(4.3E-3)(0.00626) = 3.15E-8/yr

The CDF for the above scenarios are added to obtain the Total CDF for fire area RHR1SG:

Electrical Cabinet Scenario 3	2E-10/yr
Electrical Cabinet Scenario 4	3.63E-8/yr
Electrical Cabinet Scenario 6	3.66E-8/yr
Electrical Cabinet Scenario 7	1.98E-8/yr
Electrical Cabinet Scenario 22	2E-10/yr
Electrical Cabinet Scenario 46	1.15E-7/yr
Electrical Cabinet Scenario 47	1.15E-7/yr
Electrical Cabinet Scenario 48	4.68E-8/yr
Electrical Cabinet Scenario 49	1.28E-7/yr
Electrical Cabinet Scenario 52	2E-10/yr
Electrical Cabinet Scenario 53	2E-10/yr
Electrical Cabinet Scenario 5	5.55E-8/yr
Transformer Scenario 58	3.82E-9/yr

Heaters Scenario 57	7.42E-10/yr
Transient Combustible	3.15E-8/yr
TOTAL	5.9E-7/yr

The results of this analysis indicate that the Fire Area RHR1SG is not fire risk significant. The CDF for this fire area is 5.9E-7/yr, allowing the Fire Area to screen.

4A.3.4.3.3 Phase III Analysis Of The RHR2 Complex

The approach utilized in the Phase III analysis of fire area RHR2 is to sub-compartmentalize the fire area. This approach is available based on the existence of internal barriers within the RHR Building. The two Switchgear Rooms on elevation 617'-00" are combined to form a separate fire area from the remaining portion of fire area RHR2. The new fire area for the combined Switchgear Rooms is designated as fire area RHR2SG.

4A.3.4.3.3.1 Phase III Analysis for Fire Area RHR2

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed and transient combustibles in fire area RHR2, ignition sources are identified which adversely impact Safe Shutdown circuits. These ignition sources are listed below with the Safe Shutdown components affected given ignition.

Ignition of Electrical Cabinet R1600S048 Scenario 26

Simplified fire modeling has shown that ignition of R1600S048 does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set by PSA Model R2S03 is 1.3E-5. The ignition frequency is (7/151)(2.4E-3) = 1.11E-4/yr.

CDF for this scenario is (1.11E-4)(1.3E-5) = 1.44E-9/yr.

Ignition of Electrical Cabinet R1600S049 Scenario 27

Simplified fire modeling has shown that ignition of R1600S049 does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set by PSA Model R2S03 is 1.3E-5. The ignition frequency is (7/151)(2.4E-3) = 1.11E-4/yr.

CDF for this scenario is (1.11E-4)(1.3E-5) = 1.44E-9/yr.

Ignition of Electrical Cabinet H2100P518 Scenario 29

Simplified fire modeling has shown that ignition of H2100P518 damages circuits for safe shutdown components X4103F164, X4103F166, X4103F168, X4103F169 and X4103N058B. The CCDF calculated for this damage set is enveloped by the base model CCDF of 1.3E-5. The ignition frequency is (1/151)(2.4E-3) = 1.58E-5/yr.

CDF for this scenario is (1.58E-5)(1.3E-5) = 2.05E-10/yr.

Ignition of Remaining Electrical Cabinets

The remaining 25 electrical cabinets consist of instrument racks, lighting panels, remote lamp panels and battery operated emergency lights. None of the remaining electrical cabinets are safe shutdown components. The base model CCDF of 1.3E-5 is applied to these components. The ignition frequency is (25/151)(2.40E-3) = 4.0E-4/yr.

CDF for this scenario is (4.0E-4)(1.3E-5) = 5.2E-9/yr.

Ignition of Diesel Generator R3000S003 Scenario 24

Simplified fire modeling of a one pint oil spill has shown that ignition of R3000S003 damaged circuits for safe shutdown component R1600S048 and the diesel itself. Fire modeling of 1 pint of oil was performed for the diesel generator. This quantity typifies the ignition of oily rags or oil leaking onto the hot diesel generator components. The CCDF calculated for this damage set enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (1/2)(2.6E-2) = 1.30E-2/yr.

CDF for this scenario is (1.30E-2)(1.3E-5) = 1.69E-7/yr.

Ignition of Diesel Generator R3000S004 Scenario 25

Simplified fire modeling of a one pint oil spill has shown that ignition of R3000S004 damaged circuits for safe shutdown component R1600S049 and the diesel itself. Fire modeling of 1 pint of oil was performed for the diesel generator. This quantity typifies the ignition of oily rags or oil leaking onto hot diesel generator components. The CCDF calculated for this damage set enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (1/2)(2.6E-2) = 1.30E-2/yr.

CDF for this scenario is (1.30E-2)(1.3E-5) = 1.69E-7/yr.

Ignition of Pump E1151C001B Scenario 18

Per plant personnel, a rupture of the oil reservoir on this pump is not credible. The model used (a one pint oil spill) approximated the accumulation of oil from a fuel sample line leak. The one pint oil spill is conservative since plant experience indicates that this type of leak involved only drops per day. Plant inspections occur at least once every 2 weeks. Therefore, the accumulated quantity of oil would not exceed 1 pint and, therefore, 1 pint is conservative. This simplified fire modeling has shown that ignition of E1151C001B does damage circuits for E1151C001D. The CCDF calculated for this damage set enveloped by PSA Model R2S04 is 2.8E-4. The ignition frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

CDF for this scenario is (4.6E-4)(2.8E-4) = 1.29E-7/yr.

Ignition of Pump E1151C001D Scenario 19

Per plant personnel, a rupture of the oil reservoir on this pump is not credible. The model used (a one pint oil spill) approximated the accumulation of oil from a fuel sample line leak. The one pint oil spill is conservative since plant experience indicates that this type of leak involved only drops per day. Plant inspections occur at least once every 2 weeks. Therefore, the accumulated quantity of oil would not exceed 1 pint and, therefore, 1 pint is conservative. This simplified fire modeling has shown that ignition of E1151C001D does damage circuits for E1151C001B. The CCDF calculated for this damage set enveloped by PSA Model R2S04 is 2.8E-4. The ignition frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

CDF for this scenario is (4.6E-4)(2.8E-4) = 1.29E-7/yr.

Ignition of Pump R3001C007 Scenario 22

Simplified fire modeling has shown that ignition of R3001C007 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

CDF for this scenario is (4.6E-4)(1.5E-5) = 6.9E-9/yr.

Ignition of Pump R3001C008 Scenario 23

Simplified fire modeling has shown that ignition of R3001C008 does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

CDF for this scenario is (4.6E-4)(1.5E-5) = 6.9E-9/yr.

Ignition of Pump P4500C002B Scenario 28

Simplified fire modeling has shown that ignition of P4500C002B does not result in damage to circuits for other safe shutdown components. The CCDF calculated for this damage set by PSA Model RB05S2 is 9.0E-5. The ignition frequency is (1/10)(4.6E-3) = 4.6E-4/yr.

CDF for this scenario is (4.6E-4)(9.0E-5) = 4.14E-8/yr.

Ignition of Pumps R3000C010 and R300C012 Scenario 41

The fuel oil transfer pumps are located in the Fuel Oil Tank Room. Each pump is powered by a one horsepower motor. Ignition of either pump is considered to disable the other pump. The CCDF for loss of both fuel oil transfer pumps is conservatively enveloped by PSA Model R2S03 which represents loss of Diesel Generator R3000S004. The CCDF calculated is 1.3E-5. The ignition source frequency for the two pumps is (2/10)(4.6E-3) = 9.2E-4/yr.

The CDF for this scenario is: (9.2E-4)(1.3E-5) = 1.2E-8/yr

Ignition of Pumps R3000C009 and R300C011 Scenario 42

The fuel oil transfer pumps are located in the Fuel Oil Tank Room. Each pump is powered by a one horsepower motor. Ignition of either pump is considered to disable the other pump. The CCDF for loss of both fuel oil transfer pumps is conservatively enveloped by PSA Model R2S03 which represents loss of Diesel Generator R3000S003. The CCDF calculated is 1.3E-5. The ignition source frequency for the two pumps is (2/10)(4.6E-3) = 9.2E-4/yr.

The CDF for this scenario is: (9.2E-4)(1.3E-5) = 1.2E-8/yr

Ignition of Remaining Pumps

The remaining 2 pumps consist of X4103C026 and Y5200C002, which are not safe shutdown components. The base model CCDF of 1.3E-5 is applied to these components. The ignition frequency is (2/10)(4.6E-3) = 9.2E-4/yr.

CDF for this scenario is (9.2E-4)(1.3E-5) = 1.2E-9/yr.

Ignition of Fire Protection Panels

The existing fire protection panels are not safe shutdown components. The base model CCDF of 1.3E-5 is applied to these components. The ignition frequency is 9.75E-5/yr.

CDF for this scenario is (9.75E-5)(1.3E-5) = 1.27E-9/yr.

Ignition of Transformer R1600S125B Scenario 43

Simplified fire modeling has shown that ignition of R1600S125B does not result in damage to circuits for other safe shutdown components. This transformer is not safe a shutdown component. The base model CCDF is applied to this component. The ignition frequency is (1/7)(6.8E-4) = 9.7E-5/yr.

CDF for this scenario is (9.7E-5)(1.3E-5) = 1.26E-9/yr.

Ignition of Fan X4103C005, X4103C006, X4103C007 or X4103C008 Scenarios 6-9

Simplified fire modeling has shown that ignition of any of these fans does not result in damage to circuits for other safe shutdown components. The CCDF calculated for each of these damage sets enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (4/40)(1.1E-3) = 1.1E-4/yr.

CDF for this scenario is (1.1E-4)(1.3E-5) =1.43E-9/yr.

Ignition of Fan X4103C013, X4103C014, X4103C015 or X4103C016 Scenarios 10-13

Simplified fire modeling has shown that ignition of any of these fans does not result in damage to circuits for other safe shutdown components. The CCDF calculated for each of these damage sets enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (4/40)(1.1E-3) = 1.1E-4/yr.

CDF for this scenario is (1.1E-4)(1.3E-5) = 1.43E-9/yr.

Ignition of Fan X4103C019 or X4103C020 Scenarios 14, 15

Simplified fire modeling has shown that ignition of either of these fans does not result in damage to circuits for other safe shutdown components. The base model CCDF is applied to these components. The ignition frequency is (2/40)(1.1E-3) = 5.5E-5/yr.

CDF for this scenario is (5.5E-5)(1.3E-5) = 7.1E-10/yr.

Ignition of Fan X4103C023 or X4103C024 Scenarios 16, 17

Simplified fire modeling has shown that ignition of either of these fans does not result in damage to circuits for other safe shutdown components. The CCDF calculated for each of these damage sets enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (2/40)(1.1E-3) = 5.5E-5/yr.

CDF for this scenario is (5.5E-5)(1.3E-5) = 7.1E-10/yr.

Ignition of Fan E1156C001B or E1156C001D Scenarios 20, 21

Simplified fire modeling has shown that ignition of either of these fans does not result in damage to circuits for other safe shutdown components. The CCDF calculated for each of these damage sets enveloped by PSA Model R2S04 is 2.8E-4. The ignition frequency is(2/40)(1.1E-3) = 5.5E-5/yr.

CDF for this scenario is (5.5E-5)(2.8E-4) = 1.54E-8/yr.

Ignition of Heater X4103B214 Scenario 35

Simplified fire modeling has shown that ignition of X4103B214 damages circuits for safe shutdown components X4103F139 and X4103F140. The CCDF calculated for this damage set by PSA Model R2S01 is 2.9E-4. The ignition frequency is (1/40)(1.1E-3) = 2.8E-5/yr.

CDF for this scenario is (2.8E-5)(2.9E-4) = 7.98E-9/yr.

Ignition of Heater X4103B215 Scenario 36

Simplified fire modeling has shown that ignition of X4103B215 damages circuits for safe shutdown components X4103F127 and X4103F128. The CCDF calculated for this damage set by PSA Model R2S02 is 1.5E-5. The ignition frequency is (1/40)(1.1E-3) = 2.8E-5/yr.

CDF for this scenario is (2.8E-5)(1.3E-5) = 3.6E-10/yr.

Ignition of Heater X4103B226 Scenario 37

Simplified fire modeling has shown that ignition of X4103B226 damages circuits for safe shutdown components X4103F146. The CCDF calculated for this damage set is enveloped by the base model. The ignition frequency is (1/40)(1.1E-3) = 2.8E-5/yr.

CDF for this scenario is (2.8E-5)(1.3E-5) = 3.6E-10/yr.

Ignition of Heater X4103B241 Scenario 38

Simplified fire modeling has shown that ignition of X4103B241 damages circuits for safe shutdown components X4103F169. The CCDF calculated for this damage set is enveloped by the base model. The ignition frequency is (1/40)(1.1E-3) = 2.8E-5/yr.

CDF for this scenario is (2.8E-5)(1.3E-5) = 3.6E-10/yr.

Ignition of Remaining Heaters Scenario

Simplified fire modeling has shown that ignition of the 20 remaining heaters does not result in damage to circuits for any safe shutdown components. These heaters are not safe shutdown components. The base model CCDF is applied to these components. The ignition frequency is (20/40)(1.1E-3) = 5.5E-4/yr.

CDF for this scenario is (5.5E-4)(1.3E-5) = 7.15E-9/yr.

Ignition of Transient Combustible Scenario

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. As calculated earlier, P_{tc} is 0.00626. For conservatism the worst case RHR2 CCDF (calculated by PSA Model RHR2 as 4.4E-3) is applied to the transient combustibles. The ignition frequency is (1.2E-4 + 7.2E-4 + 3.3E-4) = 1.17E-3/yr.

CDF for this scenario is (1.17E-3)(4.4E-3)(0.00626) = 3.2E-8/yr.

The CDFs for the above scenarios are added to obtain the total CDF for the fire area:

SCENARIO	CDF
Electrical Cabinet Scenario 26	1.44E-9/yr
Electrical Cabinet Scenario 27	1.44E-9/yr
Electrical Cabinet Scenario 29	2.05E-10/yr
Remaining Electrical Cabinets	5.2E-9/yr
Diesel Generator Scenario 24	1.69E-7/yr
Diesel Generator Scenario 25	1.69E-7/yr
Pump Scenario 18	1.29E-7/yr
Pump Scenario 19	1.29E-7/yr
Pump Scenario 22	6.9E-9/yr
Pump Scenario 23	6.9E-9/yr

Pump Scenario 28	4.14E-8/yr
Pump Scenario 41	1.2E-8/yr
Pump Scenario 42	1.2E-8/yr
Remaining Pumps Scenario	1.2E-9/yr
Fire Protection Panels	1.27E-9/yr
Transformer Scenario 43	1.26E-9/yr
Fan Scenarios 6-9	1.43E-9/yr
Fan Scenarios 10-13	1.43E-9/yr
Fan Scenarios 14, 15	7.1E-10/yr
Fan Scenarios 16, 17	7.1E-10/yr
Fan Scenarios 20, 21	1.54E-8/yr
Heater Scenario 35	7.98E-9/yr
Heater Scenario 36	3.6E-10/yr
Heater Scenario 37	3.6E-10/yr
Heater Scenario 38	3.6E-10/yr
Remaining Heater Scenario	7.15E-9/yr
Transient Combustible	3.2E-8/yr
TOTAL	7.3E-7/yr

The results of this analysis indicate that the fire area RHR2 is not fire risk significant. The total CDF for this fire area is 7.3E-7/yr, allowing the fire area to screen.

4A.3.4.3.3.2 Phase III Analysis for Fire Area RHR2SG

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the RHR Building (Division 2), ignition sources are identified which adversely impact Safe Shutdown circuits.

The walls surrounding fire area RHR2SG are constructed of a minimal twelve (12) inch concrete wall. These walls are sufficient to withstand the hazards associated with the areas.

The penetrations within the RHR Building are not controlled and maintained as part of the Fermi 2 Penetration Seal Surveillance Program. During a walkdown of the RHR Building, observations of randomly selected electrical and mechanical penetrations were performed. These observations indicated that the electrical and mechanical penetrations were adequately sealed against the propagation of flame and products of combustion. Therefore, the electrical and mechanical penetrations through these Switchgear Room barriers are also considered to be adequately sealed against the propagation of flame and products of combustion.

Ignition of Electrical Cabinet H2100P352 Scenario 1

Simplified fire modeling has shown that ignition of H2100P352 does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (1/151)(2.4E-3) = 1.58E-5/yr.

CDF for this scenario is (1.58E-5)(1.3E-5) = 2.05E-10/yr.

Ignition of Electrical Cabinet H2100P353 Scenario 2

Simplified fire modeling has shown that ignition of H2100P353 does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (1/151)(2.4E-3) = 1.58E-5/yr.

CDF for this scenario is (1.58E-5)(1.3E-5) = 2.05E-10/yr.

Ignition of Electrical Cabinet R1400S002D Scenario 3

Simplified fire modeling has shown that ignition of R1400S002D damages circuits for safe shutdown components X4103N056D, X4103F139 and X4103F140, as well as R1400S002D. The CCDF calculated for this damage set by PSA Model R2S01 is 2.9E-4. The ignition frequency is (4/151)(2.4E-3) = 6.36E-5/yr.

CDF for this scenario is (6.36E-5)(2.9E-4) = 1.84E-8/yr.

Ignition of Electrical Cabinet R1400S039 Scenario 4

Simplified fire modeling has shown that ignition of R1400S039 damages circuits for safe shutdown components X4100N057D, X4103F155, X4103F156 and R1400S039. The CCDF calculated for this damage set by PSA Model R2S01 is 2.9E-4. The ignition frequency is (7/151)(2.4E-3) = 1.11E-4/yr.

CDF for this scenario is (1.11E-4)(2.9E-4) = 3.21E-8/yr.

Ignition of Electrical Cabinet R1400S038 Scenario 5

Simplified fire modeling has shown that ignition of R1400S038 damages circuits for safe shutdown components X4100N057B, X4103F153, X4103F154 and R1400S038. The CCDF calculated for this damage set by PSA Model R2S02 is 1.3E-5. The ignition frequency is (6/151)(2.4E-3) = 9.54E-5/yr.

CDF for this scenario is (9.54E-5)(1.3E-5) = 1.24E-9/yr.

Ignition of Electrical Cabinet R1400S002C Scenario 30

Simplified fire modeling has shown that ignition of R1400S002C damages circuits for the panel itself. The CCDF calculated for this damage set by PSA Model R2S02 is 1.3E-5. The ignition frequency is (4/151)(2.4E-3) = 6.36E-5/yr.

CDF for this scenario is (6.36E-5)(1.3E-5) = 8.27E-10/yr.

Ignition of Electrical Cabinet R1600S018A Scenario 31

Simplified fire modeling has shown that ignition of R1600S018A does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set by PSA Model R2S02 is 1.3E-5. The ignition frequency is (23/151)(2.4E-3) = 3.66E-4/yr.

CDF for this scenario is (3.66E-4)(1.3E-5) = 4.76E-9/yr.

Ignition of Electrical Cabinet R1600S018B Scenario 32

Simplified fire modeling has shown that ignition of R1600S018B damages circuits for safe shutdown components H2100P352, H2100P518, R3000P331, R3000S007, X4103F127, X4103F128, X4103F153, X4103F154, X4103N056B and X4103N057B, and the panel itself. The CCDF calculated for this damage set by PSA Model R2S02 is 1.3E-5. The ignition frequency is (17/151)(2.4E-3) = 2.70E-4/yr.

CDF for this scenario is (2.70E-4)(1.3E-5) = 3.51E-9/yr.

Ignition of Electrical Cabinet R1600S019A Scenario 33

Simplified fire modeling has shown that ignition of R1600S019A does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set by PSA Model R2S01 is 2.9E-4. The ignition frequency is (24/151)(2.4E-3) = 3.81E-4/yr.

CDF for this scenario is (3.81E-4)(2.9E-4) = 1.10E-7/yr.

Ignition of Electrical Cabinet R3200S066 Scenario 34

Simplified fire modeling has shown that ignition of R3200S066 does not damage circuits for any other safe shutdown component. The CCDF calculated for this damage set by PSA Model R2S05 is 3.3E-4. The ignition frequency is (21/151)(2.4E-3) = 3.34E-4/yr.

CDF for this scenario is (3.34E-4)(3.3E-4) = 1.1E-7/yr.

Ignition of Electrical Cabinet R1700S015B Scenario 37

Simplified fire modeling has shown that ignition of R1700S015B damages circuits for safe shutdown component X4103F146. This cabinet is not a safe shutdown component. The CCDF calculated for this damage set is enveloped by the base model CCDF of 1.3E-5. The ignition frequency is (1/151)(2.4E-3) = 1.58E-5/yr.

CDF for this scenario is (1.58E-5)(1.3E-5) = 2.05E-10/yr.

Ignition of Electrical Cabinet R3000S007 Scenario 39

Simplified fire modeling has shown that ignition of R3000S007 damages circuits for safe shutdown components H1100P862, X4103F139, X4103F140, X4103F155, X4103F156, X4103N056D and X4103N057D, and the panel itself. The CCDF calculated for this damage set enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (1/151)(2.4E-3) = 1.58E-5/yr.

CDF for this scenario is (1.58E-5)(1.3E-5) = 2.05E-10/yr.

Ignition of Electrical Cabinet R3000S008 Scenario 40

Simplified fire modeling has shown that ignition of R3000S008 damages circuits for safe shutdown components H1100P862, X4103F127, X4103F128, X4103F153, X4103F154, X4103N056B and X4103N057B, and the panel itself. The CCDF calculated for this damage set enveloped by PSA Model R2S03 is 1.3E-5. The ignition frequency is (1/151)(2.4E-3) = 1.58E-5/yr.

CDF for this scenario is (1.58E-5)(1.3E-5) = 2.05E-10/yr.

Ignition of Transformers R1600S120B, R1600S123B, and R1700S013B Scenario 44

Simplified fire modeling has shown that ignition of R1600S120B, R1600S123B, and R1700S013B does not result in damage to circuits for other safe shutdown components. These transformers are not safe shutdown components. The base model CCDF is applied to these components. The ignition frequency is (3/7)(6.8E-4) = 2.91E-4/yr.

CDF for this scenario is (2.91E-4)(1.3E-5) = 3.79E-9/yr.

Ignition of Heater X4103B245 Scenario 34

Simplified fire modeling has shown that ignition of X4103B245 damages circuits for safe shutdown component R3200S066. The CCDF calculated for this damage set by PSA Model R2S05 is 3.3E-4. The ignition frequency is (1/40)(1.1E-3) = 2.8E-5/yr.

CDF for this scenario is (2.8E-5)(3.3E-4) = 9.24E-9/yr.

Ignition of Heater X4103B244 Scenario 45

Simplified fire modeling has shown that ignition of heater X4103B244 does not result in damage to circuits for any safe shutdown components. This heater is not a safe shutdown component. The base model CCDF is applied to this component. The ignition frequency is (1/40)(1.1E-3) = 2.8E-5/yr.

CDF for each scenario is (2.8E-5)(1.3E-5) = 3.64E-10/yr.

Ignition of Transient Combustible Scenario

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. As calculated earlier, P_{tc} is 0.00626. For conservatism the worst case RHR2 CCDF (calculated by PSA Model RHR2 as 4.4E-3) is applied to the transient combustibles. The ignition frequency is (1.2E-4 + 7.2E-4 + 3.3E-4) = 1.17E-3/yr.

CDF for this scenario is (1.17E-3)(4.4E-3)(0.00626) = 3.2E-8/yr.

The CDF for the above scenarios are added to obtain the Total CDF for fire area RHR2SG:

SCENARIO	CDF
Electrical Cabinet Scenario 1	2.05E-10/yr
Electrical Cabinet Scenario 2	2.05E-10/yr
Electrical Cabinet Scenario 3	1.84E-8/yr
Electrical Cabinet Scenario 4	3.21E-8/yr
Electrical Cabinet Scenario 5	1.24E-9/yr
Electrical Cabinet Scenario 30	8.27E-10/yr
Electrical Cabinet Scenario 31	4.76E-9/yr
Electrical Cabinet Scenario 32	3.51E-9/yr
Electrical Cabinet Scenario 33	1.10E-7/yr
Electrical Cabinet Scenario 34	1.10E-7/yr
Electrical Cabinet Scenario 37	2.05E-10/yr
Electrical Cabinet Scenario 39	2.05E-10/yr

Electrical Cabinet Scenario 40	2.05E-10/yr
Remaining Electrical Cabinets	3.30E-9/yr
Transformer Scenario 44	3.79E-9/yr
Heater Scenario 34	9.24E-9/yr
Heater Scenario 45	3.64E-10/yr
Transient Combustible	3.2E-8/yr
TOTAL	3.3E-7/yr

The results of this analysis indicate that the Fire Area RHR2SG is not fire risk significant. The total CDF for this scenario for this fire area is 3.3E-7/yr, allowing the Fire Area to screen.

4A.3.4.3.4 Phase III Analysis for the Radwaste Building

Based upon a review of the routing for the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the Radwaste Building, ignition sources are identified which adversely impact Safe Shutdown circuits.

Ignition of R1400S026 and R1400S024 Scenario 1:

Simplified fire modeling has shown that ignition of R1400S026 would damage circuits supporting components R1400S026 and N2103C002. Simplified fire modeling has shown that ignition of R1400S024 would damage circuits supporting components R1400S024, H2100P623 and G1112P600. PSA Model RW1 calculated the CCDF for this damage set as 4.6E-4. The ignition frequency of two 480V substation buses is (24/452)(8.7E-3) = 4.62E-4/yr.

CDF for Ignition of R1400S026 and R1400S024: (4.62E-4)(4.6E-4) = 2.12E-7/yr

Ignition of R1400S027 Scenario 2:

Simplified fire modeling has shown that ignition of R1400S027 would damage circuits supporting components R1400S027, H2100P623 and G1112P600. PSA Model RW2 calculated the CCDF for this damage set as 4.6E-4. The ignition frequency of one 480V substation bus is (12/452)(8.7E-3) = 2.31E-4/yr.

CDF for Ignition of R1400S027: (2.31E-4)(4.6E-4) = 1.06E-7/yr

Ignition of R1400S001V, R1400S001W & R1400S001L Scenario 3:

Simplified fire modeling has shown that ignition of R1400S001V and R1400S001W would damage circuits supporting components R1400S001V, R1400S001W and R3200S067. Simplified fire modeling has shown that ignition of R1400S001L would damage circuits supporting components R1400S001L, R1600S011D, R3200017 and R3200S018. PSA Model RW3 calculated the CCDF for this damage set as 5.1E-4. The ignition frequency of three 4160V switchgear buses is (9/452)(8.7E-3) = 1.73E-4/yr.

CDF for Ignition of R1400S001V, R1400S001W & R1400S001L: (1.73E-4)(5.1E-4) = 8.83E-8/yr

Ignition of H2100P623 Scenario 4:

Simplified fire modeling has shown that ignition of H2100P623 would damage circuits supporting components R1400S001V, R1400S001W, R3200S017, G1112P600, C3601P001, and H2100P623. PSA Model RW4 calculated the CCDF for this damage set as 4.6E-4. The ignition frequency of the one electrical cabinet is (1/452)(8.7E-3) = 1.9E-5/yr.

CDF for Ignition of H2100P623: (4.6E-4)(1.9E-5) = 8.85E-9/yr

Ignition of R1400S001A Scenario 5:

Simplified fire modeling has shown that ignition of R1400S001A would damage circuits supporting components C3601P001 and R1400S001A. PSA Model RW5 calculated the CCDF for this damage set as 4.6E-4. The ignition frequency of one 4160V switchgear bus is (9/452)(8.7E-3) = 1.73E-4/yr.

CDF for Ignition of R1400S001A: (1.73E-4)(4.6E-4) = 7.97E-8/yr

Ignition of R1400S001D Scenario 6:

Simplified fire modeling has shown that ignition of R1400S001D would damage circuits supporting components C3601P001, R3200S067, R1400S001D and N2103C002. PSA Model RW6 calculated the CCDF for this damage set as 5.1E-4. The ignition frequency of one 4160V switchgear is (9/452)(8.7E-3) = 1.73E-4/yr.

CDF for Ignition of R1400S001D: (5.1E-4)(1.73E-4) = 8.82E-8/yr

Ignition of R3200S067, R3200S009A, R3200S009B, R3200S012 and R3200S005 Scenario 7:

Simplified fire modeling has shown that ignition of R3200S067 would damage circuits supporting components R3200S067. Simplified fire modeling has shown that ignition of R3200S005 would damage circuits supporting components R3200S005. Simplified fire modeling has shown that ignition of R3200S009A and R3200S009B would damage circuits supporting components R3200S018, R3200S009A and R3200S009B. Simplified fire modeling has shown that ignition of R3200S009B and R3200S012 would damage circuits supporting components R3200S009B and R3200S012. PSA Model RW7 calculated the CCDF for this damage set as 5.2E-4. The ignition frequency of one electrical cabinet and one 260/130V dual battery is (13/452)(8.7E-3) = 2.5E-4/yr.

CDF for Ignition of R3200S067, R3200S009A, R3200S009B, R3200S012 and R3200S005: (2.5E-4)(5.2E-4) = 1.30E-7/yr

Ignition of R3100S014 and R3100S015 Scenario 8:

Simplified fire modeling has shown that ignition of R3100S014 and R3100S015 would damage circuits supporting components R3100S014 and R3100S015. PSA Model RW8 calculated the CCDF for this damage set as 4.7E-4. The ignition frequency of two UPS cabinets is (2/452)(8.7E-3) = 3.8E-5/yr.

CDF for Ignition of R3100S014 and R3100S015: (4.7E-4)(3.8E-5) = 1.79E-8/yr

Ignition of H2100P304 Scenario 9:

Simplified fire modeling has shown that ignition of H2100P304 would damage circuits supporting components H2100P304 and N2103C002. PSA Model RW9 calculated the CCDF for this damage set as 4.7E-4. The ignition frequency of one electrical cabinet is (1/452)(8.7E-3) = 1.9E-5/yr.

CDF for Ignition of H2100P304: (1.9E-5)(4.7E-4) = 8.9E-9/yr

Ignition of V4100B001A Scenario 10:

Simplified fire modeling has shown that ignition of V4100B001A would damage circuits supporting component V4100B001A. The baseline PSA Model calculated the CCDF for this damage set as 1.3E-5. The ignition frequency of one HVAC unit is (1/13)(3.7E-4) = 2.85E-5/yr.

CDF for Ignition of V4100B001A: (2.85E-5)(1.3E-5) = 3.71E-10/yr

Ignition of Remaining Miscellaneous Components Scenario 11:

Simplified fire modeling for the remaining miscellaneous components (371) showed there would be no damage to any other SSD circuits. Therefore, the base model CCDF is applied to the remaining ignition sources. The ignition frequency is (371/452)(8.7E-3) = 7.14E-3/yr.

CDF for Ignition of the remaining miscellaneous components: (7.14E-3)(1.3E-5)= 9.28E-8/yr

Ignition of Remaining HVAC Scenario 12:

Simplified fire modeling for the remaining HVAC units (12) showed there would be no damage to any other SSD circuits. Therefore, the base model CCDF is applied to the remaining ignition sources. The ignition frequency is (12/13)(3.7E-4) = 3.42E-4/yr

CDF for Ignition of the remaining HVAC: (3.42E-4)(1.3E-5)= 4.45E-9/yr

Ignition of Fire Protection Panels Scenario 13:

Simplified fire modeling for the fire protection panels (8) showed there would be no damage to any other SSD circuits. Therefore, the base model CCDF is applied to the remaining ignition sources. The ignition frequency is 9.7E-5/yr.

CDF for Ignition of the fire protection panels: (9.7E-5)(1.3E-5)= 1.26E-9/yr

Ignition of Battery Chargers Scenario 14:

Simplified fire modeling for the five battery chargers showed there would be no damage to any other SSD circuits. Therefore, the base model CCDF is applied to the remaining ignition sources. The ignition frequency is 1.2E-3/yr

CDF for Ignition of the remaining battery chargers: (1.2E-3)(1.3E-5)= 1.56E-8/yr

Ignition of Transient Combustible Scenario 15:

The CCDF representing loss of the entire fire area, 1.4E-2, was applied to the transient combustibles. This CCDF was calculated by PSA Model RADWST. The ignition frequency for transient combustibles is 1.17E-3/yr.

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. As calculated earlier in this report the P_{tc} is defined as .00626. Therefore, utilizing P_{tc}, the Transient Combustible the CDF for this Scenario is: (1.17E-3)(.00626)(1.4E-2)= 1.03E-7/yr

The individual CDF values for the above scenarios are added to obtain the total CDF:

SCENARIO	CDF
Scenario 1	2.12E-7/yr
Scenario 2	1.06E-7/yr
Scenario 3	8.83E-8/yr
Scenario 4	8.85E-9/yr
Scenario 5	7.97E-8/yr
Scenario 6	8.82E-8/yr
Scenario 7	1.3E-7/yr
Scenario 8	1.79E-8/yr
Scenario 9	8.9E-9/yr
Scenario 10	3.71E-10/yr
Scenario 11	9.28E-8/yr
Scenario 12	4.45E-9/yr
Scenario 13	1.26E-9/yr
Scenario 14	1.56E-9/yr
Scenario 15	1.03E-7/yr
TOTAL	9.57E-7/yr

The results of this analysis indicate that the Fire Area Radwaste Building is not fire risk significant. The total CDF for this fire area is 9.57E-7/yr, allowing the Fire Area to screen. No recommendations for further action are necessary.

4A.3.4.3.5 Phase III Analysis of Fire Area TB

The approach utilized in the Phase III analysis of fire area TB is to sub-compartmentalize the fire area TB by floor. This approach is available based on the existence of internal barriers within the Turbine Building (i.e. three foot thick reinforced concrete floors). Unsealed penetrations exist in the floors as mechanical penetrations, electrical penetrations and open stairways between the floors. Considering the floor area at each elevation, ceiling height and the quantity of combustibles, propagation of significant amounts of flame, heat or other products of combustion are not expected to adversely affect plant components on other elevations. This approach is similar to the division of the Reactor Building into separate fire areas. The fire areas for the Turbine Building are fire areas TBb (Basement), TB1 (first floor), TB2 (second floor) and TB3 (third floor).

Additionally, the Waste Oil Room is compartmentalized from the Turbine Building as a separate fire area. The barriers for the Waste Oil Room are a minimum twelve inch concrete barriers, thus sufficiently confining flame and other products of combustion within the area. The Waste Oil Room is designated as fire area TBWOR.

Based upon a review of the routing of the Safe Shutdown circuits, fire modeling of the fixed combustibles and consideration of compartmental and spatial separation within the Turbine

Building, ignition sources are identified which adversely impact Safe Shutdown circuits. Miscellaneous hydrogen fires are applied across the entire Turbine Building as the hydrogen lines run through each floor. Additionally, the transient source is applied evenly to all floors. The remaining ignition sources are not considered to damage any Safe Shutdown circuits. The following scenarios describe the ignition sources determined to impact Safe Shutdown circuits.

4A.3.4.3.5.1 Phase III Analysis of Fire Area TBb

Ignition of R1600S006C and R1600S011B Scenario 1:

Simplified fire modeling has shown that ignition of either R1600S006C and R1600S011B would damage circuits supporting N2103C001, N2100F002, N2100F003, N2103F002 and N2103F003. PSA Model TB2 calculated the CCDF for this damage set as 4.6E-4. The ignition frequency for this event is 1.1E-3/yr = (72/867)(1.3E-2). Therefore, the CDF for this scenario is (1.1E-3)(4.6E-4) = 5E-7/yr

Ignition of Standby Feedwater Pumps Scenario 17:

The ignition of the Standby Feedwater Pumps were conservatively assumed to damage circuits supporting H21P623, N2103C001, N2103C002 and also the Standby Feedwater Pumps themselves. PSA Model TB2 calculated the CCDF for this damage set as 4.6E-4. The ignition frequency for this event is 3.2E-4/yr = (2/32)(5.2E-3).Therefore, the CDF for this scenario is (3.2E-4)(4.6E-4) =1.5E-7/yr.

The Transient Ignition and the Miscellaneous Hydrogen Fire Scenario:

Both the transient ignition frequency and the miscellaneous hydrogen fire are divided evenly over the four floors (basement and first through third floors). The ignition frequency at each floor is conservatively applied to the worst case PSA Model CCDF calculated for that floor. The worst case CCDF calculated for the basement is 4.6E-4. Considering that the ignition frequency for transient and hydrogen ignition is applied across all four floors, the ignition frequency for this scenario is (1/4)(7.4E-5) + (1/4)(1.17E-3)(.00626) = 2.03E-5/yr. Therefore, the CDF for this scenario is (2.03E-5)(4.6E-4) = 9.3E-9/yr.

The CDF for the above scenarios are added to obtain the Total CDF for fire area TBb:

SCENARIO	CDF
Electrical Cabinet Scenario 1	5E-7/yr
SBFW Pump Scenario 17	1.5E-7/yr
Transient and Hydrogen Scenario	9.3E-9/yr
TOTAL	6.6E-7/yr

The results of this analysis indicate that the Fire Area TBb is not fire risk significant. The CCDF for this scenario for this fire area is 6.6E-7/yr, allowing this fire area to screen. No recommendations for further action are necessary.

4A.3.4.3.5.2 Phase III Analysis of Fire Area TB1

Ignition of P5001D001 Scenario 2:

In discussions with plant personnel, station air compressors contain approximately fifteen gallons of oil, however a large spill is highly unlikely. The most probable quantity of oil would result from the leakage of a few drops of oil a day. Assuming that fourteen days would be the longest period for the oil drops to accumulate (based on period between housekeeping inspections), a quart of oil is conservatively modeled. Simplified fire modeling has shown that ignition of one quart of oil would damage circuit 227682 (tray 0K-343) supporting component C3601P001. Analysis of circuit 227682 indicates that loss of this circuit would not adversely impact the performance of component C3601P001 (given off-site power available). Therefore, the base model CCDF (1.3E-5) is applied to this scenario. The ignition frequency for this event is 5.3E-4/yr = (1/3)(1.6E-3).

Therefore, the CDF for this scenario is (5.3E-4)(1.3E-5) =6.9E-9/yr.

Ignition of P5001D002 Scenario 3:

In discussions with plant personnel, station air compressors contain approximately fifteen gallons of oil, however a large spill is highly unlikely. The most probable quantity of oil would result from the leakage of a few drops of oil a day. Assuming that fourteen days would be the longest period for the oil drops to accumulate (based on period between housekeeping inspections), a quart of oil is conservatively modeled. Simplified fire modeling has shown that ignition of one quart of oil would damage circuits supporting H21P623. PSA Model TB2 calculated the CCDF for this damage set as 4.6E-4. The ignition frequency for this event is 5.3E-4/yr = (1/3)(1.6E-3).

Therefore, the CDF for this scenario is (5.3E-4)(4.6F-4) = 2.4E-7/yr.

Ignition of P5001D003 Scenario 4:

In discussions with plant personnel, station air compressors contain approximately fifteen gallons of oil, however a large spill is highly unlikely. The most probable quantity of oil would result from the leakage of a few drops of oil a day. Assuming that fourteen days would be the longest period for the oil drops to accumulate (based on period between housekeeping inspections), a quart of oil is conservatively modeled. Simplified fire modeling has shown that ignition of one quart of oil would damage circuits supporting N2103C001. PSA Model TB2 calculated the CCDF for this damage set as 4.6E-4. The ignition frequency for this event is 5.3E-4/yr = (1/3)(1.6E-3).

Therefore, the CDF for this scenario is (5.3E-4)(4.6E-4) = 2.4E-7/yr.

Ignition of R1600S026 Scenario 6:

Simplified fire modeling has shown that ignition of R1600S026 would damage circuits 227681 and 227682 (tray 0K-157) supporting component C3601P001. Analysis of circuit 227681 and 227682 indicates that loss of these circuits would not adversely impact the performance of component C3601P001 (given off-site power available). Therefore, the base model CCDF (1.3E-5) is applied to this scenario. The ignition frequency for this event is 2.1E-4/yr = (14/867)(1.3E-2).

Therefore, the CDF for this scenario is (2.1E-4)(1.3E-5) = 2.7E-9/yr.

Ignition of N2003C014 and N2003C013 Scenario 7:

In discussion with plant personnel, the heater feed pumps were described as likely to lose a large amount of oil. Therefore, the probability of the large versus small oil fire is applied to this fire area. The large oil spill if considered to be a one gallon spill, covering 120 square feet. The small oil spill is approximated as a one pint spill (oil leakage and oily rags) covering approximately fifteen square feet. Simplified fire modeling has shown that ignition of a large oil spill (probability of 0.18) would damage circuits supporting C3601P001 and N2103C002. PSA Model TB11 calculated the CCDF for this damage set as 9.3E-4. Simplified fire modeling has shown that ignition of a small oil spill (probability of 0.82) would damage only circuit 227682 supporting component C3601P001. Analysis of circuit 227682 indicates that loss of this circuit would not adversely impact the performance of component C3601P001 (given off-site power available). Therefore, the base model CCDF (1.3E-5) is applied to this small oil spill scenario. The ignition frequency for this event is 3.2E-4/yr = (2/32)(5.2E-3).

Therefore, the CDF for this scenario is: (0.18)(3.2E-4)(9.3E-4) + (0.82)(3.2E-4)(1.3E-5) = 5.7E-8/yr.

Ignition of N2102C022 and N2102C023 Scenario 8:

In discussion with plant personnel, the Reactor Feed Pumps Seal Water Injection Pumps contain approximately one gallon of oil. The probability of the large versus small oil fire is applied to this fire area. The large oil spill is considered to be a one gallon spill, covering 120 square feet. The small oil spill is approximated as a one pint spill (oil leakage and oily rags) covering approximately fifteen square feet. Simplified fire modeling has shown that ignition of a large oil spill (probability of 0.18) would damage circuits supporting C3601P001, N2103C002 and R3200S0018. Analysis of circuit 227682 (for component C3601P001) indicates that loss of this circuit would not adversely impact the performance of component C3601P001 (given off-site power available). Therefore, PSA Model TB12 calculated the CCDF for loss of N2103C002 and R3200S0018 as 9.9E-4. Simplified fire modeling has shown that ignition of a small oil spill (probability of 0.82) would not damage C3601P001, N2103C002 and R3200S0018. Therefore, the base model CCDF (1.3E-5) is applied to this small oil spill scenario. The ignition frequency for this event is 3.2E-4/yr = (2/32)(5.2E-3)

Therefore, the CDF for this scenario is: (0.18)(3.2E-4)(9.9E-4) + (0.82)(3.2E-4)(1.3E-5) = 6.0E-8/yr.

Ignition of N3020C042 and P1100C001A,B,C Scenario 9:

The ignition of N3020C042 and P1100C001A,B,C is conservatively assumed to would damage circuit 227682 (tray 0K-331) supporting component C3601P001. Tray 0K-331 is the nearest Safe Shutdown raceway within approximately twenty feet of the ignition sources. Analysis of circuit 227682 indicates that loss of this circuit would not adversely impact the performance of component C3601P001 (given off-site power available). Therefore, the base model CCDF (1.3E-5) is applied to this scenario. The ignition frequency for this event is 6.5E-4/yr = (4/32)(5.2E-3).

Therefore, the CDF for this scenario is (6.5E-4)(1.3E-5) = 8.5E-9/yr.

Ignition of N2003C012 Scenario 10:

In discussion with plant personnel, the heater feed pumps were described as likely to lose a large amount of oil. Therefore, the probability of the large versus small oil fire is applied to this fire area. The large oil spill if considered to be a one gallon spill, covering 120 square feet. The small oil spill is approximated as a one pint spill (oil leakage and oily rags) covering approximately fifteen square feet. Simplified fire modeling has shown that ignition of a large oil spill (probability of 0.18) would damage circuits supporting C3601P001, N2103P003, N2103P002 and

N2103C002. PSA Model TB13 calculated the CCDF for this damage set as 9.3E-4. Simplified fire modeling has shown that ignition of a small oil spill (probability of 0.82) would damage only circuit 227682 supporting component C3601P001. Analysis of circuit 227682 indicates that loss of this circuit would not adversely impact the performance of component C3601P001 (given off-site power available). Therefore, the base model CCDF (1.3E-5) is applied to this small oil spill scenario. The ignition frequency for this event is 1.6E-4/yr = (1/32)(5.2E-3).

Therefore, the CDF for this scenario is: (0.18)(1.6E-4)(9.3E-4) + (0.82)(1.6E-4)(1.3E-5) = 2.9E-8/yr.

Ignition of N2002C010A Scenario 11:

Simplified fire modeling has shown that ignition of N2002C010A would damage circuits supporting H21P623. PSA Model TB2 calculated the CCDF for this damage set as 4.6E-4. The ignition frequency for this event is 1.6E-4/yr = (1/32)(5.2E-3).

Therefore, the CDF for this scenario is (1.6E-4)(4.6E-4) = 7.4E-8/yr.

Ignition of N2002C009A,B and N2002C010E,D Scenario 12:

In discussion with plant personnel, the Condensate Polishing Demineralizer Pumps contain approximately one gallon of oil. The probability of the large versus small oil fire is applied to this fire area. The large oil spill is considered to be a one gallon spill, covering 120 square feet. The small oil spill is approximated as a one pint spill (oil leakage and oily rags) covering approximately fifteen square feet. Simplified fire modeling has shown that ignition of a large oil spill (probability of 0.18) would damage circuits supporting C3601P001, N2100F001, N2100F002 and N2100F003. Analysis of circuit 227682 (for component C3601P001) indicates that loss of this circuit would not adversely impact the performance of component C3601P001 (given off-site power available). Therefore, PSA Model TB2 calculated the CCDF for loss of N2100F001, N2100F002 and N2100F003 as 4.6E-4. Simplified fire modeling has shown that ignition of a small oil spill (probability of 0.82) would not damage N2100F001, N2100F002 and N2100F003. Therefore, the base model CCDF (1.3E-5) is applied to this small oil spill scenario. The ignition frequency for this event is 6.5E-4/yr = (4/32)(5.2E-3).

Therefore, the CDF for this scenario is: (0.18)(6.5E-4)(4.6E-4) + (0.82)(6.5E-4)(1.3E-5) = 6.1E-8/yr.

The Transient Ignition and the Miscellaneous Hydrogen Fire Scenario:

Both the transient ignition frequency and the miscellaneous hydrogen fire are divided evenly over the four floors (basement and first through third floors). The ignition frequency at each floor is conservatively applied to the worst case PSA Model CCDF calculated for that floor. The worst case CCDF calculated for the first floor is 4.5E-3. Considering that the ignition frequency for transient and hydrogen ignition is applied across all four floors, the ignition frequency for this scenario is (1/4)(7.4E-5) + (1/4)(1.17E-3)(.00626) = 2.03E-5/yr.

Therefore, the CDF for this scenario is (2.03E-5)(4.5E-3) = 9.13E-8/yr.

The CDF for the above scenarios are added to obtain the total revised CDF for fire area TB1:

SCENARIO	CDF
Compressor Scenario 2	6.9E-9/yr

Compressor Scenario 3	2.4E-7/yr
Compressor Scenario 4	2.4E-7/yr
Electrical Cabinet Scenario 6	2.7E-9/yr
Pump Scenario 7	5.7E-8/yr
Pump Scenario 8	6.0E-8/yr
Pump Scenario 9	8.5E-9/yr
Pump Scenario 10	2.9E-8/yr
Pump Scenario 11	7.4E-8/yr
Pump Scenario 12	6.1E-8/yr
Transient and Hydrogen Scenario	9.13E-8/yr
TOTAL	8.7E-7/yr

The results of this analysis indicate that the Fire Area TB1 is not fire risk significant. The total revised CDF for this scenario for this fire area is 8.7E-7/yr, allowing this fire area to screen. No recommendations for further action are necessary.

4A.3.4.3.5.3 Phase III Analysis of Fire Area TB2

Ignition of R3200S017 Scenario 13:

Simplified fire modeling has shown that ignition of R3200S017 would damage circuits supporting R3200S017, N2103P001, N2103P002 and N2103P003. PSA Model TB4 calculated the CCDF for this damage set as 5.2E-4. The ignition frequency for this event is 1.8E-4/yr = (12/867)(1.3E-2).

Therefore, the CDF for this scenario is (1.8E-4)(5.2E-4) = 9.4E-8/yr.

Ignition of R1600S015A Scenario 14:

Simplified fire modeling has shown that ignition of R1600S015A would damage circuits supporting R3200S017, N2103P001, N2103P002 and N2103P003. PSA Model TB4 calculated the CCDF for this damage set as 5.2E-4. The ignition frequency for this event is 4.2E-4/yr = (28/867)(1.3E-2).

Therefore, the CDF for this scenario is (4.2E-4)(5.2E-4) =2.2E-7/yr.

Ignition of H21P629 Scenario 15:

Simplified fire modeling has shown that ignition of H21P629 would damage circuits supporting R3200S017 and H21P629. PSA Model TB4 calculated the CCDF for this damage set as 5.2E-4. The ignition frequency for this event is 1.5E-5/yr = (1/867)(1.3E-2). Therefore, the CDF for this scenario is (1.5E-5)(5.2E-4) =7.8E-9/yr.

Ignition of R3200S018 Scenario 16:

Simplified fire modeling has shown that ignition of R3200S018 would damage circuits supporting R3200S018. PSA Model TB5 calculated the CCDF for this damage set as 5.2E-4. The ignition frequency for this event is 1.0E-4/yr = (7/867)(1.3E-2).

Therefore, the CDF for this scenario is (1.0E-4)(.E-4) =5.2E-8/yr.

The Transient Ignition and the Miscellaneous Hydrogen Fire Scenario:

Both the transient ignition frequency and the miscellaneous hydrogen fire are divided evenly over the four floors (basement and first through third floors). The ignition frequency at each floor is conservatively applied to the worst case PSA Model CCDF calculated for that floor. The worst case CCDF calculated for the second floor is 5.2E-4. Considering that the ignition frequency for transient and hydrogen ignition is applied across all four floors, the ignition frequency for this scenario is (1/4)(7.4E-5) + (1/4)(1.17E-3)(.00626) = 2.03E-5/yr.

Therefore, the CDF for this scenario is (2.03E-5)(5.2E-4) = 1.1E-8/yr.

The CDF for the above scenarios are added to obtain the Total CDF for fire area TB2:

SCENARIO	CDF
Electrical Cabinet Scenario 13	9.4E-8/yr
Electrical Cabinet Scenario 14	2.2E-7/yr
Electrical Cabinet Scenario 15	7.8E-9/yr
Electrical Cabinet Scenario 16	5.2E-8/yr
Transient and Hydrogen Scenario	1.1E-8/yr
TOTAL	3.8E-7/yr

The results of this analysis indicate that the Fire Area TB2 is not fire risk significant. The CCDF for this scenario for this fire area is 3.8E-7/yr, allowing this fire area to screen. No recommendations for further action are necessary.

4A.3.4.3.5.4 Phase III Analysis of Fire Area TB3

The Transient Ignition and the Miscellaneous Hydrogen Fire Scenario:

Both the transient ignition frequency and the miscellaneous hydrogen fire are divided evenly over the four floors (basement and first through third floors). The ignition frequency at each floor is conservatively applied to the worst case PSA Model CCDF calculated for that floor. The worst case CCDF calculated for the third floor is 5.2E-4. Considering that the ignition frequency for transient and hydrogen ignition is applied across all four floors, the ignition frequency for this scenario is (1/4)(7.4E-5) + (1/4)(1.17E-3)(.00626) = 2.03E-5/yr. Therefore, the CDF for this scenario is (2.03E-5)(5.2E-4) = 1.1E-8/yr.

The results of this analysis indicate that the Fire Area TB3 is not fire risk significant. The CCDF for this scenario for this fire area is 1.1E-8, allowing this fire area to screen. No recommendations for further action are necessary.

4A.3.4.3.5.5 Phase III Analysis of Fire Area TBWOR

Ignition of any of the six pumps in the Waste Oil Room Scenario:

The six pump ignition sources in this fire area are: N3014C040, P7000C040, P7000C041, P7000C042, P7000C044 and P7000C045. The probability of the large versus small oil fire is applied to this fire area. Simplified fire modeling has shown that ignition of a large oil spill (probability of 0.18) originating from any of the six pumps would damage circuits supporting

C3601P001 and R1400S001B. This is based on a conservative loss of the entire area from a large oil spill fire. PSA Model TB3 calculated the CCDF for this large oil spill damage set as 4.3E-3. Simplified fire modeling has shown that ignition of a small oil spill (probability of 0.82) would not damage circuits supporting C3601P001 and R1400S001B and that only the originating pumps would be lost. The six pumps are not Safe Shutdown components nor are they included in the B.O.P. list for this fire area. Therefore, the base model CCDF (1.3E-5) is applied to this small oil spill scenario. The ignition frequency for this event is 9.8E-4/yr = (6/32)(5.2E-3).

Therefore, the CDF for this scenario is: (0.18)(9.8E-4)(4.3E-3) + (0.82)(9.8E-4)(1.3E-5) = 7.69E-7/yr

Ignition of Transient Combustible Scenario

The probability of transient fire exposure (P_{tc}) was calculated as defined in Section 6.3.7.2 of the FIVE Methodology. As calculated earlier, P_{tc} is 0.00626. For conservatism the Waste Oil Room CCDF (calculated by PSA Model TB3 as 4.3E-3) is applied to the transient combustibles. The ignition frequency is (1.2E-4 + 7.2E-4 + 3.3E-4) = 1.17E-3/yr.

Therefore, the CDF for this scenario is (1.17E-3)(0.00626)(4.3E-3) = 3.15E-8/yr.

The CDF for the above scenarios are added to obtain the Total CDF for fire area TBWOR:

SCENARIO	CDF
Pump Scenario	7.69E-7/yr
Transient Scenario	3.15E-8/yr
TOTAL	8.0E-7/yr

The results of this analysis indicate that the Fire Area TBWOR is not fire risk significant. The CDF for this scenario for this fire area is 8.0E-7/yr, allowing this fire area to screen. No recommendations for further action are necessary.

4A.3.4.3.6 Phase III Analysis for the USRCD

The Division 1 and Division 2 Underground Safety Related Cable Ducts (USRCD) extend from the RHR Complex to the Auxiliary Building cable vault. The Ducts are separated by at least ten (10) feet of soil and covered by two (2) feet of soil. Both Ducts are accessible by a manhole structure (M.H. 16946 and M.H. 16947). The top of each manhole is covered with approximately one (1) foot of soil and gravel. The cables in each of the ducts are routed through 30 fiber conduits. The conduits in each duct are separated by approximately three (3) inches of concrete.

The CCDF for loss of the individual Division 1 and Division 2 Ducts calculated by PSA Model USRCD1 and USRCD2 are 2.9E-2 and 5.2E-3 respectively.

The ignition frequency for this area is 9.0E-4/yr. This ignition frequency is based solely on transient ignition sources. Self ignition of the cables is not considered viable as the cables are qualified to be equivalent to IEEE 383 cables.

Based on the construction of the Underground Safety Related Cable Ducts (USRCD), loss of both Divisions resulting from a fire in the yard is unrealistic. The most credible scenario is the loss of a single Division from a transient source while the Duct is exposed (maintenance, etc.).

Therefore the probability of fire exposure from transient combustibles (P_{tc}) is evaluated. P_{tc} is calculated to be 0.00626. The CDF for a transient fire scenario is:

CDF for Loss of Division 1: (9.00E-4)(2.9E-2)(.00626) = 1.63E-7/yr

CDF for Loss of Division 2: (9.00E-4)(5.2E-3)(.00626) = 2.93E-8/yr

Total CDF for the USRCD: 1.92E-7/yr

The results of this analysis indicate that the Fire Area USRCD is not fire risk significant. The total CDF for this fire area is 1.92E-7/yr, allowing the Fire Area to screen. No recommendations for further action are necessary.

4A.3.4.3.7 Phase III analysis for Fire Area YARD

Four scenarios were identified based on the damage sets of Safe Shutdown circuits routed through the YARD area: Loss of R1200S002 and R1100S058; Loss of H2100P492 and H2100P623; Loss of C3601P002, U001583T01 and U001583T02; and Loss of R1400S011B and R1400S001B. The CCDF calculated for each scenario is as follows:

SCENARIO	CCDF
Loss of R1200S002 and R1100S058 (PSA Model Yard2)	5.2E-3
Loss of H2100P492 and H2100P623 (PSA Model Yard4)	4.6E-4
Loss of C3601P002, U001583T01 and U001583T02 (PSA Model Yard5)	4.6E-4
Loss of R1400S011B and R1400S001B (PSA Model Yard3)	4.5E-3

Based on fire modeling and spatial separation, the ignition sources were identified which could impact each scenario. In all scenarios, the transient ignition source is considered.

SCENARIO	IGNITION SOURCE
Loss of R1200S002 and R1100S058	S.S. No. 64 Reg. Trans. (R1200S002) and transient
Loss of H2100P492 and H2100P623	Condensate Storage Rack (H2100P492) and transient
Loss of C3601P002, U001583T01 and U001583T02	Transient
Loss of R1400S011B and R1400S001B	13.8 KV SWGR Pos A6 (R1400S011B), C.T.G. No. 11 Transformer and transient

Ignition frequencies were developed for these ignition sources. Considering the size of the YARD area, the portion of the YARD available for transient damage of Safe Shutdown circuits/equipment was conservatively estimated to be a quarter (1/4) of its area and was applied evenly to the four scenarios. The calculated ignition frequencies for the scenarios are as follows.

SCENARIO	IGNITION FREQUENCY
Loss of R1200S002 and R1100S058	6.2E-5/yr = (1.6E-3)(1/26) + (1/16)(1.17E-3) (6.26E-3)

Loss of H2100P492 and H2100P623	6.2E-5/yr = (1.6E-3)(1/26) + (1/16)(1.17E-3) (6.26E-3)
Loss of C3601P002, U001583T01 and U001583T02	4.6E-7/yr = (1/16)(1.17E-3)(6.26E-3)
Loss of R1400S011B and R1400S001B	1.2E-4/yr = (1.6E-3)(2/26) + (1/16)(1.17E-3) (6.26E-3)
Remaining Ignition Sources	3.5E-3/yr

The remaining ignition sources do not affect any Safe Shutdown circuits or equipment. The frequency of the remaining ignition sources is applied to the base model CCDF. The total CDF for this area is calculated as:

SCENARIO	**CDF**
Loss of R1200S002 and R1100S058	(5.2E-3)(6.2E-5) = 3.2E-7/yr
Loss of H2100P492 and H2100P623	(4.6E-4)(6.2E-5) = 2.8E-8/yr
Loss of C3601P002, U001583T01 and U001583T02	(4.6E-4)(4.6E-7) = 2.1E-10/yr
Loss of R1400S011B and R1400S001B	(4.5E-3)(1.2E-4) = 5.4E-7/yr
Remaining Ignition Sources	(1.3E-5)(3.5E-3) = 4.6E-8/yr
TOTAL:	9.4E-7/yr

The results of this analysis indicate that the Fire Area YARD is not fire risk significant. The total CDF for this fire area is 9.4E-7/yr, allowing the Fire Area to screen. No recommendations for further action are necessary.

4B FIRE HAZARDS ANALYSIS – AUXILIARY BUILDING

4B.0 FIRE HAZARDS METHODOLOGY AND ASSUMPTIONS

This section provides a summary of the overall fire hazard analysis for the Fermi 2 Auxiliary Building using the FIVE methodology. The summarized information is presented in the same sequence as the methodology discussed in Section 4.0.

The FIVE methodology and this evaluation consist of three phases:

Phase I: Fire Area Screen (Qualitative Analysis)

Phase II: Critical Fire Compartment Screen (Quantitative Analysis)

Phase III: Plant Walkdown/Verification and Documentation

The three phase process of the FIVE methodology is shown in Figure 4-1.

The following major assumptions are used in FIVE analysis of the Fermi 2 Auxiliary Building:

1. In the Phase I analysis, it is assumed that a reactor trip would be generated (either automatically or manually) for all fires in the Auxiliary Building. This assumption is made due to the existence of Abnormal Operating Procedures (AOP) 20.000.18 and 20.000.20.
2. A 24-hour period is assumed as the base mission time for this analysis. This time is consistent with the IPE internal events analysis, NUREG-1335, and the FIVE methodology that states:

   "In the FIVE methodology, a safe and stable condition is that point in reactor shutdown where sub-critical reactivity and reactor coolant inventory temperature and pressure can be maintained at target values for a period of at least 24 hours without damage to the core."
3. In FIVE Phase II, motor control centers (MCCs) and other metal-enclosed components are not considered to be vulnerable to a low-intensity external exposure fire. However, unprotected cables entering and exiting the metal-enclosed component are considered to be vulnerable. Internal cabinet fires are conservatively assumed to disable the entire MCC or cabinet.
4. In FIVE Phase I, all equipment in a compartment is assumed susceptible to fire damage.
5. The delineations and boundaries employed in the Appendix R analyses are used as a starting point in this analysis. They are, however, examined to ensure consistency with the FIVE boundary criteria.
6. It is assumed that offsite power is initially available. Offsite power is not failed as a pre-condition to the analysis. This is consistent with the FIVE methodology which states:

   "FIVE and external event risk assessment are not bound by the initial condition assumptions required when performing Appendix R Safe Shutdown Analysis (i.e., loss of off-site power (LOSP) at the same time as a fire)."
7. Primary containment fires are appropriately not analyzed due to the nitrogen atmosphere inside the drywell during normal operation.

8. Fire effects on reactivity control functions are not modeled. It is assumed that the electrical portion of the reactor scram system fails safe. However, the event tree model includes the base PSA point estimate failure probability for failure to scram.

9. Systems for which cabling has not been tracked and located is assumed to be disabled for all fires in the plant.

10. The quantitative screening threshold used in this analysis to determine the significance of postulated fire induced core damage sequences is consistent with NUREG-1407 and the FIVE methodology (i.e., 1E-6/yr).

11. Fire rated barriers will contain fires up to the listed rating.

12. All cables credited in the PSA models are or are equivalent to IEEE 383 rated cable.

13. Welding units, which are normally not energized, and "frisker" units are not included in the ignition frequency totals.

14. Hot Shutdown is the successful end state used in this analysis. Although proceeding to Cold Shutdown and initiating RHR Shutdown Cooling is an alternative success path in the probabilistic models, it is not required.

4B.1 FIVE PHASE I EVALUATION

The FIVE Phase I analysis of the Auxiliary Building relies heavily on the existing Appendix R documentation, most notably, the Fermi 2 Fire Protection Analysis and the Fermi CECO and cable databases. The Fermi 2 fire zones are the starting point for identifying FIVE fire compartments. A list of safe shutdown systems that could be disabled by a fire was obtained from the Fire Protection Analysis for each of the fire zones. Fire zones that do not contain safe shutdown equipment were examined for the potential for a Fire Initiated Event (FIE). FIVE defines FIE as:

"An event from a fire in any area that (1) results in a demand for safe shutdown functions or (2) damages safe shutdown components of at least one train or shutdown path, unless it can be shown with confidence that the postulated fire will not cause a demand for plant trip or shutdown within 8 hours of the event."

The Phase I effort consists of the following tasks:

- Identify Appendix R safe shutdown systems
- Identify fire areas and associated compartments
- Identify Appendix R safe shutdown equipment in each fire compartment
- Perform fire area vs. Appendix R safe shutdown system screen
- Perform fire area vs. Appendix R safe shutdown function evaluation
- Perform Fire Compartment Interaction Analysis (FCIA)

The Phase I analyses performed for the Fermi 2 Auxiliary Building are described below.

4B.1.1 Identify Appendix R Safe Shutdown Systems

The Fermi 2 Fire Protection Analysis provides the information needed to determine whether Appendix R safe shutdown equipment are present in each of the Auxiliary Building fire zones. The Fermi 2 Fire Protection Analysis lists the 10CFR50, Appendix R safe shutdown systems for Fermi 2 with assigned system codes. These system codes were used in combination with drawing reviews, walkdowns, and CECO database and DECO cable database searches to identify the location of the Appendix R systems in the Auxiliary Building. The Fermi 2 Appendix R safe shutdown systems and associated system codes are listed in Table 4-7.

4B.1.2 Identify Fire Areas and Associated Compartments

The Fermi 2 Fire Protection Analysis fire zones are the starting point for identifying FIVE fire compartments. The Auxiliary Building fire zones, as provided in the Fire Protection Analysis, are listed in Table 4-8. This information is entered into the QUICK FIVE software.

4B.1.3 Identify Safe Shutdown Equipment By Compartment

A database of safe shutdown equipment categorized by fire zone was developed using information obtained from drawing reviews, walkdowns, and CECO database and DECO cable database searches. This database, CONSISE.DBF, is part of the QUICK FIVE set of computer files.

4B.1.4 Perform Fire Area Vs. Safe Shutdown System Screen

This initial screening step allows fire areas that do not contain safe shutdown equipment and do not result in a fire initiated event (FIE) to be screened from further evaluation.

The FIVE methodology makes clear that a fire initiated event should be assumed if there is any doubt:

"Plant operators and/or electrical/systems engineers may need to be consulted to determine the appropriate response for a given fire scenario. . . . If, however, there is any doubt whether the plant would shutdown for a fire in a given fire area, assume the plant would shutdown and do not screen the fire area out in Phase I."

- FIVE (p. 5-5)

For a fire in many areas of the Auxiliary Building a manual scram is relatively assured, as evidenced by the following excerpts from EF2 procedures:

"IF fire is in one of the "3L" zones (4,6,8,9A,11,12,12A,14,16), PERFORM 20.000.18, 'Control of the Plant From the Dedicated Shutdown Panel,' with this procedure."

- AOP 20.000.22, Rev. 23 (p. 2)

"For conditions of fire in any of the identified Zones . . .

6. *If any one of the following symptoms exists, proceed to Step 7.*
   - *Fire Brigade Leader reports damage to plant components such as cables, cable runs, or electrical panels which could render multiple components or systems inoperable.*
   - *Fire Brigade Leader reports fire is out of control and damage to cables, cable runs, or electrical panels which could render multiple components or systems inoperable or imminent*
   - *Control Room is inaccessible as a result of a fire.*
   - *Actual Spurious Operation of Components or Failure of Components occur. . . .*
7. *From the Main Control Room:*
   a. *Scram Reactor . . ."*

- AOP 20.000.18, Rev. 16, (p.1)

The zones listed in the above procedure refer to fire detection zones in the plant. These zones correspond to FHA/FIVE zones 01ABN, 01ABS, 02ABN, 02ABS, 03AB, 07AB, 08AB, 09AB, 11AB, and 13AB, not respectively. If a fire occurs in one of these areas, AOP 20.000.18 is to be performed concurrently. AOP 20.000.18 directs scramming the reactor if a fire in any of the "3L" zones renders multiple components or systems inoperable, or if such damage is imminent.

The above procedures do not explicitly direct scramming the reactor for a fire in other compartments of the Auxiliary Building. However, it is conservatively assumed, consistent with the FIVE methodology, that the reactor will be scrammed if a fire occurs anywhere within the EF2 Auxiliary Building.

As such, no areas of the Auxiliary Building screen out at this step in the analysis.

4B.1.5 Perform Fire Area Vs. Safe Shutdown Function Evaluation

No fire areas are screened at this step. Refer to Section 4B.1.4.

4B.1.6 Fire Compartment Interaction Analysis (FCIA)

Those fire zones that were not screened out in the prior steps (i.e., all fire zones in the Auxiliary Building) were then reviewed for the potential for fire spread (PFS) between compartments; this review is termed Fire Compartment Interaction Analysis (FCIA). Those fire compartments that are determined during the FCIA analysis to meet the FIVE screening criteria for fire boundaries, and:

- do not contain safe shutdown equipment, or
- do not induce a fire initiated event,

are screened from further analysis. Those that remain unscreened are combined or divided, if necessary and as appropriate per the FIVE boundary criteria, to create FIVE fire compartments for analysis in FIVE Phase II.

The fire compartment interaction analysis (FCIA) establishes the adequacy of fire boundaries. The Fermi 2 Fire Protection Analysis provides rating information for the boundaries of each fire zone. Concrete walls are typically rated at 3-hrs. Fire doors have two ratings, as follows:

- Class A: 3 hour
- Class B: 1 1/2 hour

More detailed information on fire boundary ratings was obtained, as necessary, from the following Fermi 2 procedures:

- FPP 28.507.01, "Fire Barrier Inspection"
- FPP 28.507.02, "Fire Door Surveillance Test"
- FPP 28.507.04, "Test and Inspection of Fire Dampers"

These procedures also verified that all boundaries credited in the FIVE analysis for the Fermi 2 Auxiliary Building are included in a surveillance program, as required by the FIVE methodology. The following boundaries in the Auxiliary Building are unrated: 1) walls between fire zones 14AB and 15AB, and 2) floor between fire zones 13AB and 15AB. These two unrated boundaries are treated appropriately in this analysis using FIVE boundary screening criteria.

Another parameter provided in the Fire Protection Analysis is the calculated fire loading in the area. This value is quoted in Btu/sq. Ft. The combination of this fire loading, the boundary rating information, and a plant walkdown were used to establish the FIVE fire compartments. The criteria used to determine whether a particular boundary was adequate with respect to the FIVE methodology are provided in the EPRI FIVE document.

A FIVE Fire Compartment Interaction Analysis (FCIA) form was completed for each of the Auxiliary Building fire zones. Based on this FCIA information, no Auxiliary Building fire zones were screened in Phase I of the FIVE analysis.

The majority of the Auxiliary Building fire zones can be treated directly as FIVE fire compartments in the Phase II analysis. However, based on the FCIA, it was determined that the following fire zones could be subdivided:

- 04AB: The Auxiliary Building Division I switchgear room, fire zone 04AB, was divided into a north fire compartment, 04ABN, and a south fire compartment, 04ABS. Fire compartment 04ABS is a Division II cable chase with fire detection, surrounded by concrete walls and an "A" rated fire door (which are fire rate separation barriers included in the surveillance procedure for this zone), located inside the primarily Division I fire zone 04AB.
- 11AB: The Auxiliary Building miscellaneous rooms on el. 643' 6", fire zone 11AB, were divided into a west fire compartment, 11ABW, and an east fire compartment, 11ABE. Fire compartment 11ABW contains Division II equipment and is separated from the primarily Division I fire compartment 11ABE by a 4" thick concrete wall (rated at 1-1/2 hrs and included in the surveillance procedure for this zone) and an "A" rated fire door.

The two major stairwells in the Auxiliary Building (i.e., the northeast and the southwest stairwells) are devoid of any significant combustible material, verified by the Fire Protection Analysis and a walkdown. The northeast and southwest stairwells are enclosed by 2-hour rated walls and with Class B (1 1/2 hrs) fire doors.**[4.3]** The stairwells are also equipped with hose reels and fire extinguishers. The above information is used to screen the northeast and southwest stairwells as fire propagation paths. For the Phase II analysis, the northeast stairwell is combined with fire compartment 03AB and the southwest stairwell is combined with 04ABN.

4B.1.7 FIVE Phase I Summary

A detailed summary of the Phase I Fire Compartment Interaction Analysis for the Fermi 2 Auxiliary Building is provided in Table 4-9. The critical fire compartments for analysis in FIVE Phase II are provided in the Phase I screening summary in Table 4-10. A total of nineteen (19) Auxiliary Building fire compartments are identified for Phase II analysis.

4B.2 FIVE PHASE II EVALUATION

The FIVE Phase II analysis consists of the following three elements:

- Calculate compartment fire initiation frequency
- Calculate safe shutdown failure probability
- Calculate or evaluate fire propagation, damage, and suppression system effectiveness if required

The analysis for each of these elements is described in the subsections below. To facilitate the analysis process, two principal computer programs were used. ERIN Engineering's QUICK FIVE computer program was used to calculate the compartment fire initiation frequency, and to evaluate fire propagation and target damage likelihood. The RISKMAN computer program was used to calculate safe shutdown system conditional failure probabilities.

4B.2.1 Compartment Fire Ignition Frequency

The ignition frequencies for the Auxiliary Building fire compartments were recalculated using the QUICK FIVE computer program. QUICK FIVE contains the latest base fire ignition frequencies and modified treatment of transient combustible sources from the final FIVE report.**[4.2]** The inputs required by the FIVE Fire Compartment Ignition Source Data Sheet (ISDS) were obtained from the equipment and cable information obtained from drawing reviews, walkdowns, and the CECO and cable databases. An example QUICK FIVE Ignition Source Data Sheet is shown in Table 4-15. The fire ignition frequencies by fire compartment for the Fermi 2 Auxiliary Building are summarized in Table 4-11. No compartments screened at this step in the analysis.

4B.2.2 Safe Shutdown Failure Probability

The calculation of the conditional safe shutdown failure probability given a fire initiating event was performed using the Level 1 Fermi 2 PSA Models. The analysis considered all available shutdown paths modeled in the PSA.

The resulting conditional core damage probability is then multiplied by the fire ignition frequency to obtain the core damage frequency for each compartment. If the screening quantification is less than 1.0E-6 per year, the compartment can be screened from further consideration. If the initial screening quantification is greater than 1.0E-6/yr, deterministic fire modeling is then performed to assess the expected fire scenarios in the compartment.

The FIVE methodology refers to the calculation of P_2, redundant system failure probability, for the determination of core damage frequency screening values. Rather than individual system failure probability calculations, the RISKMAN code was used to calculate the conditional core damage probability (CCDP) given fire induced failures. Refer to Section 4.3.

The initial screening quantifications for the Auxiliary Building were generally intended to be performed assuming that all modeled equipment and cabling in the compartment is damaged due to a postulated fire. However, in many cases this rough approximation could be predicted, prior to quantification, to produce a core damage frequency many orders of magnitude above the FIVE screening criterion of 1E-6/yr. As such, almost all compartments of the Auxiliary Building required a detailed investigation into cable routings and associated circuit functions. Although this detailed investigation was conservative (e.g., when a circuit function was unclear, it was assumed to cause system failure), this investigation did result in deleting a number of obvious non-system failures, that would have otherwise been considered failures, from the initial screening quantification runs.

The initial screening quantifications were coarse screening runs that typically resulted in overly conservative core damage frequencies. In the case of the cable spreading room, the cable tray room and the Control Room Complex (fire compartments 07AB, 08AB, and 09AB, respectively), initial screening quantifications were not performed as large core damage frequencies were expected a priori. The results of the initial screening quantifications are presented in Table 4-12. As can be seen from this table, these initial runs resulted in the screening of just three (3) of the nineteen (19) compartments:

- 10ABE, East Battery Room - Division I
- 10ABW, West Battery Room - Division II
- 15AB, Ventilation Equipment Area

The remaining sixteen (16) compartments required an evaluation of fire ignition, fire growth, target damage and suppression system effectiveness to remove the conservatism inherent in the initial screening analyses. These analyses are discussed below in Section 4B.2.3.

4B.2.3 Fire Damage Modeling and Suppression System Effectiveness

The following Auxiliary Building compartments did not screen during the initial screening quantification runs:

- 01AB, Basement
- 02AB, Mezzanine/Cable Tray Area
- 03AB, Relay Room
- 04ABN, Major Division I Portion of Div. I SWGR Room
- 04ABS, Division II Cable Chase of 04AB
- 05ABE, East Cable Tunnel - Division I
- 05ABW, West Cable Tunnel - Division II
- 06AB, 2nd Floor Miscellaneous Rooms
- 07AB, Cable Spreading Room
- 08AB, Cable Tray Area
- 09AB, Control Room
- 11ABE, Misc. Rooms - Majority of Area (Div. I)
- 11ABW, Misc. Rooms - Div. II Battery Charger Area
- 12AB, Division II SWGR Room
- 13AB, Ventilation Equipment Area
- 14AB, CCHV Equip./SGTS Area

These areas are further analyzed here to remove conservatisms present in the initial screening quantifications. These analyses consider: credibility of ignition sources, frequency of critical combustible loading, proximity of combustible source to target, and fire suppression.

4B.2.3.1 Basement - 01AB

The initial screening quantification of the Auxiliary Building first floor mezzanine and cable tray area resulted in a core damage frequency approximately an order of magnitude higher than the 1E-6/yr FIVE screening criterion. This initial screening quantification is overly conservative. Therefore, this area is analyzed further to consider more realistic fire effects.

Ignition Frequency

A review of the base ignition frequency calculation for this area showed that the contributors to the fire ignition frequency of this area include:

- Two (2) cooling units: 5.7E-5/yr
- Two (2) battery operated lights and four (4) instrument racks: 2.7E-4/yr
- Two (2) air compressors: 9.5E-4/yr
- Transient ignition sources: 2.1E-4/yr
- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr

It is appropriate to exclude the instrument racks and battery operated lights as credible fire scenarios that would result in core damage end states; they are relatively small units and can be excluded from the fire modeling process on the basis of insufficient fire intensity.

The remaining credible ignition sources are the cooling units, the air compressors, hot work, and transient ignition sources.

Fire Damage Scenarios

The potential fire damage scenarios, based on a more realistic review of the ignition sources as described above, include:

- Compressor fire
- Cooling unit fire
- Transient ignited fire
- Cable/transient fire due to hot work

Compressor Fires

This area contains two divisional control air compressors, north compressor P5002D001 and south compressor P5002D002. A control air compressor fire may include an electrical motor fire or a lubricating oil spill fire. The most conservative fire is that associated with an oil spill. As such, the north and south compressor fires are modeled as postulated oil spill and ignition fires. This compartment has an area-wide automatic sprinkler system that, if successfully actuated, will prevent any damage to non-BOP cables from a compressor fire, as discussed below.

The Fermi 2 Fire Protection Analysis indicates that approximately 5 gallons of oil is present in compartment 01AB. It is assumed that 2.5 gallons are contained within each control air compressor. The modeled oil spill fire for each compressor is postulated to include the entire 2.5 gallon oil inventory. The oil is modeled a having the combustion characteristics of transformer oil and the spill characteristics of DTE 797 lubricating oil (these correspond to representative parameters provided in the FIVE methodology).

The postulated spill is treated as a pseudo-confined spill (i.e., no physical barriers exist in the area to confine the spill). A postulated unconfined oil spill would have resulted in a calculated fire with a duration of just 8 seconds. The spill size was adjusted to obtain a fire duration of approximately 1 minute. This corresponds to a peak fire intensity of 5,265 Btu/sec and a spill surface area of 39 sq. ft.

A walkdown of the compartment and a review of arrangement and cable tray drawings determined that the nearest non-BOP cable target (BOP cable targets were not investigated) to the postulated oil spill for the north control air compressor is cable tray 1C-018 at el. 573' 4". This cable tray passes approximately 22 feet directly above the fire source. The automatic sprinkler heads in the area are equipped with bulb type elements and have a temperature rating of 165°F. The FIVE methodology recommends an actuation delay time in the range of 120-240 seconds for bulb type heads. The fire modeling analysis assumes a conservative value of 240 seconds for actuation delay. The nearest sprinkler head is approximately 5 feet away laterally and approximately 10 feet above the fire source. Using the QUICK FIVE fire modeling worksheets, it was determined that suppression system actuation would occur in approximately 28 seconds, whereas target damage would begin to occur in 85 seconds. Therefore, successful actuation should be credited as being effective in preventing damage to all non-BOP equipment.

The nearest non-BOP cable target to the postulated oil spill for the south control air compressor are conduits, approximately 2 feet away, containing RCIC cabling. Therefore, the RCIC system should be assumed failed. However, the next closest cable target is cable tray 2K-011 at el. 577' 6". This cable tray passes approximately 3 feet laterally and 26 feet above the fire source. The nearest sprinkler head, like the north compressor, is approximately 5 feet away laterally and approximately 10 feet above the fire source. Using the QUICK FIVE fire modeling worksheets, it was determined that suppression system actuation would occur in approximately 28 seconds, whereas target damage would begin to occur in 136 seconds. Therefore, successful actuation should be credited as being effective in preventing target damage to all non-BOP equipment, except for RCIC.

Cooling Unit Fires

This area contains two control air compressor cooling units, north cooling unit T4100B029 and south cooling unit T4100B030. The ignition source component of these cooling units is a motor-driven grease lubricated fan. The fan motors are approximately 4' 6" above the floor.

A walkdown of the compartment and a review of arrangement and cable tray drawings determined that the nearest non-BOP cable target (BOP cable targets were not investigated) to the fan motor of the north cooling unit is cable tray 1K-022 at el. 559' 6". This cable tray passes approximately 4.5 feet laterally and 4 feet above the fan motor (a line of site distance of slightly over 6 feet). As the fan motor fire would be a motor windings fire inside the motor housing, the fan motor fire is modeled using the FIVE methodology radiant exposure approach. As the FIVE methodology does not provide heat release rates for motors, a bounding approach was used to determine if target damage is likely. The spacial and other fire modeling information was input into the QUICK FIVE automated radiant exposure worksheet, and then the peak fire intensity was adjusted until the critical radiant flux distance equaled 6 feet (the line of site distance from the motor to 1K-022). The result was a peak fire intensity of 1130 Btu/sec. This is an overly conservative heat release rate. It is triple that of a 32-gallon trash container fire, as provided in the FIVE methodology. Therefore, fire modeling does not indicate that damage will occur to any non-BOP cabling due to a fan motor fire in the north cooling unit.

The nearest non-BOP cable target to the fan motor of the south cooling unit is a Division II cable tray approximately 6 feet away (line of sight distance). Therefore, the conclusion for the north cooling unit fan applies to the south cooling unit fan.

Transient Ignited Fires

A walkdown of the area did not identify any transient ignition sources or combustible materials. However, to bound the transient combustible fire scenario, a trash container fire is conservatively postulated in the room. This fire was modeled with the QUICK FIVE software as having a duration of 5 minutes and a peak fire intensity of 380 Btu/sec.; this is consistent with fire modeling examples in the FIVE methodology. The height of the point source trash container fire was assumed to be 4 feet above the floor, consistent with the FIVE methodology.

The critical height for the Target-In-Plume case is approximately 17 feet above the floor if the container is located in a corner (14 feet if against a wall and 12 feet if in center of room), and the critical radiant flux distance is approximately 3.4 feet. As this area contains cable trays lower than 17 feet above the floor, the analysis of this area needs to address the likelihood of a transient combustible source placed under horizontal cable trays or near vertical cable trays and the possibility of a resulting fire.

Hot Work Fires

Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets or transient combustibles to cause ignition.

Accident Sequence Quantification

Compressor Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to fixed ignition sources:

$F_{fx} = F_{if} \times P_{fs} \times CCDP$, where:

F_{fx}	=	core damage frequency (due to fixed sources)
F_{if}	=	fire ignition source frequency (due to fixed sources)
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

Per Attachment 10.3 of the FIVE methodology, the failure probability of an automatic wet pipe sprinkler system is 2.0E-2.

A bounding conditional core damage probability, given failure of the suppression system, can be conservatively estimated using the more conservative of the two initial screening quantification runs (F1AB02, CCDP=3.9E-3 - see Table 4-4).

Therefore, the core damage frequency for this area due to a compressor fire is conservatively estimated as follows:

F_{fx} = 9.5E-4 x 2.0E-2 x 3.9E-3 = 7.41E-8/yr

Cooling Unit Fires

The fire modeling performed for the north and south cooling unit fans, as discussed earlier, does not indicate damage to any nearby non-BOP cabling or equipment. However, the effect on BOP cabling was not evaluated. Therefore, the core damage frequency estimate for this ignition contributor is conservatively modeled as leading to a plant trip with loss of the main condenser and loss of the hard-piped vent. No credit is taken for fire suppression. The conditional core damage probability associated with this damage is conservatively estimated at 2E-4. This CCDP is conservatively based on an existing run, 14AB2, that includes failure of the main condenser, the hard-piped vent and RCIC.

Therefore, the core damage frequency for this area due to a cooling unit fan is conservatively estimated as follows:

5.7E-5 x 1.0 x 2.0E-4 = 1.14E-8/yr

Transient Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to transient combustible sources:

$F_t = F_{it} \times u \times p \times w \times P_{fs} \times CCDP$, where:

F_t	=	core damage frequency (due to transients)
F_{it}	=	fire ignition source frequency (due to transients)
u	=	transient combustible located in the range of target components
p	=	probability of combustible being exposed
w	=	frequency of finding a critical combustible loading versus the frequency of inspecting and removing the transient combustible fire source.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters u, P_{fs} and p are conservatively set to 1.0.

Per the FIVE methodology, w is calculated as follows:

$w = (x/2) \times \ln(1/x)$, where $x = F_{ccl}/F_w$

F_{ccl}	=	The frequency of having a critical combustible loading present. FIVE provides a screening value of 1 event/year/compartment.

F_w	=	Frequency of combustible material inspections that would find the transient combustible fire source before a fire occurred.

The value of 1 event/year is used here for F_{ccl}.

Administrative procedure NPP-FP1-01 specifies that: 1) monthly fire protection inspections are to be performed, and 2) an hourly compensatory fire watch is to tour all areas appearing on the Out of Specification Log Sheet. This same procedure states that the accumulation of trash, oil rags, combustible materials, and similar fire hazards is prohibited. It is judged that either of the above tours would identify and correct a trash container inappropriately placed near a cable tray riser. As the hourly compensatory fire watch may not tour this area hourly or every day, a conservative estimate of one tour per week is assumed (F_w = 52/yr). Therefore, w for this area is estimated as follows:

$w = ((1/52)/2) \times \ln(1/(1/52)) = 3.79E\text{-}2$

A bounding conditional core damage probability, given failure of the suppression system, can be conservatively estimated using the more conservative of the two initial screening quantification runs (F1AB02, CCDP=3.9E-3 - see Table 4-4).

Therefore, the core damage frequency for this area due to transient combustible fires is conservatively estimated as follows:

$F_t = 2.1E\text{-}4 \times 1.0 \times 1.0 \times 3.79E\text{-}2 \times 1.0 \times 3.9E\text{-}3 = 3.10E\text{-}8/yr$

Hot Work Fires

Hot work induced core damage frequency is estimated as follows:

$F_{hw} = F_{ihw} \times u \times HEP_{hw} \times P_{fs} \times CCDP$, where:

F_{hw}	=	core damage frequency (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.
u	=	probability of hot work performed in close enough proximity to important cable targets or transient combustibles such that the possibility exists for ignition
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters P_{fs} and u are conservatively set to 1.0.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

A bounding conditional core damage probability of the above fire can be conservatively estimated using the more conservative of the two initial screening quantification runs (F1AB02, CCDP=3.9E-3 - see Table 4-4.)

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

F_{hw} = 8.4E-4 x 1.0 x 5.0E-2 x 1.0 x 3.9E-3 = 1.64E-7/yr

Summary

Summing the individual fire scenario core damage frequencies, the total core damage frequency due to fire in this area is conservatively estimated at 2.80E-7/yr. This estimate is below the 1E-6/yr FIVE screening criterion. Therefore, this area is screened from further analysis.

4B.2.3.2 Mezzanine/Cable Tray Area - 02AB

The initial screening quantification of the Auxiliary Building first floor mezzanine and cable tray area resulted in a core damage frequency over an order of magnitude higher than the 1E-6/yr FIVE screening criterion. This initial screening quantification is overly conservative.

A review of the base ignition frequency calculation for this area showed that the fixed ignition sources in this area include:

- Two (2) fire protection panels
- One (1) battery operated light

A credible fire in either of the panels or the emergency light would be small and confined to within the boundaries of the panel. As such, a postulated fire due to the fixed ignition sources would not result in damage beyond failure of an emergency light or a fire protection panel. In addition, these postulated fires would quickly extinguish due to limited combustibles and would realistically not result in a plant trip.

Therefore, the only credible fire ignition sources are transient sources and ignition sources due to hot work (e.g., open flames, welding, grinding, or arc techniques). However, Administrative Procedure NPP-FP1-01 places restrictions on transient ignition sources. In addition, NPP-FP1-01 prescribes certain controls and requirements be satisfied prior to the start of hot work activities. These requirements involve the consideration of combustible or flammable material, protection of combustibles from ignition sources, and the establishment of fire protection measures. NPP-FP1-01 requires the establishment of fire watches during and for at least 30 minutes after the completion of any hot work.

The combination of the lack of significant fixed ignition sources and the fire ignition control measures are considered adequate to preclude a credible fire event from damaging cables in this compartment. However, in order to develop a fire risk screening value, a conservative analysis is presented below.

Fire Ignition Frequency

Based on the discussion above, the contributors to the area fire ignition frequency are as follows:

- Two (2) fire protection panels: 2.4E-5/yr
- One (1) emergency light: 4.3E-5/yr

- Transient ignition sources: 2.1E-4/yr
- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr

Based on the discussion above, it is appropriate to exclude the fire protection panels and emergency light as credible fire scenarios that would result in core damage end states.

In keeping with the conservative nature of the analysis, no adjustment is made to more realistically characterize the frequency of transient or hot work ignition sources in this area.

Fire Damage Scenarios

As stated above, the only credible fires are those due to transient combustible fire sources and hot work (i.e., there are no credible fixed fire sources).

Transient Fires

A walkdown of the area did not identify any transient ignition sources or combustible materials. However, to bound the transient combustible fire scenario, a trash container fire is conservatively postulated in the room. This trash container fire is postulated for only the north and south (el. 583' 6") ends of the area; the center mezzanine area (el. 603' 6") is not a location where a trash container would be located because this area is accessible only by a ladder at the south end and by a locked and controlled security door at the north end. This fire was modeled with the QUICK FIVE software as having a duration of 5 minutes and a heat rate of 380 Btu/sec.; this is consistent with fire modeling examples in the FIVE methodology. The height of the point source trash container was assumed to be 4 feet above the floor, consistent with the FIVE methodology.

The Target-In-Plume and Radiant Exposure worksheets indicate that if the trash container was located in the center of either end of the area that cable damage would not occur. The critical radiant flux distance is approximately 3.4 feet; vertical trays are farther away, typically at the walls. The critical height for the Target-In-Plume case for a trash container placed in a corner is approximately 17 feet above the floor, whereas, the height above the floor of all overhead horizontal cable trays is approximately 20 feet or more. However, if the trash container was located next to vertical raceways, critical conditions could exist which would damage cabling. Therefore, the analysis of this area needs to address the likelihood of a transient combustible source placed near vertical cable trays and the possibility of a resulting fire.

Hot Work Fires

Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets or transient combustibles to cause ignition. No deterministic fire modeling is performed to define the conditions necessary for accidental hot work induced cable ignition. Rather, the critical distance is based on the trash container fire modeling results discussed above.

Accident Sequence Quantification

Transient Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to transient combustible sources (note that fixed combustible fire sources are not included in this formula, for the reasons discussed above):

$F_t = F_{it} \times u \times p \times w \times P_{fs} \times CCDP$, where:

F_t	=	core damage frequency (due to transients)
F_{it}	=	fire ignition source frequency (due to transients)
u	=	transient combustible located in the range of target components
p	=	probability of combustible being exposed
w	=	frequency of finding a critical combustible loading versus the frequency of inspecting and removing the transient combustible fire source.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters P_{fs} and p are conservatively set to 1.0.

Per the FIVE methodology, u is calculated as follows:

$u = (A_s + A_{sr})$ / Net Area, where:

A_s	=	Exposed surface area of targets facing floor (e.g., width x length of horizontal cable trays)
A_{sr}	=	Area around radiant target determined by the critical separation distance (determined to be 3.4 ft., as described above)

A review of cable tray drawings shows that an approximate value of u is 0.25. The range of estimates is approximately 0.2 (only Appendix R trays considered) to 0.4 (all trays in room considered). This estimate employed conservative estimates for A_s and A_{sr} to simplify the analysis.

Per the FIVE methodology, w is calculated as follows:

$w = (x/2) \times \ln(1/x)$, where $x = F_{ccl}/F_w$

F_{ccl}	=	The frequency of having a critical combustible loading present. FIVE provides a screening value of 1 event/year/compartment.
F_w	=	Frequency of combustible material inspections that would find the transient combustible fire source before a fire occurred.

The value of 1 event/year is used here for F_{ccl}.

Administrative procedure NPP-FP1-01 specifies that: 1) monthly fire protection inspections are to be performed, and 2) an hourly compensatory fire watch is to tour all areas appearing on the Out of Specification Log Sheet. This same procedure states that the accumulation of trash, oil rags, combustible materials, and similar fire hazards is prohibited. It is judged that either of the above tours would identify and correct a trash container inappropriately placed near a cable tray riser. As the hourly compensatory fire watch may not tour this area hourly or every day, a conservative estimate of one tour per week is assumed (F_w = 52/yr). Therefore, w for this area is estimated as follows:

$w = ((1/52)/2) \times \ln(1/(1/52)) = 3.79E\text{-}2$

A bounding conditional core damage probability of the above fire can be conservatively estimated using the more conservative of the two initial screening quantification runs (2AB01, CCDP=3.5E-2 - see Table 4-4). This is a bounding estimate as it includes fire damage to many cable trays.

Therefore, the core damage frequency for this area due to transient combustible fires is conservatively estimated as follows:

$F_t = 2.1E\text{-}4 \times 0.25 \times 1.0 \times 3.79E\text{-}2 \times 1.0 \times 3.5E\text{-}2 = 6.96E\text{-}8/yr$

Hot Work Fires

Hot work induced core damage frequency is estimated as follows:

$F_{hw} = F_{ihw} \times u \times HEP_{hw} \times P_{fs} \times CCDP$, where:

F_{hw}	=	core damage frequency (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.
u	=	probability of hot work performed in close enough proximity to important cable targets or transient combustibles such that the possibility exists for ignition
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameter P_{fs} is conservatively set to 1.0.

The parameter u is estimated with a value of 0.25. This value is the same as that used in the transient combustible case and is based on a 5 min. trash container fire with a heat rate of 380 Btu/sec.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

A bounding conditional core damage probability of the above fire can be conservatively estimated using the more conservative of the two initial screening quantification runs (2AB01, CCDP=3.5E-2 - see Table 4-4). This is a bounding estimate as it includes fire damage to many cable trays.

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

$F_{hw} = 8.4E\text{-}4 \times 0.25 \times 5.0E\text{-}2 \times 1.0 \times 3.5E\text{-}2 = 3.68E\text{-}7/yr$

Summary

Summing F_t and F_{hw}, the total core damage frequency due to fire in this area is conservatively estimated at 4.37E-7/yr. This estimate is below the 1E-6/yr FIVE screening criterion. Therefore, this area is screened from further analysis.

4B.2.3.3 Relay Room - 03AB

The initial screening quantification for the Relay Room involved numerous individual panel fire scenarios. The resulting sum of the individual core damage frequencies exceeds the 1E-6/yr FIVE screening criterion by more than an order of magnitude. The following analysis is performed to provide a more realistic characterization of the risk.

The initial screening quantification involved a review of the DECO cable database for each of the cabinets in 3AB included in the Appendix R program. The impacts due to a fire in each cabinet were defined accordingly based on this review. The remaining cabinets were classified into one of two categories:

- Plant trip with main condenser available
- plant trip with main condenser unavailable

All cabinet fire scenario core damage frequencies were summed together to obtain the total core damage frequency for the compartment. Transient fires are encompassed with this approach by dividing the total compartment fire ignition frequency (which includes the transient fires contribution) by the total number of panels in the room, and then using this panel fire ignition frequency for each fire scenario.

However, the following panels in the relay room contain only cable terminations (i.e., termination cabinets):

H11P820	H11P821
H11P822	H11P823
H11P837	H11P838
H11P839	H11P840
H11P853	H11P855
H11P856	H11P861
H11P862	H11P866
H11P868	H11P869
H11P877	H11P878
H11P879	H11P880
H11P888	H11P889
H11P891	H11P854

Each of the above panels was analyzed in the initial screening quantification as a panel fire that resulted in a plant trip and damage to a wide array of equipment. However, such a characterization is overly conservative. These panels contain no more than bolted cable terminations (i.e., no circuit breakers, transformers, relays, etc.) and are not considered credible fire scenarios. Therefore, the summed core damage contribution of these panels, 4.56E-6/yr, should be deleted from the compartment core damage frequency.

In addition, the initial screening quantification analyzed 30 fire protection panels and 66 other miscellaneous panels as causing a manual scram but not damaging any equipment necessary for shutdown. The assumption of a manual scram for a miscellaneous panel fire is conservative. A credible fire in these types of panels would be small and confined to within the boundaries of the panel. In addition, these fires would quickly self-extinguish due to limited combustibles and would realistically not result in a plant trip. Therefore, the summed core damage contribution of these panels, 1.46E-7/yr (refer to Table 4-12), should be deleted from the compartment core damage frequency.

Therefore, the resulting core damage frequency for this area is estimated by subtracting 4.56E-6/yr and 1.46E-7/yr from the initial screening estimate of 7.48E-6/yr. The result is a core damage frequency of 2.77E-6/yr that is above the FIVE screening criterion of 1E-6/yr. The remaining fire scenarios are considered credible. The automatic Halon system does not provide any mitigation, as the system is not judged to actuate, if at all, until the panel fire has damaged the panel.

The core damage estimate of 2.77E-6/yr may still be conservative as the system and equipment impacts for the individual panel fires were identified conservatively. A cable To-And-From database was used to identify system circuits contained within many of the panels. Detailed electrical circuit reviews may provide a more realistic characterization of system and equipment impacts, and reduce compartment core damage frequency.

4B.2.3.4 Major Division I Portion of Div. I SWGR Room - 04ABN

The initial screening quantification of this compartment resulted in a core damage frequency over an order of magnitude higher than the 1E-6/yr FIVE screening criterion. This initial screening quantification is overly conservative.

Fire Ignition Frequency

A review of the base ignition frequency calculation for this area shows the following contributors to the fire ignition frequency:

- Eleven (11) fire protection panels: 1.3E-4/yr
- Ten (10) emergency lights: 1.4E-4/yr
- Seven (7) cooling unit components: 2.0E-4/yr
- 480V 72C bus & 4kV/480V transformer: 3.0E-4/yr
- 480V 72B bus & 4kV/480V transformer: 3.0E-4/yr
- 4160V 64C bus: 1.0E-4/yr
- 4160V 64B bus: 1.4E-4/yr
- 64T crosstie breaker cabinet: 1.4E-5/yr
- MPU 1 and integral voltage regulator: 4.9E-4/yr

- MPU 3 and integral voltage regulator: 4.9E-4/yr
- MCC 72B-2A: 2.6E-4/yr
- Cabinet 72C-2D: 1.7E-4/yr
- 480V-120V transformer/distribution cabinet 72C-2D-1: 9.7E-5/yr
- 130V DC cabinet 2PA2-14: 1.9E-4/yr
- Bus 64C local control panel: 1.4E-5/yr
- MCC 72B-2A local control panel: 1.4E-5/yr
- MCC 72C-F isolating contactor (Div. I): 1.4E-5/yr
- Miscellaneous electrical items: 9.8E-5/yr
- Transient ignition sources: 2.1E-4/yr
- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr

As discussed earlier, the fire protection panels and the emergency lights are eliminated from further analysis. The credible fires in either the fire protection panels or the emergency lights would be confined to within the boundaries of the unit and would realistically not result in a plant trip.

The remaining ignition sources are maintained for further analysis.

Fire Damage Scenarios

The potential fire damage scenarios, based on review of the ignition sources, as described above, include:

- Cooling unit fire
- Bus fire
- MCC/cabinet/panel fire
- Miscellaneous electrical fire
- Transient ignited fire
- Cable/transient fire due to hot work

Each of the credible fire scenarios is discussed below.

__Cooling Unit Fire__

Two cooling unit systems are located in this area, east cooling unit T4100B002 and west cooling unit T4100B003. However, each is located far from safety related cable trays and equipment. The primary ignition source for each cooling unit is a motor windings fire. Each motor of the cooling units is located 5 feet (line of sight distance) or more from the nearest divisional cable tray. As the motor fire would be a windings fire inside the housing, the fan motor is best modeled using the FIVE methodology radiant exposure case. An overly conservative peak fire intensity (approximately 400 Btu/sec.) would be required to create a critical flux distance of five feet. Therefore, deterministic fire modeling does not indicate that damage will occur to any safety equipment or cables due to a cooling unit fire.

However, the motor on the west cooling unit is located approximately 2 feet below BOP cable tray 0C-118. A motor windings fire can be expected to cause failure of the cables in this tray. Due to the small fire source represented by the motor, ignition of the cables and subsequent fire propagation to other cable trays is not expected. Cable tray 0C-118 does not contain any cables supporting equipment modeled in the PSA. Therefore, the cooling unit fire scenarios are modeled as either cooling unit leading to a plant trip with availability of all modeled equipment.

480V 72C Bus/Transformer Fire

A postulated fire would result in loss of power at this bus. An additional concern involves the potential for this fire to propagate to cable trays 0P-409, 0P-434, and 0P-530 which are located approximately 2 feet above the switchgear breaker section, and approximately 18" above the transformer section. Propagation to the Division 0 trays would result in a fire that could damage the Division I raceways at the next higher level in the tray rack and result in disabling all Division I power supplies. In addition, fire damage to the cables in the overhead Division 0 cable trays results in MSIV closure due to loss of the Division I RPS power supply and DC Panel 2PA2-6.

The FIVE methodology provides no guidance on the heat release rate or damage thresholds for switchgear and buses. However, the FIVE methodology cites SANDIA tests on control board panels when referring the reader to guidance on electrical panel heat release rates. These SANDIA tests show peak fire intensities in the approximate range of 800-1200 Btu/sec. The FIVE methodology neither recommends nor requires these heat release rates, nor does the methodology provide guidance on the relationship of these values and the applicability of these test results to switchgear equipment. If the low end of this range is used in a Fire-In-Plume worksheet, the cables above reach temperatures that would cause cable insulation ignition.

However, the EPRI Fire PRA Implementation Guide **[4.32]**, which was published to "determine realistic fire risk", indicates that the above heat release rates are "high". The EPRI Fire PRA Implementation Guide recommends a heat release rate of 65 Btu/sec. for vertical cabinets containing only qualified cable. However, the EPRI Guide also does not provide recommendations on the use of these heat release rates when analyzing a bus.

Given the lack of specific guidance in the FIVE methodology and the associated EPRI Fire PRA Implementation Guide, an investigation into the construction of the switchgear units was performed to aid in the realistic fire modeling of this equipment. A diagram showing the general structure of the Fermi 2 switchgear units is provided in Figure 4-2. The switchgear unit shown in Figure 4-2 is a 4.16kV unit but the general construction is representative of 480V switchgear units, as well. The switchgear is composed of discrete steel compartments to protect against both personnel and fire hazards. As can be seen from Figure 4-2, the switchgear portion of bus 72C is composed of the following completely segregated areas:

- instrument compartment (front top)
- circuit breaker compartment (front middle & bottom)
- current transformer compartment (center bottom)
- bus compartment (center)

- cable compartment (back bottom)

The Fermi 2 480V switchgear are a front-and-back construction. The general construction is similar to that shown in Figure 4-2, with an additional "front" (i.e., breaker and instrument compartments) sandwiched to the back side. This construction results in the location of the bus, cable and current transformer compartments in the lower center interior of the unit, with breaker and instrument compartments on the front and back sides.

Given the above information, the Fermi 2 480V switchgear buses are characterized by the following:

- internal design is composed of segregated steel compartments
- breaker cubicles and instrument cubicles are separated from one another and from the bus and cable compartments
- cable and bus compartments are located toward the bottom

In addition to the switchgear portion, the 72C switchgear bus is joined with a 4kV/480V step-down transformer. The transformer portion of the 72C switchgear bus is located at the east end of the 72C unit. Figure 4-3 provides a depiction of the switchgear bus and associated transformer.

The credible and most significant fires in the 72C switchgear bus unit are judged to initiate in the breaker cubicles or the 4kV/480V transformer. The cable and bus compartments are judged not to be sources of credible switchgear fires. Discussion with a switchgear expert from ABB confirmed that fires in the bus or cable compartments are not likely.**[4.31]** These compartments contain primarily steel and glass materials and passive equipment connected with bolted joints. The current transformer compartment (used for metering and instrumentation) represents a credible fire ignition location; however, the location of this steel compartment in the interior bottom of the switchgear unit does not warrant its treatment as a separate fire scenario. The current transformer compartment fire scenario is bounded by the breaker cubicle fire scenario. Therefore, the postulated 72C switchgear bus fire is analyzed here as initiating in one of the following two locations:

- 4kV/480V step-down transformer
- circuit breaker cubicle

Given the lack of specific fire modeling guidance in the FIVE methodology, a postulated fire at bus 72C was evaluated with three fire scenarios:

- <u>Scenario 1</u>: bounding worst case fire
- <u>Scenario 2</u>: 4kV/480V transformer fire
- <u>Scenario 3</u>: circuit breaker cubicle fire

The first scenario involves a bounding worst case fire with a very high heat release rate. Such a fire is considered to be very unlikely. However, it is conservatively assumed that 10% of the postulated 72C switchgear bus fires are very large fires that propagate beyond the boundaries of the switchgear enclosure and damage the raceways located above. This worst case fire scenario makes no distinction as to the source of the ignition (i.e., in a breaker cubicle or the transformer); rather, the total ignition frequency of the 72C switchgear bus and transformer, calculated per the FIVE methodology, is used. In addition, no deterministic calculations are made regarding fire intensity; rather, the fire is guaranteed in the analysis to fail the bus and all cable trays located above.

The second and third fire scenarios are initiated by (1) a postulated transformer fire and (2) a postulated breaker cubicle fire. These two fire scenarios are modeled more realistically, using the EPRI Fire PRA Implementation Guide suggested cabinet fire heat release rate for the breaker cubicle fire and a heat release rate for the 4kV/480V transformer based on its nominal rating. These two fire scenarios are assumed to comprise the remaining 90% of the total postulated fire scenarios initiated in the 72C bus.

A transformer fire has the potential to represent a significant threat. However, available transformer protective devices are available that would isolate power to the transformer given a major upset condition. Therefore, the most credible scenario involves a relatively low intensity fire that results in loss of the entire bus. Large transformer fires are encompassed by the worst case fire scenarios (i.e., Scenario 1).

The available industry guidance does not provide any specific recommendations for fire modeling dry transformers. A conservative fire intensity was selected based on the power rating of the transformer. Twenty-five percent (25%) of the 1500 KVA nominal rating of the 4kV/480V transformer was selected. This corresponds to a fire intensity of approximately 375 Btu/sec.

The spacing between the transformer coils inside the transformer cabinet and the nearest overhead cable trays (i.e., 0P-409, 0P-434, and 0P-530 which comprise a single elevation run overhead of the 480V switchgear) is 4 feet. The distance between the top of the transformer coils and the top of the transformer cabinet is approximately 2.5 feet. A additional 18" of spacing exists between the top of the transformer cabinet and the overhead cable trays.

Inputting these parameters into the FIVE Radiant Exposure Worksheet yields a critical radiant flux distance of 3.45 ft. This is less than the actual spacing of 4 feet between the transformer coils and the overhead trays, and shows that no damage to the overhead cable trays is expected. This result is conservative because the shielding provided by the steel top of the transformer cabinet is not credited. Accounting for the shielding of the top of the transformer cabinet would produce a shorter critical distance. The methodology presented in Appendix G of the EPRI Fire PRA Implementation Guide allows an adjustment to the critical radiant flux to account for shielding, as follows:

$$T = 0.81q^{0.55}$$

where: T = air temperature rise on opposite side of shielding, °C

q = heat flux at shielding, W/m^2

Application of this adjustment is not performed here, as damage to the above cable trays is not indicated, as discussed above, even without consideration of shielding provided by the top of the transformer cabinet.

An addition, the calculated critical flux distance of 3.45 feet would not result in damage to nearby equipment. The closest equipment of significance to the 72C 4kV/480V transformer is the Bus 64C local control panel located approximately 3 feet in front of the transformer cabinet. The distance of the fire source, the transformer coils inside the transformer cabinet, to the Bus 64C local control panel is approximately 3.5 feet - indicating no damage would occur to this panel from the postulated transformer fire.

Therefore, it is concluded that a postulated fire of the 72C 4kV/480V transformer would result in only loss of power at the 72C bus. No other fire induced failures are indicated by the fire modeling.

The other potential fire scenario involves a postulated fire within a breaker cubicle. Due to the steel cubicle construction of the circuit breaker compartment, flames and hot gases are expected to be contained in and attenuated by the individual steel breaker cubicle in which the fire is postulated. This postulated fire also has the potential to create a radiant exposure case that could affect targets located immediately above the switchgear. The breaker cubicle fire is modeled with a heat rate of 65 Btu/sec and is conservatively placed at the top of the switchgear unit. The 65 Btu/sec heat rate is recommended by the EPRI FIRE PRA Implementation Guide for electrical cabinet fires. The critical radiant flux distance obtained from the FIVE Radiant Exposure Worksheet is 1.44 ft. As the overhead cable trays are approximately 2 feet above the top of the switchgear, damage to the overhead cable trays is not indicated. Margin in excess of 0.56 feet actually exists, as the above calculation, conservatively does not credit the thermal shielding provided by the steel top of the breaker cubicle.

A postulated breaker cubicle fire is shown above to result only in damage internal to the switchgear. Such damage can be conservatively modeled as loss of power on the entire bus. Therefore, a postulated switchgear transformer or breaker cubicle fire can be modeled as loss of the entire bus. These two fire scenarios are conservatively judged, as discussed above, to represent approximately 90% of the total number of 72C fire scenarios. The remaining 10% can be modeled as large fires that disable all functions in the room.

480V 72B Bus/Transformer Fire

The discussion provided above for 480V Bus 72C applies equally to 480V Bus 72B. Cable trays pass overhead approximately 2 feet above the top of the unit. No other significant equipment is located closer than approximately 4 feet. The credible fires are judged to initiate in the 4kV/480V transformer or a breaker cubicle. Ninety percent (90%) of the spectrum of 72B fires are modeled as breaker and transformer fires, resulting in loss of the bus. Ten percent (10%) of the postulated 72B fires are modeled as large fires that disable all functions in the room.

4160V 64C Bus Fire

The discussion provided above for 480V Bus 72C applies equally to 4160V Bus 64C. Cable trays pass overhead approximately 2 feet above the top of the unit. No other significant equipment is located closer than approximately 4 feet. A minor exception to the Bus 72C configuration, is that this switchgear bus unit does not contain a large transformer component. The postulated fires are assumed to occur in the breaker cubicles. Each cubicle is analyzed as an electrical panel. Ninety percent (90%) of the postulated 64C fires are modeled as resulting in damage internal to the switchgear bus unit. If a fire occurs in a cubicle associated with the offsite power feed to the 64C bus then, in addition to loss of the 64C bus, the offsite feed will be lost to both the 64C and the 64B bus. However, the 64B bus can still be powered by the emergency diesel generators (the 64C bus is the source of the fire and is assumed completely disabled). If a fire occurs in a cubicle not associated with the offsite power feed, then only power on the 64C bus is lost (i.e., the offsite feed to Bus 64B is still available). The remaining ten percent (10%) of the postulated 64C fires are modeled as large fires that disable all functions in the room.

4160V 64B Bus Fire

The discussion provided above for 4160V Bus 64C applies equally to 4160V Bus 64B. Cable trays pass overhead approximately 2 feet above the top of the unit. No other significant equipment is located closer than approximately 4 feet. The credible fires are judged to initiate in the breaker cubicles. Ninety percent (90%) of the postulated 64B fires are modeled as resulting in damage internal to the switchgear bus unit. If a fire occurs in a cubicle associated with the offsite power feed to the 64B bus then, in addition to loss of the 64B bus, the offsite feed will be lost to both the 64B and the 64C bus. However, the 64C bus can still be powered by the emergency diesel generators (the 64B bus is the source of the fire and is assumed completely disabled). If a fire occurs in a cubicle not associated with the offsite power feed, then only power on the 64B bus is lost (i.e., the offsite feed to Bus 64C is still available). The remaining ten percent (10%) of the postulated 64B fires are modeled as large fires that disable all functions in the room.

64T Breaker Cabinet Fire

The 64T breaker cabinet is a steel switchgear unit located between the 64B and 64C switchgears. The offsite transformer feeds enter the top of the switchgear unit via enclosed, louvered, metal ducts. Cable trays 1C-133 and 1P-078 pass overhead of this unit at heights of 18" and 3 1/2', respectively. Due to the similar construction, this unit is evaluated in a similar manner as the 64B & C and 72B & C switchgear units. Ninety percent (90%) of the postulated 64T fires are modeled as resulting in damage internal to the unit. Like the breaker cubicle fires of the 64 and 72 buses, using the electrical cabinet fire heat rate suggested by the EPRI FIRE PRA Implementation Guide indicates that damage will not occur to the overhead trays or any other nearby equipment. A fire in this unit can be expected to result in loss of the 64 transformer feed to both 4kV buses 64B and 64C. The remaining ten percent (10%) of the postulated 64T fires are modeled as large fires that disable all functions in the room.

MPU 1 Fire

A postulated fire of Modular Power Unit MPU 1 would result in loss of power to the loads supplied by MPU 1. Review of the loads supplied by MPU 1 shows that the only significant effect would be MSIV closure due to loss of control power to the drywell pneumatic supply valves. In addition, a fire in MPU 1 could be postulated to damage the cables in cable tray 1C-094 which runs overhead. The postulated failure of the circuits in 1C-094 would result in failures equivalent to loss of 4 KV buses 64B and 64C.

The spacial arrangement of this compartment is such that MPU 1 is located near the west wall of the room. Cable tray 0C-118 is located approximately 18 inches directly above the panel. Although tray 0C-118 contains no cables supporting functions modeled in the fire PSA model, this cable tray represents an intervening combustible. The critical target is cable tray 1C-094 that is located approximately 2 feet above 0C-118. Ignition of the cables in tray 0C-118 could result in damaging the cables above in tray 1C-094.

Similar to the 480V and 4kV switchgear units, the available industry guidance does not provide any specific recommendations for fire modeling electrical components such as MPU 1. Therefore, an investigation into the construction of MPU 1 was performed to aid in the realistic fire modeling of this equipment. MPU 1 is a power distribution panel consisting of three main compartments. The top compartment has louvered sides and contains cable terminations and automatic transfer switching. The middle compartment contains fused disconnect switches for power distribution. The lower compartment is louvered and contains two regulating transformers.

The dominant credible fire scenario of MPU 1 is expected to involve a transformer fire. Fires involving the fused disconnect switches or the upper compartment switching mechanisms are judged less likely and are expected to have a low heat rate. A postulated transformer fire is expected to be characterized by significant smoke generation and a heat rate similar to that of a control panel fire. A large transformer fire is not expected to occur since it would require a concurrent failure of overcurrent protective devices.

The design of the MPU is such that cable penetrations exist at the panel top. These penetrations are assumed to be unsealed. Cables routed through these penetrations are terminated at either the fused disconnect switches or the transfer switching. Internal panel wiring provides the connection from these locations to the postulated transformer fire source at the bottom of the panel. A postulated transformer fire can be expected to disable some or all of the internal panel wiring. However, due to the cubicle construction of the unit, such a fire is not expected to result in propagation via the cables routed through the top panel penetrations.

As the fire is located within the confines of the MPU, the fire is modeled as a radiant exposure case. However, consistent with the FIVE methodology, the postulated fire is conservatively modeled as if it were located at the top of the panel. This is consistent with the guidance provided in the EPRI Fire PRA Implementation Guide.

A critical radiant flux of 1 Btu/sec/ft^2 was used in the FIVE Radiant Exposure Worksheet to assess potential ignition of tray 0C-118 above MPU 1. This critical radiant flux is conservative because the shielding provided by the metal enclosure of the MPU is not credited. Appendix G of the EPRI Fire PRA Implementation Guide indicates that the critical radiant flux could be as high as 5 Btu/sec/ft^2 if the shielding provided by the top of the MPU was credited.

The MPU fire was modeling as having an intensity of 65 Btu/sec. (the heat rate recommended in the EPRI Fire PRA Implementation Guide for an electrical cabinet fire). The critical radiant flux distance obtained from the FIVE Radiant Exposure Worksheet is 1.44 ft. This is based on the recommended 0.40 radiant heat release fraction. With the fire source placed at the top of the unit, cable tray 0C-118 located about 1.5 feet above MPU 1, and considering the conservatism input in the analysis by not crediting shielding, ignition of cables in tray 0C-118 and subsequent damage to Division 1 cables in tray 1C-094 are not indicated by the fire modeling. In addition, since the fire is not expected to damage any of the cables routed above the panel, no other fire induced failures should be postulated.

MPU 3 Fire

Modular Power Unit MPU 3 is located a few feet south of the west cooling unit system. There is no safety related cable or equipment near enough to MPU 3 to be damaged by a postulated fire in the MPU. Cable tray 0C-118 passes directly overhead at a height of approximately 2 feet above the top of the MPU. However, unlike MPU 1, Division 1 cable trays are not located above MPU 3. A fire in MPU 3 can be postulated that would damage the cables in 0C-118; however, the cables in 0C-118 are not included in the PSA model and have no affect on the safe shutdown of the plant. Therefore, a postulated fire in MPU 3 can be best modeled as a plant trip with availability of all modeled systems (loss of MPU 3 itself does not fail any modeled systems).

MCC 72B-2A Fire

A postulated fire in MCC 72B-2A would result in loss of power to the loads supplied by MCC 72B-2A. This would result in loss of battery charger 2A-1 and swing charger 2A1-2. This in turn would result in loss of 260 Vdc panel 2PA-1 and 130 Vdc panel 2PA2-6 following battery depletion. In addition, MCC 72B-2A is located adjacent to MPU 1; as such, the spacial details discussed for MPU 1, associated vulnerabilities to overhead cables, and fire modeling treatment are also applicable for MCC 72B-2A.

Cabinet 72C-2D Fire

Cabinet 72C-2D is an enclosed steel cabinet located on the south wall of the room. Cables enter directly through the top of the cabinet (i.e., not via conduit). However, there are no safety related cables or equipment nearby that would be damaged by a postulated fire in the cabinet. The closest equipment is a steel cabinet, approximately 6" to the left of 72C-2D, which contains a number of miscellaneous disconnect switches. The credible fire in cabinet 72C-2D would be confined to the cabinet and only impact the functions supported by the cabinet.

Based on a load review, the only significant loads supplied by cabinet 72C-2D are alternate power to RPS-A and power to the control cabinet for main transformer 2A. Therefore, a postulated fire in cabinet 72C-2D is best modeled as a plant trip with availability of all modeled systems.

Cabinet 72C-2D-1 Fire

Cabinet 72C-2D-1 is powered by cabinet 72C-2D. The modeling approach for cabinet 72C-2D, as described above, applies to 72C-2D-1 as well. Therefore, a postulated fire in cabinet 72C-2D-1 is best modeled as plant trip with availability of all modeled systems.

Cabinet 2PA2-14 Fire

Cabinet 2PA2-14 is an enclosed steel cabinet located on the north wall of the room. Cable entries into and out of the cabinet are via conduit. In addition, this cabinet is sufficiently distant from nearby equipment and cables such that a postulated fire in the cabinet will damage only the equipment within the cabinet.

Cabinet 2PA2-14 supplies 130V DC control power to all Division I AC buses (i.e., buses 64B, 64C, 72B, and 72C). Although AC power remains available on the Division I AC buses, control power is not available to operate the breakers. Therefore, any Division I standby equipment will not automatically initiate when and if demanded. HPCI, RCIC, and BOP equipment are not affected. Therefore, a postulated fire in cabinet 2PA2-14 can be modeled as a plant trip with automatic initiation failure of the following Division I system trains: ADS/SRVs, LPCS, EECW/EESW, NIAS, and RHR.

Bus 64C Local Control Panel Fire

This panel is an enclosed steel cabinet attached to a concrete support column located in the center of the room. The only cable entries into and out of the cabinet are via conduit. In additi[illegible] this cabinet is sufficiently distant from nearby equipment and cables such that a postulated fire in the panel will damage only functions containe[illegible] [illegible]in the panel. Therefore, a postulated fire in this panel can be conservatively modeled as loss [illegible] [illegible]ower on bus 64C.

72B-2A Local Control Panel

This is an enclosed steel cabinet. This cabinet is sufficiently distant from nearby equipment and cables such that a postulated fire in the panel will damage only functions contained within the panel. Therefore, a postulated fire in this panel is best modeled as loss of all power on MCC 72B-2A. See MCC 72B-2A discussion above for impacts.

MCC 72C-F Isolating Contactor (Div. I) Fire

MCC 72C-F Isolating Contactor (Div. I) is an enclosed steel panel that is located at the west end of bus 72C. It is separated from bus 72C by approximately 1 1/2 feet. A fire in this panel is expected to remain within the boundaries of the panel and not damage nearby equipment or cables.

This panel contains contacts and circuitry connecting bus 72C to the swing bus 72C-F. Swing bus 72C-F powers all RHR LPCI mode injection valves. Swing bus 72C-F is assumed to be normally fed by bus 72C. Failure of MCC 72C-F Isolating Contactor (Div. I) due to a fire can be postulated to result in loss of the 72C feed to the swing bus. Electrical schematics were reviewed to determine whether the automatic transfer to 72F power would remain available. It was determined that the other power feed (i.e., 72F) to the swing bus is an independent circuit routed in a separate area of the plant and, thus, the automatic transfer to 72F power would still be available given a fire in the MCC 72C-F Isolating Contactor (Div. I) panel. Therefore, this postulated fire is best modeled as a failure of power feed 72C to swing bus 72C-F.

Miscellaneous Electrical Item Fires

Fire compartment 04AB contains a number of small miscellaneous electrical items (i.e., those items not discussed individually above). This miscellaneous items category is comprised of disconnect switches, an instrument rack, and a radiation monitor. These miscellaneous items were categorized together as the associated fires are judged to be small and would result in, at most, a plant trip. Therefore, this group of miscellaneous electrical items is best modeled collectively as resulting in a plant trip.

Transient Fires

A walkdown of the area did not identify any transient ignition sources or combustible materials. However, to bound the transient combustible fire scenario, a trash container fire is conservatively postulated in the room. This fire was modeled with the QUICK FIVE software as having a duration of 5 minutes and a heat rate of 380 Btu/sec.; this is consistent with fire modeling examples in the FIVE methodology. The height of the point source trash container was assumed to be 4 feet above the floor, consistent with the FIVE methodology.

The Target-In-Plume and Radiant Exposure worksheets indicate that the critical radiant flux distance is approximately 3.4 feet. The critical height for the Target-In-Plume case for a trash container extends to the ceiling. Therefore, the analysis of this area needs to address the likelihood of a transient combustible source placed under cable trays or near equipment, and the possibility of a resulting fire.

Hot Work Fires

Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets or transient combustibles to cause ignition. No deterministic fire modeling is performed to define the conditions necessary for accidental hot work induced cable ignition. Rather, the critical distance is based on the trash container fire modeling results discussed above.

Accident Sequence Quantification

Cooling Unit Fire

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to fixed ignition sources:

$F_{fx} = F_{if} \times P_{fs} \times CCDP$, where:

F_{fx} = core damage frequency (due to fixed sources)

F_{if} = fire ignition source frequency (due to fixed sources)

P_{fs} = probability of fire suppression failure

CCDP = Conditional Core Damage Probability

This area is not equipped with an automatic fire suppression system. No credit is taken here for fire brigades.

Fire modeling for the cooling units indicates that fire induced failure of the cooling units is best modeled as leading to a plant trip with availability of all modeled equipment. The conditional core damage probability for this event is conservatively based on an existing run (BASE, CCDP=1.3E-5) that includes failure of the main condenser.

Therefore, the core damage frequency for the cooling units in this area is conservatively calculated as follows:

F = 2.0E-4 x 1.0 x 1.3E-5 = 2.60E-9/yr

480V 72C Bus/Transformer Fire

The fire modeling discussion above indicates that the core damage frequency associated with a postulated fire of the 72C switchgear unit is best modeled as three separate scenarios:

- Scenario 1: Bounding worst case fire. This fire is conservatively assumed to result in loss of all functions in the room.
- Scenario 2: 4kV/480V transformer fire. This fire has been shown to result in loss of power on the entire bus.
- Scenario 3: Circuit breaker cubicle fire. As discussed above, this fire can be conservatively modeled as resulting in loss of power on the entire bus.

The bounding worst case fire is conservatively assumed to comprise 10% of all postulated 72C fires. The remaining 90% of the fires are assumed to be comprised of transformer and circuit breaker cubicle fires. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated Bus 72C fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.4E-2 (run 4AB04 - see Table 4-4). The conditional core damage probability used for Scenarios 2 and 3 is conservatively based on run 4AB15 (CCDP=1.6E-4) which assumes failure of both 480V Bus 72C and associated 4kV Bus 64C.

Scenario 1: F = 0.1 x 3.0E-4 x 1.0 x 3.4E-2 = 1.02E-6/yr

Scenarios 2 & 3: F = 0.9 x 3.0E-4 x 1.0 x 1.6E-4 = 4.32E-8/yr

Therefore, the total core damage frequency associated with postulated fires of 480V Bus 72C is 1.06E-6/yr.

480V 72B Bus/Transformer Fire

The core damage frequency associated with postulated fires of 480V Bus 72B is calculated in the same manner as discussed above for Bus 72C. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated Bus 72B fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.4E-2 (run 4AB04 - see Table 4-4). The conditional core damage probability used for Scenarios 2 and 3 is conservatively based on run 4AB13 (CCDP=1.4E-4 - see Table 4-4) which assumes failure of both 480V Bus 72B and associated 4kV Bus 64B.

Scenario 1: F = 0.1 x 3.0E-4 x 1.0 x 3.4E-2 = 1.02E-6/yr

Scenarios 2 & 3: F = 0.9 x 3.0E-4 x 1.0 x 1.4E-4 = 3.78-8/yr

Therefore, the total core damage frequency associated with postulated fires of 480V Bus 72B is 1.05E-6/yr.

4160V 64C Bus Fire

The fire modeling discussion above indicates that the core damage frequency associated with a postulated fire of the 64C switchgear unit is best modeled as three separate scenarios:

- Scenario 1: Bounding worst case fire. This fire is conservatively assumed to result in loss of all functions in the room.
- Scenario 2: Circuit breaker cubicle fire resulting in loss of power feed from 64 transformer and loss of the 64C bus. One breaker compartment out of a total of seven can potentially result in this scenario.
- Scenario 3: Circuit breaker cubicle fire resulting in loss of power on the 64C bus. Six breaker compartments out of a total of seven can potentially result in this scenario.

The bounding worst case fire is conservatively assumed to comprise 10% of all postulated 64C fires. The remaining 90% of the fires are assumed to be comprised of breaker compartment fires. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated Bus 64C fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.4E-2 (run 4AB04 - see Table 4-4). The conditional core damage probability used for Scenario 2 is 1.9E-3 (run 4AB14 - see Table 4-4). The conditional core damage probability for Scenario 3 is 1.6E-4 (run 4AB15 - see Table 4-4).

Scenario 1: F = 0.1 x 1.0E-4 x 1.0 x 3.4E-2 = 3.40E-7/yr

Scenario 2: F = 0.9 x (1/7) x 1.0E-4 x 1.0 x 1.9E-3 = 2.44E-8/yr

Scenario 3: F = 0.9 x (6/7) x 1.0E-4 x 1.0 x 1.6E-4 = 1.23E-8/yr

Therefore, the total core damage frequency associated with postulated fires of 4kV Bus 64C is 3.74E-7/yr.

4160V 64B Bus Fire

The fire modeling discussion above indicates that the core damage frequency associated with a postulated fire of the 64B switchgear unit is best modeled as three separate scenarios:

- Scenario 1: Bounding worst case fire. This fire is conservatively assumed to result in loss of all functions in the room.
- Scenario 2: Circuit breaker cubicle fire resulting in loss of power feed from 64 transformer and loss of the 64B bus. One breaker compartment out of a total of ten can potentially result in this scenario.
- Scenario 3: Circuit breaker cubicle fire resulting in loss of power on the 64B bus. Nine breaker compartments out of a total of ten can potentially result in this scenario.

The bounding worst case fire is conservatively assumed to comprise 10% of all postulated 64C fires. The remaining 90% of the fires are assumed to be comprised of breaker compartment fires. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated Bus

64B fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.4E-2 (run 4AB04 - see Table 4-4). The conditional core damage probability used for Scenario 2 is 3.6E-4 (run 4AB12 - see Table 4-4). The conditional core damage probability for Scenario 3 is 1.4E-4 (run 4AB13 - see Table 4-4).

Scenario 1: F = 0.1 x 1.4E-4 x 1.0 x 3.4E-2 = 4.76E-7/yr

Scenario 2: F = 0.9 x (1/10) x 1.4E-4 x 1.0 x 3.6E-4 = 4.54E-9/yr

Scenario 3: F = 0.9 x (9/10) x 1.4E-4 x 1.0 x 1.4E-4 = 1.59E-8/yr

Therefore, the total core damage frequency associated with postulated fires of 4kV Bus 64C is 4.96E-7/yr.

64T Breaker Cabinet Fire

The fire modeling discussion above indicates that the core damage frequency associated with a postulated fire of the 64T breaker cabinet is best modeled as two separate scenarios:

- Scenario 1: Bounding worst case fire. This fire is conservatively assumed to result in loss of all functions in the room.
- Scenario 2: Circuit breaker cubicle fire. As discussed above, this fire can be conservatively modeled as resulting in loss of the 64 transformer feed to both 4kV buses 64B and 64C.

The bounding worst case fire is conservatively assumed to comprise 10% of all postulated 64T fires. The remaining 90% of the fires are assumed to be comprised of circuit breaker cubicle fires. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated 64T breaker cabinet fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.4E-2 (run 4AB04 - see Table 4-4). The conditional core damage probability used for Scenario 2 is 2.0E-4 (run TRANS - see Table 4-4).

Scenario 1: F = 0.1 x 1.4E-5 x 1.0 x 3.4E-2 = 4.76E-8/yr

Scenario 2: F = 0.9 x 1.4E-5 x 1.0 x 2.0E-4 = 2.52E-9/yr

Therefore, the total core damage frequency associated with postulated fires of the 64T breaker cabinet is 5.01E-8/yr.

MPU 1 Fire

Fire modeling for MPU 1 indicates that fire induced failure of MPU 1 is best modeled as leading to a plant trip with loss of the main condenser. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated MPU 1 fire scenario is calculated below. The conditional core damage probability for this event is based on an existing run (BASE, CCDP=1.3E-5 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of MPU 1 is calculated as follows:

F = 4.9E-4 x 1.0 x 1.3E-5 = 6.37E-9/yr

MPU 3 Fire

Fire modeling for MPU 3 indicates that fire induced failure of MPU 3 is best modeled as leading to a plant trip with availability of all modeled systems. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated MPU 3 fire scenario is calculated below.

The conditional core damage probability for this event is conservatively based on an existing run (BASE, CCDP=1.3E-5) which includes loss of the main condenser.

Therefore, the core damage frequency for a postulated fire of MPU 3 is conservatively calculated as follows:

F = 4.9E-4 x 1.0 x 1.3E-5 = 6.37E-9/yr

MCC 72B-2A Fire

Fire modeling for MCC 72B-2A indicates that fire induced failure of this MCC can be conservatively modeled as leading to a plant trip with loss of Division I DC. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated MCC fire scenario is calculated below. The conditional core damage probability for this event is conservatively based on an existing run (10AB1, CCDP=4.0E-4 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of MCC 72B-2A is conservatively calculated as follows:

F = 2.6E-4 x 1.0 x 4.0E-4 = 1.04E-7/yr

Cabinet 72C-2D Fire

Fire modeling for cabinet 72C-2D indicates that fire induced failure of this cabinet can be best modeled as leading to a plant trip with availability of all modeled systems. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated MCC fire scenario is calculated below. The conditional core damage probability for this event is 1.3E-5 (run BASE - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of cabinet 72C-2D is calculated as follows:

F = 1.7E-4 x 1.0 x 1.3E-5 = 2.21E-9/yr

Cabinet 72C-2D-1 Fire

Fire modeling for cabinet 72C-2D-1 indicates that fire induced failure of this cabinet can be best modeled as leading to a plant trip with availability of all modeled systems. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated MCC fire scenario is calculated below. The conditional core damage probability for this event is 1.3E-5 (run BASE - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of cabinet 72C-2D-1 is calculated as follows:

F = 9.7E-5 x 1.0 x 1.3E-5 = 1.26E-9/yr

Cabinet 2PA2-14 Fire

Fire modeling for cabinet 2PA2-14 indicates that fire induced failure of this cabinet can be best modeled as leading to a plant trip with automatic initiation failure of the following Division I system trains: ADS/SRVs, LPCS, EECW/EESW, NIAS, and RHR. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated MCC fire scenario is calculated below. The conditional core damage probability for this event is conservatively based on an existing run (10AB1, CCDP=4.0E-4) which assumes failure of all Division I DC.

Therefore, the core damage frequency for a postulated fire of cabinet 2PA2-14 is calculated as follows:

F = 1.9E-4 x 1.0 x 4.0E-4 = 7.60E-8/yr

Bus 64C Local Control Panel Fire

Fire modeling for Bus 64C local control panel indicates that fire induced failure of this panel can be conservatively modeled as leading to a plant trip with loss of all power on Bus 64C. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated panel fire scenario is calculated below. The conditional core damage probability for this event is 1.6E-4 (run 4AB15 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of Bus 64C local control panel is conservatively calculated as follows:

F = 1.4E-5 x 1.0 x 1.6E-4 = 2.24E-9/yr

72B-2A Local Control Panel

Fire modeling for 72B-2A local control panel indicates that fire induced failure of this panel can be modeled as resulting in the same effects as a fire in MCC 72B-2A. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated panel fire scenario is calculated below. The conditional core damage probability for this event is conservatively based on an existing run (10AB1, CCDP=4.0E-4 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of 72B-2A local control panel is conservatively calculated as follows:

F = 1.4E-5 x 1.0 x 4.0E-4 = 5.60E-9/yr

MCC 72C-F Isolating Contactor (Div. I) Fire

Fire modeling for MCC 72C-F Isolating Contactor (Div. I) panel indicates that fire induced failure of this panel can be best modeled as a failure of power feed 72C to swing bus 72C-F. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated panel fire scenario is calculated below. The conditional core damage probability for this event is 3.5E-5 (run 4AB07 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of this panel is calculated as follows:

F = 1.4E-5 x 1.0 x 3.5E-5 = 4.90E-10/yr

Miscellaneous Electrical Item Fires

Fire modeling of the remaining miscellaneous electrical items indicates that fire induced failure of this group of components can be best modeled collectively as a leading to a plant trip with availability of all modeled systems. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of these postulated fire scenarios is calculated below. The conditional core damage probability for this event is conservatively based on an existing run (BASE, CCDP=1.3E-5) which assumes loss of the main condenser.

Therefore, the core damage frequency for these postulated fires is conservatively calculated as follows:

F = 9.8E-5 x 1.0 x 1.3E-5 = 1.27E-9/yr

Transient Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to transient combustible sources:

$F_t = F_{it} \times u \times p \times w \times P_{fs} \times CCDP$, where:

F_t	=	core damage frequency (due to transients)
F_{it}	=	fire ignition source frequency (due to transients)
u	=	transient combustible located in the range of target components
p	=	probability of combustible being exposed
w	=	frequency of finding a critical combustible loading versus the frequency of inspecting and removing the transient combustible fire source.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters P_{fs} and p are conservatively set to 1.0.

Per the FIVE methodology, u is calculated as follows:

$u = (A_s + A_{sr})$ / Net Area, where:

A_s	=	Exposed surface area of targets facing floor (e.g., width x length of horizontal cable trays)
A_{sr}	=	Area around radiant target determined by the critical separation distance (determined to be 3.4 ft., as described above)

A review of cable tray drawings shows that an approximate value of u is 0.75. This value is comparatively high due to the large number of cable trays in the ceiling and the large switchgear units in the center of the room. There is very little floor space in the room where a trash container fire would not result in damage to nearby equipment or cables.

Per the FIVE methodology, w is calculated as follows:

$w = (x/2) \times \ln(1/x)$, where $x = F_{ccl}/F_w$

F_{ccl}	=	The frequency of having a critical combustible loading present. FIVE provides a screening value of 1 event/year/compartment.
F_w	=	Frequency of combustible material inspections that would find the transient combustible fire source before a fire occurred.

The value of 1 event/year is used here for F_{ccl}.

Administrative procedure NPP-FP1-01 specifies that: 1) monthly fire protection inspections are to be performed, and 2) an hourly compensatory fire watch is to tour all areas appearing on the Out of Specification Log Sheet. This same procedure states that the accumulation of trash, oil rags, combustible materials, and similar fire hazards is prohibited. It is judged that either of the above tours would identify and correct a trash container inappropriately placed near a cable tray riser. As the hourly compensatory fire watch may not tour this area hourly or every day, a conservative estimate of one tour per week is assumed ($F_w = 52/yr$). Therefore, w for this area is estimated as follows:

$w = ((1/52)/2) \times \ln(1/(1/52)) = 3.79E\text{-}2$

A bounding conditional core damage probability of the above fire can be conservatively estimated using the initial screening quantification run (4AB04, CCDP=3.4E-2 - see Table 4-4). This is a bounding estimate as it includes fire damage to many cable trays.

Therefore, the core damage frequency for this area due to transient combustible fires is conservatively estimated as follows:

$F_t = 2.1E\text{-}4 \times 0.75 \times 1.0 \times 3.79E\text{-}2 \times 1.0 \times 3.4E\text{-}2 = 2.03E\text{-}7/yr$

Hot Work Fires

Hot work induced core damage frequency is estimated as follows:

$F_{hw} = F_{ihw} \times u \times HEP_{hw} \times P_{fs} \times CCDP$, where:

F_{hw}	=	core damage frequency (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.
u	=	probability of hot work performed in close enough proximity to important cable targets or transient combustibles such that the possibility exists for ignition
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameter P_{fs} is conservatively set to 1.0.

The parameter u is estimated with a value of 0.75. This value is the same as that used in the transient combustible case and is based on a 5 min. trash container fire with a heat rate of 380 Btu/sec.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

A bounding conditional core damage probability of the above fire can be conservatively estimated using the initial screening quantification run (4AB04, CCDP=3.4E-2 - see Table 4-4). This is a bounding estimate as it includes fire damage to many cable trays.

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

F_{hw} = 8.4E-4 x 0.75 x 5.0E-2 x 1.0 x 3.4E-2 = 1.07E-6/yr

Summary

The total core damage frequency due to fire in this area is calculated by summing the estimated core damage frequencies of the above fire scenarios. The core damage frequency for this area is estimated at 4.51E-6/yr. This estimate is above the 1E-6/yr FIVE screening criterion. The FIVE methodology used in this analysis does not support a core damage frequency estimate due to fire in this area less than the 1E-6/yr FIVE screening criterion. Refer to Section 7 of this report for recommendations.

4B.2.3.5 Division II Cable Chase of 04AB - 04ABS

The initial screening quantification of this compartment resulted in a core damage frequency higher than the 1E-6/yr FIVE screening criterion. Further analysis was performed to reduce the conservatism in the initial screening analysis.

A review of the base ignition frequency calculation for this area showed that there are no fixed ignition sources in this area.

The only fire ignition sources in the base ignition frequency calculation are transient sources and ignition sources due to hot work (e.g., open flames, welding, grinding, or arc techniques). However, Administrative Procedure NPP-FP1-01 places restrictions on transient ignition sources. In addition, NPP-FP1-01 prescribes certain controls and requirements be satisfied prior to the start of hot work activities. These requirements involve the consideration of combustible or flammable material, protection of combustibles from ignition sources, and the establishment of fire protection measures. NPP-FP1-01 requires the establishment of fire watches during and for at least 30 minutes after the completion of any hot work.

The combination of the lack of significant fixed ignition sources and the fire ignition control measures are considered adequate to preclude a credible fire event from damaging cables in this compartment. However, in order to develop a fire risk screening value, a conservative analysis is presented below.

Fire Ignition Frequency

Based on the discussion above, the contributors to the area fire ignition frequency are as follows:

- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr
- Transient ignition sources: 2.1E-4/yr

Potential transient ignition sources, based on the FIVE methodology, include cigarette smoking, extension cords, heaters, candles, overheating, and hot pipes. The only transient sources listed in the base fire ignition frequency calculation for the cable spreading room are extension cords and heaters. Compartment 4ABS is a cable chase. It is a small room (approximately 8' x 12') that is accessible by a single door. The use of extension cords or heaters in this chase is not considered credible. Therefore, transient fire scenarios are not quantified.

In addition, due to the size and function of this compartment, the storage of significant combustible materials in the room while at power is also not considered credible. Therefore, the ignition of transient combustibles due to hot work activity is not quantified.

As stated above, administrative procedure NPP-FP1-01 prescribes controls on hot work that minimize or reduce to negligible the likelihood of cable ignition in this compartment. However, although considered unlikely while at power, the hot work ignition of cables is conservatively maintained for analysis.

Fire Damage Scenarios

As stated above, the only credible fires are cable ignitions due to hot work. Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets to cause ignition. No deterministic fire modeling is performed to define the conditions necessary for accidental hot work induced cable ignition.

Accident Sequence Quantification

Hot work induced core damage frequency is estimated as follows:

$F_{hw} = F_{ihw} \times u \times HEP_{hw} \times P_{fs} \times CCDP$, where:

F_{hw}	=	core damage frequency (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.
u	=	probability of hot work performed in close enough proximity to important cable targets such that the possibility exists for ignition
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters u, HEP_{hw} and P_{fs} are conservatively set to 1.0.

A bounding conditional core damage probability of the above fire can be conservatively estimated using the initial screening quantification run (4AB03, CCDP=2.0E-3 - see Table 4-4). This is a bounding estimate as it includes fire damage to many cable trays.

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

$F_{hw} = 1.2E\text{-}4 \times 1.0 \times 1.0 \times 1.0 \times 2.04E\text{-}3 = 2.45E\text{-}7/yr$

This estimate is below the 1E-6/yr FIVE screening criterion. Therefore, this area is screened from further analysis.

4B.2.3.6 East Cable Tunnel - 05ABE

The initial screening quantification of the east cable tunnel resulted in a core damage frequency higher than the 1E-6/yr FIVE screening criterion. Further analysis was performed to reduce the conservatism in the initial screening analysis.

A review of the base ignition frequency calculation for this area showed that there are no fixed ignition sources in the room.

The other fire ignition sources in the base ignition frequency calculation are transient sources and ignition sources due to hot work (e.g., open flames, welding, grinding, or arc techniques). However, Administrative Procedure NPP-FP1-01 places restrictions on transient ignition sources. In addition, NPP-FP1-01 prescribes certain controls and requirements be satisfied prior to the start of hot work activities. These requirements involve the consideration of combustible or flammable material, protection of combustibles from ignition sources, and the establishment of fire protection measures. NPP-FP1-01 requires the establishment of fire watches during and for at least 30 minutes after the completion of any hot work.

The combination of the lack of significant fixed ignition sources and the fire ignition control measures are considered adequate to preclude a credible fire event from damaging cables in the east cable tunnel. However, in order to develop a fire risk screening value, a conservative analysis is presented below.

Fire Ignition Frequency

Based on the discussion above, the contributors to the area fire ignition frequency are as follows:

- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr
- Transient ignition sources: 2.1E-4/yr

Potential transient ignition sources, based on the FIVE methodology, include cigarette smoking, extension cords, heaters, candles, overheating, and hot pipes. The only transient sources listed in the base fire ignition frequency calculation for the cable spreading room are extension cords and heaters. The use of extension cords or heaters in either of the cable tunnels while at power is not considered credible. Therefore, transient fire scenarios are not quantified.

There is no clear floor space in the cable tunnel. Therefore, the storage of combustible materials in the tunnel is not considered credible. As such, the ignition of transient combustibles due to hot work activity is not quantified.

As stated above, administrative procedure NPP-FP1-01 prescribes controls on hot work that minimize or reduce to negligible the likelihood of cable ignition in the cable tunnels. However, although considered unlikely while at power, the hot work ignition of cables is conservatively maintained for analysis.

Fire Damage Scenarios

As stated above, the only credible fires are cable ignitions due to hot work. Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets to cause ignition. No deterministic fire modeling is performed to define the conditions necessary for accidental hot work induced cable ignition. Rather, the critical distance is based on the trash container fire modeling results discussed above.

Accident Sequence Quantification

Hot work induced core damage frequency is estimated as follows:

$F_{hw} = F_{ihw} \times u \times HEP_{hw} \times P_{fs} \times CCDP$, where:

F_{hw} = core damage frequency (due to hot work)

F_{ihw} = fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.

u = probability of hot work performed in close enough proximity to important cable targets such that the possibility exists for ignition

HEP_{hw} = human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.

P_{fs} = probability of fire suppression failure

CCDP = Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters u and P_{fs} are conservatively set to 1.0.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

A bounding conditional core damage probability of the above fire can be conservatively estimated using the initial screening quantification run (5AB03, CCDP=3.3E-2 - see Table 4-4). This is a bounding estimate as it includes fire damage to many cable trays.

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

F_{hw} = 1.2E-4 x 1.0 x 5.0E-2 x 1.0 x 3.3E-2 = 1.98E-7/yr

This estimate is below the 1E-6/yr FIVE screening criterion. Therefore, this area is screened from further analysis.

4B.2.3.7 West Cable Tunnel - 05ABW

The initial screening quantification of the west cable tunnel resulted in a core damage frequency higher than the 1E-6/yr FIVE screening criterion. Further analysis was performed to reduce the conservatism in the initial screening analysis.

A review of the base ignition frequency calculation for this area showed that there are no fixed ignition sources in the room.

Fire Ignition Frequency

The ignition frequency discussion provided earlier for compartment 05ABE applies here for compartment 05ABW.

Fire Damage Scenarios

The core damage frequency of a fire in compartment 05ABW is calculated in the same manner as that calculated earlier for compartment 05ABE. The one difference is the value for the conditional core damage probability. The CCDP is conservatively based on the initial screening quantification run for 05ABW (5AB04, CCDP=3.5E-2 - see Table 4-4). This is a bounding estimate as it includes fire damage to many cable trays.

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

F_{hw} = 1.2E-4 x 1.0 x 5.0E-2 x 1.0 x 3.5E-2 = 2.10E-7/yr

This estimate is below the 1E-6/yr FIVE screening criterion. Therefore, this area is screened from further analysis.

4B.2.3.8 2nd Floor Miscellaneous Rooms - 06AB

The initial screening quantification of the Auxiliary Building second floor miscellaneous rooms resulted in a core damage frequency greater than the 1E-6/yr FIVE screening criterion. Therefore, further analysis was performed to reduce the conservatism in the initial screening analysis.

Fire Ignition Frequency

A review of the base ignition frequency calculation for this area showed that the ignition sources in this area include:

- Four (4) fire protection panels 4.9E-5/yr
- Three (3) battery operated lights 1.4E-4/yr
- One (1) instrument panel 4.5E-5/yr
- One (1) air conditioning unit 2.9E-5/yr
- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr
- Transient ignition sources: 2.1E-4/yr

A credible fire in either of the panels or the emergency light would be small and confined to within the boundaries of the panel. In addition, these postulated fires would quickly extinguish due to limited combustibles and would realistically not result in a plant trip. Therefore, these components are excluded as credible ignition sources that could lead to a core damage accident.

As this compartment contains a personnel changing area and other miscellaneous rooms, the existence of transient combustibles and transient ignition sources are maintained for further analysis. Hot work ignition sources are also maintained.

Fire Damage Scenarios

Based on the ignition source frequency review discussed above, the following fire damage scenarios are postulated:

- A/C unit fire
- Transient ignition sources

- Cable fires due to hot work
- Transient combustible fires due to hot work

A/C Unit Fires

This area contains A/C unit T4100B057. The ignition source component of this A/C unit is judged to be equivalent to a motor-driven grease lubricated fan.

A walkdown of the compartment and a review of arrangement and cable tray drawings determined that the nearest non-BOP cable target (BOP cable targets were not investigated) to the A/C unit are cable trays 1P-056 (el. 626' 6") and DC1P-056 (el. 625'). Conservatively placing the motor four feet off the floor, the closest radial distance to these targets is approximately 8 feet. As the fan motor fire would be a motor windings fire inside the motor housing, the fan motor fire is modeled using the FIVE methodology radiant exposure approach. As the FIVE methodology does not provide heat release rates for motors, a bounding approach was used to determine if target damage is likely. The spacial and other fire modeling information was input into the QUICK FIVE automated radiant exposure worksheet, and then the peak fire intensity was adjusted until the critical radiant flux distance equaled 8 feet (the line of site distance from the motor to the closest target, DC1P-056). The result was a peak fire intensity over 2000 Btu/sec. This is an overly conservative heat release rate. It is approximately four times that of a 32-gallon trash container fire, as provided in the FIVE methodology. Therefore, fire modeling does not indicate that damage will occur to any non-BOP cabling due to a fire initiated in A/C unit T4100B057.

Transient Fires

Transient combustibles in this compartment consist of material utilized for equipment calibration and, potentially, personnel clothing. To bound the transient combustible fire scenario, a trash container fire is conservatively postulated in the room. This fire was modeled with the QUICK FIVE software as having a duration of 5 minutes and a heat rate of 380 Btu/sec; this is consistent with fire modeling examples in the FIVE methodology. The height of the point source trash container fire was assumed to be 4 feet above the floor, consistent with the FIVE methodology.

The critical height for the Target-In-Plume case extends to the ceiling of this compartment if the trash can is placed in a corner. As this area contains cable trays suspended from the ceiling, the analysis of this area needs to address the likelihood of a transient combustible source placed under horizontal cable trays or near vertical cable trays and the possibility of a resulting fire.

Hot Work Fires

Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets or transient combustibles to cause ignition. No deterministic fire modeling is performed to define the conditions necessary for accidental hot work induced cable ignition. Rather, the critical distance is based on the trash container fire modeling results discussed above.

Accident Sequence Quantification

A/C Unit Fire

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to fixed ignition sources:

$F_{fx} = F_{if} \times P_{fs} \times CCDP$, where:

F_{fx}	=	core damage frequency (due to fixed sources)
F_{if}	=	fire ignition source frequency (due to fixed sources)
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

Fire suppression is conservatively not credited.

The fire modeling performed for the A/C unit does not indicate damage to any nearby non-BOP cabling or equipment. However, the effect on BOP cabling was not evaluated. Therefore, the core damage frequency estimate for this ignition contributor is conservatively modeled as leading to a plant trip with loss of the main condenser and loss of the hard-piped vent. The conditional core damage probability associated with this damage is conservatively estimated at 2E-4 (rather than perform an additional run, this CCDP is conservatively based on an existing run, 14AE2, that included failure of the main condenser, the hard-piped vent <u>and</u> RCIC).

Therefore, the core damage frequency for this area due to an A/C unit fire is conservatively estimated as follows:

$F_{fx} = 2.9E\text{-}5 \times 1.0 \times 2.0E\text{-}4 = 5.70E\text{-}9/yr$

<u>Transient Fires</u>

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to transient combustible sources:

$F_t = F_{it} \times u \times p \times w \times P_{fs} \times CCDP$, where:

F_t	=	core damage frequency (due to transients)
F_{it}	=	fire ignition source frequency (due to transients)
u	=	transient combustible located in the range of target components
p	=	probability of combustible being exposed
w	=	frequency of finding a critical combustible loading versus the frequency of inspecting and removing the transient combustible fire source.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters P_{fs} and p are conservatively set to 1.0.

Per the FIVE methodology, u is calculated as follows:

$u = (A_s + A_w)$ / Net Area, where:

A_s	=	Exposed surface area of targets facing floor (e.g., width x length of horizontal cable trays)

A_{sr} = Area around radiant target determined by the critical separation distance (determined to be 3.4 ft., as described above)

A review of cable tray drawings shows that an approximate value of u is 0.20.

Per the FIVE methodology, w is calculated as follows:

$w = (x/2) \times \ln(1/x)$, where $x = F_{ccl}/F_w$

F_{ccl} = The frequency of having a critical combustible loading present. FIVE provides a screening value of 1 event/year/compartment.

F_w = Frequency of combustible material inspections that would find the transient combustible fire source before a fire occurred.

The value of 1 event/year is used here for F_{ccl}.

Administrative procedure NPP-FP1-01 specifies that: 1) monthly fire protection inspections are to be performed, and 2) an hourly compensatory fire watch is to tour all areas appearing on the Out of Specification Log Sheet. This same procedure states that the accumulation of trash, oil rags, combustible materials, and similar fire hazards is prohibited. It is judged that either of the above tours would identify and correct a trash container inappropriately placed near a cable tray riser. As the hourly compensatory fire watch may not tour this area hourly or every day, a conservative estimate of one tour per week is assumed (F_w = 52/yr). Therefore, w for this area is estimated as follows:

$w = ((1/52)/2) \times \ln(1/(1/52)) = 3.79E\text{-}2$

A bounding conditional core damage probability of the above fire can be conservatively estimated using the initial screening quantification run (6AB02, CCDP=6.3E-2 - see Table 4-4). This is a bounding estimate as it includes fire damage to many cable trays.

Therefore, the core damage frequency for this area due to transient combustible fires is conservatively estimated as follows:

$F_t = 2.1E\text{-}4 \times 0.20 \times 1.0 \times 3.79E\text{-}2 \times 1.0 \times 6.3E\text{-}2 = 1.00E\text{-}7/yr$

Hot Work Fires

Hot work induced core damage frequency is estimated as follows:

$F_{hw} = F_{ihw} \times u \times HEP_{hw} \times P_{fs} \times CCDP$, where:

F_{hw} = core damage frequency (due to hot work)

F_{ihw} = fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.

U = probability of hot work performed in close enough proximity to important cable targets or transient combustibles such that the possibility exists for ignition

HEP_{hw} = human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.

P_{fs} = probability of fire suppression failure

CCDP = Conditional Core Damage Probability

Fire suppression is conservatively not credited.

The parameter u is estimated with a value of 0.20. This value is the same as that used in the transient combustible case and is based on a 5 min. trash container fire with a heat rate of 380 Btu/sec.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

A bounding conditional core damage probability of the above fire can be conservatively estimated using the initial screening quantification run (6AB02, CCDP=6.3E-2 - see Table 4-4). This is a bounding estimate as it includes fire damage to many cable trays.

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

F_{hw} = 8.4E-4 x 0.20 x 5.0E-2 x 1.0 x 6.3E-2 = 5.29E-7/yr

Summary

Summing F_{fx}, F_t and F_{hw}, the total core damage frequency due to fire in this area is conservatively estimated at 6.35E-7/yr. This estimate is below the 1E-6/yr FIVE screening criterion. Therefore, this area is screened from further analysis.

4B.2.3.9 Cable Spreading Room - 07AB

An initial screening quantification of the cable spreading room was judged unnecessary. A postulated fire in this room could potentially affect sufficient plant systems to result in a core damage frequency orders of magnitude above the 1E-6/yr FIVE screening criterion. However, such a postulated fire is overly conservative.

Walkdown of the cable spreading room and review of the base fire ignition frequency calculation shows that the only fixed ignition sources in the room are two (2) enclosed fire protection panels (T80P402A and T82P452A). A credible fire in either of these panels would be small and confined to within the boundaries of the panel. As such, a postulated fire due to the fixed ignition sources is not expected to result in damage beyond failure of a fire protection panel. In addition, a fire protection panel fire would quickly self-extinguish due to limited combustibles and would realistically not result in a plant trip.

The remaining fire ignition sources are transient ignition sources and ignition sources due to hot work (e.g., open flames, welding, grinding or arc techniques). However, Administrative Procedure NPP-FP1-01 places restrictions on transient ignition sources. In addition, NPP-FP1-01 prescribes certain controls and requirements be satisfied prior to the start of hot work activities. These requirements involve the consideration of combustible or flammable material, protection of combustibles from ignition sources, and the establishment of fire protection measures. NPP-FP1-01 requires the establishment of fire watches during and for at least 30 minutes after the completion of any hot work. In addition, it is judged, based on plant operations and culture, that hot work and transient combustibles in the cable spreading room simply will not be allowed while at power.

The combination of the walkdown results and the fire ignition control measures are considered adequate to preclude a credible fire event from damaging cables in the cable spreading room. However, in order to develop a fire risk screening value, further analysis is presented below.

Ignition Frequency

The base fire ignition frequency calculation for the cable spreading room identified the following contributors:

- Two (2) fire protection panels: 2.4E-5/yr
- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr
- Transient ignition sources: 2.1E-4/yr

Based on the discussion above, it is appropriate to exclude the fire protection panels as credible fire scenarios that would result in core damage end states. As such, the fire ignition frequency contribution due to the fire protection panels are deleted from the fire ignition frequency estimate for the cable spreading room.

As stated above, administrative procedure NPP-FP1-01 prescribes controls on hot work that minimize or reduce to negligible the likelihood of ignitions in the cable spreading room. However, although considered unlikely while at power, the hot work ignition source is conservatively maintained for analysis.

Potential transient ignition sources, based on the FIVE methodology, include cigarette smoking, extension cords, heaters, candles, overheating, and hot pipes. The only transient sources listed in the base fire ignition frequency calculation for the cable spreading room are extension cords and heaters. As the cable spreading room is part of the control center HVAC system, it is judged here that the placement of a heater in the cable spreading room is not likely. In addition, the use of extension cords in the cable spreading room while at power is not considered credible. Therefore, transient fire scenarios are not quantified.

Fire Damage Scenarios

Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets or transient combustibles to cause ignition. No deterministic fire modeling is performed to define the conditions necessary for accidental hot work induced cable ignition. Rather, the critical distance is based on trash container fire modeling results.

The cable spreading room is provided with a full coverage automatic Halon suppression system **[4.3]** Therefore, two accident scenarios can be postulated.

- <u>Scenario #1</u>: Hot work fire with unsuccessful suppression
- <u>Scenario #2</u>: Hot work fire with successful suppression

Accident Sequence Quantification

<u>Scenario #1</u>

The core damage frequency for Scenario #1 is estimated as follows:

$F_{hw1} = F_{ihw} \times u \times HEP_{hw} \times P_{fs} \times CCDP$, where:

F_{hw1}	=	core damage frequency of Scenario #1 (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.
u	=	probability of hot work performed in close enough proximity to important cable targets such that the possibility exists for cable ignitions
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

The parameter u is estimated with a value of 0.25. This value is based on a 5 min. trash container fire with a heat rate of 380 Btu/sec.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

Per Attachment 10.3 of the FIVE methodology, the failure probability of a Halon suppression system is 5.0E-2.

If the automatic Halon suppression system fails, it is conservatively assumed here that all control room functions become damaged and shutdown from outside the control room using AOP 20.000.18 is required. A conditional core damage probability of 1.0E-2 is assumed in this case. This value is based on an assumed screening human error probability of 0.1 for shutdown outside the control room.

Therefore, the core damage frequency estimate for Scenario #1 is calculated as follows:

$F_{hw1} = 8.4E\text{-}4 \times 0.25 \times 5.0E\text{-}2 \times 5.0E\text{-}2 \times 1.0E\text{-}2 = 5.25E\text{-}9/yr$

Note that this calculation conservatively assumes that transient combustibles may be located in the area.

Scenario #2

The core damage frequency for Scenario #2 is estimated as follows:

$F_{hw2} = F_{ihw} \times u \times HEP_{hw} \times P_{sfs} \times CCDP$, where:

F_{hw2}	=	core damage frequency of Scenario #2 (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.

u	=	probability of hot work performed in close enough proximity to important cable targets such that the possibility exists for cable ignitions
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{sfs}	=	probability of successful fire suppression failure
CCDP	=	Conditional Core Damage Probability

Per Attachment 10.3 of the FIVE methodology, the failure probability of a Halon suppression system is 5.0E-2; therefore, the reliability of the system is 0.950.

The successful actuation of the Halon system is appropriately considered effective in limiting fire induced damage such that shutdown from the control room can still be effectively performed. To address the postulated initial fire damage, an assumed conditional core damage probability of 1.0E-2 is used. This is a conservative value based on review of other initial quantification runs (e.g., quantification runs involving multiple trains of equipment damage typically result in CCDPs in the mid E-3 range).

Therefore, the core damage frequency estimate for Scenario #2 is calculated as follows:

F_{hw2} = 8.4E-4 x 0.25 x 5.0E-2 x 0.950 x 1.0E-2 = 9.98E-8/yr

Note that this calculation conservatively assumes that transient combustibles may be located in the area.

Summary

The total core damage frequency contribution for postulated fires in the cable spreading room is the sum of the estimates for Scenario #1 and Scenario #2, as calculated above. The resulting core damage frequency contribution is 1.05E-7/yr, which is below the 1E-6/yr FIVE screening criterion.

4B.2.3.10 Cable Tray Area - 08AB

Like the cable spreading room, an initial screening quantification of the cable tray area was not performed as a core damage frequency orders of magnitude above the 1E-6/yr FIVE screening criterion was expected.

Walkdown of the cable tray area and review of the base fire ignition frequency calculations show that there are no fixed ignition sources in the room. The combination of the lack of fixed ignition sources and the fire ignition control measures prescribed by NPP-FP1-01 are considered adequate to preclude a credible fire event from damaging cables in the cable tray area. However, in order to develop a fire risk screening value, further analysis is presented below.

Ignition Frequency

The base fire ignition frequency calculation for the cable tray area identified the following contributors:

- Cable fires due to hot work: 1.2E-4/yr

- Transient combustible fires due to hot work: 7.2E-4/yr
- Transient ignition sources: 2.1E-4/yr

Like the analysis discussed earlier for the cable spreading room, the use of extension cords and heaters in the cable tray room is not considered credible; therefore, transient ignited fires are not quantified. In addition, although considered unlikely, the hot work ignition source is conservatively maintained for analysis.

Fire Damage Scenarios

Hot work is assum[illegible] esult in cable fires and cable damage if procedural controls are not followed and the work is being [illegible]formed in close enough proximity to important cable targets to cause ignition. No deterministic fire modeling is performed to define the conditions necessary for accidental hot work induced cable ignition. Rather, the critical distance is based on trash container fire modeling results.

The cable tray room is provided with a full coverage automatic carbon dioxide suppression system.**[4.3]** Therefore, two accident scenarios can be postulated:

- Scenario #1: Hot work fire with unsuccessful suppression
- Scenario #2: Hot work fire with successful suppression

Accident Sequence Quantification

Scenario #1

The formula for calculating the hot work induced core damage frequency for this area is the same as that shown for Scenario #1 for the cable spreading room.

Per Attachment 10.3 of the FIVE methodology, the failure probability of a CO2 suppression system is 4.0E-2.

Therefore, the core damage frequency estimate for Scenario #1 for the cable tray room is calculated as follows:

F_{hw1} = 8.4E-4/yr x 0.25 x 5.0E-2 x 4.0E-2 x 1.0E-2 = 4.20E-9/yr

Note that this calculation conservatively assumes that transient combustibles may be located in the area. In addition, the 1E-2 CCDP takes credit for the 3M brand 1-hour fire barrier material on the Dedicated Shutdown circuit that passes through this room.

Scenario #2

The formula for calculating the hot work induced core damage frequency for this area is that same as that shown for Scenario #2 for the cable spreading room.

Per Attachment 10.3 of the FIVE methodology, the failure probability of a CO2 suppression system is 4.0E-2; therefore, the reliability of the system is 0.960.

Therefore, the core damage frequency estimate for Scenario #1 for the cable tray room is calculated as follows:

F_{hw2} = 8.4E-4/yr x 0.25 x 5.0E-2 x 0.960 x 1.0E-2 = 1.01E-7/yr

Note that this calculation conservatively assumes that transient combustibles may be located in the area. In addition, the 1E-2 CCDP takes credit for the 3M brand 1-hour fire barrier material on the Dedicated Shutdown circuit that passes through this room.

Summary

The total core damage frequency contribution for postulated fires in the cable tray room is the sum of estimates for Scenario #1 and Scenario #2, as calculated above. The resulting core damage frequency contribution is 1.05E-7/yr, which is below the 1E-6/yr FIVE screening criterion.

4B.2.3.11 Control Room Complex - 09AB

Like the cable spreading room and the cable tray area, an initial screening quantification of the Control Room Complex was not performed as a core damage frequency orders of magnitude above the 1E-6/yr FIVE screening criterion was expected.

As the Control Room is continuously staffed, the best estimate fire is one that will be detected immediately by personnel and extinguished. A fire in the Control Room will be detected by ionization and photoelectric detectors above the drop ceiling, ionization and heat detectors in the peripheral rooms, ionization detectors behind the control room panels, ionization detectors within the control board panels, and by operators continuously staffing the Control Room.

The principal ignition sources in the Control Room are contained in the panels and cabinets. A walkdown of the Control Room was performed to study the functions of each control panel and to determine which panels are enclosed such that a fire would not spread from one panel to another. Bulkheads are incorporated into the main control panel design specifically for the purpose of preventing fire spread. Damage to adjacent control panels separated by a bulkhead wall is prevented by the fire bulkhead wall and the open tops of the panels which allow hot gases to escape upward. Damage is assumed in adjacent control panels for panels not separated by a fire bulkhead.

Therefore, the Control Room was analyzed as individual panel fire scenarios (this is consistent with guidance provided in the EPRI Fire PRA Implementation Guide). The fire ignition frequency per panel was determined by dividing the total Control Room fire ignition frequency, as determined by the FIVE methodology, by the total number of individual panels and cabinets. This is considered conservative for the following reasons:

Fire protection panels and emergency lighting are included in the panel ignition frequency and, thus, are implicitly assumed to be ignition sources resulting in credible core damage scenarios. However, as discussed earlier for fire modeling in other areas, fire protection panels and emergency lights can be appropriately dismissed from further analysis.

Hot work is included in the panel ignition frequency. Hot work in the Control Room while at power is not considered credible.

Transient ignited fires are included in the panel ignition frequency without the benefit of reduced frequency that would result from the analysis of critical combustible loading and the probability of proximity to exposed targets.

Each postulated panel fire scenario was analyzed as proceeding along either of two distinct accident sequence paths:

- Manual suppression by operators successful

- Manual suppression by operators unsuccessful

If manual suppression by operators is successful, then the panel fire is quantified using the fire PSA models, with all functions controlled by the panel disabled. However, the control room is not abandoned. If manual suppression by operators is unsuccessful, then the quantification considers the following:

- Evacuation of the Control Room
- Shutdown of the plant using the dedicated shutdown panel

Panel Fires With Successful Suppression

The core damage frequency of each panel fire scenario with successful suppression is calculated as follows:

(Panel Ignition Frequency) x (Successful Suppression) x (Conditional Core Damage Probability)

Forty-seven (47) panels and cabinets are located in the Control Room Complex. The overall Control Room fire ignition frequency is 1.09E-2/yr; therefore, the fire ignition frequency, F_i, per panel is calculated as 2.32E-4/yr.

Due to continuous staffing of the Control Room, the probability of successful and timely suppression of a panel fire is estimated as highly reliable.

The conditional core damage probability (CCDP) per panel fire is determined based on quantification of the fire PSA models. Fourteen (14) of the 47 panels and cabinets in the Control Room contain controls of equipment credited in the PSA models; these are the main control panels. These fourteen control panels (six of which are combined into two sets of panels due to the lack of fire barrier bulkheads) are listed below with the associated modeled failures:

PANEL	RUN ID	MODELED FAILURES
H11-P602 and -P812	9AB01	MC, SRVs II, LPCS II, RHR II, HPCI, RBCCW, EECW/EESW II, NIAS II, EDGs 13 and 14
H11-P603, -P804, -P805, -P813	9AB02	MC, RPT I & II, SLC I & II, TBCCW, Condensate, M/U to CSTs
H11-P806	BASE	Main condenser (MC)
H11-P809	9AB04	Div. I 4kV buses, EDG 11, EDG 12
H11-P810	9AB05	Div. II 4kV buses, EDG 13, EDG 14

H11-P811	9AB06	Offsite power supply
PANEL	RUN ID	MODELED FAILURES
H11-P807	9AB07	MC, RHRSW I & II, NIAS I & II, Station Air
H11-P808	9AB08	Hard pipe containment vent
H11-P817	9AB08	Hard pipe containment vent
H11-P601	3AB11	MC, ADS/SRV I, LPCS I, RHR I, NIAS I, RCIC, EECW/EESW I, SBFW, EDGs 11 and 12

A fire in any of the remaining 33 cabinets would not directly damage safe shutdown or alternate safe shutdown equipment. All these other cabinets were conservatively assumed to result in loss of the main condenser. The conditional core damage probability for each of these remaining cabinets is based on run BASE, 1.3E-5.

The results of the successfully suppressed panel fire scenarios are initially calculated assuming that even with successful suppression all functions in the panel are disabled. This assumption does not take credit for manual suppression successfully preventing failure of redundant controls on a panel nor does it take credit for the fact that all postulated control board fires do not consume the entire panel. These initial panel quantifications are summarized below:

Panel	Fi		Suppression Probability		CCDP	Run ID		CDF
-P602, -P812	4.64E-4	x	~ 1.0	x	1.0E-2	9AB01	=	4.64E-6/yr
-P603, -P804, -P805,-P813	9.28E-4	x	~ 1.0	x	2.4E-4	9AB02	=	2.32E-7/yr
-P806	2.32E-4	x	~ 1.0	x	1.3E-5	9AB03	=	3.01E-9/yr
-P809	2.32E-4	x	~ 1.0	x	4.3E-3	9AB04	=	9.97E-7/yr
-P810	2.32E-4	x	~ 1.0	x	2.9E-3	9AB05	=	6.73E-7/yr
-P811	2.32E-4	x	~ 1.0	x	8.0E-4	9AB06	=	1.86E-7/yr
-P807	2.32E-4	x	~ 1.0	x	1.5E-3	[1]	=	3.48E-7/yr

[1]A fire in panel -P807, if assumed to disable all functions in the panel, would result in loss of the main condenser, hard pipe vent, and RHRSW. As such, the core damage frequency for this panel fire is composed entirely of loss of decay heat removal accident sequences. Approximately 24 hours are available before the onset of core damage for an operator to locally operate

-P808	2.32E-4	x	~ 1.0	x	1.2E-5	9AB08	=	2.78E-9/yr
-P817	2.32E-4	x	~ 1.0	x	1.2E-5	9AB08	=	2.78E-9/yr
-P601,	2.32E-4	x	~ 1.0	x	3.0E-3	3AB11	=	6.96E-7/yr
Remaining								
Panels	7.65E-3	x	~ 1.0	x	1.3E-5	BASE	=	9.95E-8/yr
				TOTAL			=	7.89E-6/yr

This core damage frequency estimate is conservative. As stated above, this initial quantification assumes failure of all functions on a control panel even with successful suppression. This is conservative for the following reasons:

- The control panel fires in the EPRI Fire Events Database do not appear to have resulted in significant damage. The recorded fires have self-extinguished or been suppressed before damage to redundant controls in the panel occurred. **[4.32]**
- The control panel fires do not take into account the design of the control boards. A typical main control board is subdivided into a few discrete areas (see Figure 4-4). Fires initiated in non-critical zones (e.g., annunciator panel) are not expected to result in damage to critical controls.
- Some functions on the main control board panels are assumed failed due to the implicit assumption of a hot short (i.e., shorting of two wires, one of which is powered, such that a circuit is made up). The hot short failure applies to those functions in which an active failure must occur in order to fail the function (e.g., hot short initiates closure of normally open LPCI injection valve). Studies have shown that the creation of hot shorts during a control panel fire is difficult. **[4.34, 4.35]** For a hot short to be successful, the correct wires must have the insulation melted off and the wires melted or pressed together, and the circuit fuses and other protection devices must not be damaged by the fire and must continue to function normally.

If each of these issues were addressed explicitly in the evaluation of each of the main control boards, the overall core damage frequency due to suppressed panel fires is judged to be reduced significantly (approximately an order of magnitude). However, no guidance is provided in the FIVE methodology or the <u>EPRI Fire PRA Implementation Guide</u> as to the details of such an analysis. As such, the core damage frequency due to unsuppressed panel fires is conservatively multiplied by a factor of 0.5. Therefore, the core damage frequency due to unsuppressed Control Room panel fires is conservatively estimated at 3.94E-6/yr.

<u>Panel Fires With Unsuccessful Suppression (Control room evacuation required)</u>

The core damage frequency of each panel fire with unsuccessful suppression is calculated as follows:

(Panel Ignition Frequency) x (Unsuccessful Suppression) x (Shutdown Probability Outside CR)

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breakers or valves to regain RHRSW or the hard pipe vent. An HEP of 1.5E-3, based on the EPRI cause-based human reliability analysis method, is assigned for the operator failing to recover decay heat removal within 24 hours. **[4.14]**

As stated earlier, failure of immediate suppression by the operators is judged to be of a low likelihood. Credit is appropriately taken in this analysis for the continuous staffing of the Control Room by personnel whose job function stresses vigilance. The probability for manual suppression failure is estimated here at 3E-3/panel fire. The 3E-3 value for failure to suppress is consistent with the EPRI Fire Implementation Guide best estimate for failure to suppress a Control Room panel fire within 15 minutes.

If the fire is not extinguished, it is assumed the operators leave the Control Room, with a probability of 1.0, and transfer control to the remote shutdown panels. This is conservative, as it assumes a large panel fire that generates a large amount of smoke and heat that would make the Control Room completely uninhabitable despite the use of self contained breathing apparatus. A conditional core damage probability estimate of 1E-2/yr is applied for failure to achieve safe shutdown using the dedicated shutdown panel given a panel fire and failure to suppress the fire in a sufficiently timely fashion to remain in the control room.

The summed contribution of the individual unsuppressed panel fire scenarios is as follows:

Panels		Fi		Failure to Suppress		CCDP	CDF
47	x	2.32E-4	x	3E-3	x	1E-2 =	3.27E-7/yr

The summed core damage frequency contribution of the individual Control Room panel fire scenarios is as follows:

- Successful suppression cases: 3.94E-6/yr
- Unsuccessful suppression cases: 3.27E-7/yr
- TOTAL: 4.27E-6/yr

The Control Room Complex remains above the FIVE 1E-6/yr screening criterion.

4B.2.3.12 Misc. Rooms (Div. I Area) - 11ABE

The initial screening quantification of this compartment resulted in a core damage frequency higher than the 1E-6/yr FIVE screening criterion. Therefore, further analysis was performed to reduce the conservatism in the initial screening quantification.

Ignition Frequency

A review of the base ignition frequency calculation for this area shows the following contributors to the fire ignition frequency:

- Five (5) emergency lights: 2.2E-4/yr
- Ten (10) fire protection panels: 1.2E-4/yr
- Six (6) battery chargers: 1.4E-3/yr
- Two (2) cooling unit components: 5.7E-5/yr
- One (1) Division I SRV cabinet: 4.4E-5/yr
- One (1) Division II SRV cabinet: 4.4E-5/yr

- MCC 2PA-1: 9.3E-4/yr
- MCC 2PB-1: 1.3E-3/yr
- Distribution cabinet 2PA-2: 8.4E-4/yr
- Two (2) RPS MG sets: 5.5E-3/yr
- Five (5) miscellaneous cabinets: 2.2E-4/yr
- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr
- Transient ignition sources: 2.1E-4/yr

As discussed earlier, the credible fires in either the fire protection panels or the emergency lights would be small and confined to within the boundaries of the unit. In addition, these postulated fires would quickly extinguish due to limited combustibles and would realistically not result in a plant trip. Therefore, the fire protection panels and the emergency lights are eliminated from further analysis.

Standby battery chargers R3200S025 and R3200S020C can be eliminated from further analysis as they are deenergized and must be manually aligned.

The remaining ignition sources are maintained for further analysis.

Fire Damage Scenarios

The potential fire damage scenarios, based on review of the ignition sources, as described above, include:

- Battery charger fire
- Cooling unit fire
- Div. I SRV cabinet fire
- Div. II SRV cabinet fire
- MCC 2PA-1 fire
- MCC 2PB-1 fire
- Distribution cabinet 2PA-2 fire
- RPS MG Set fire
- Miscellaneous panel fire
- Transient ignited fire
- Cable/transient fire due to hot work

<u>**Battery Charger Fire**</u>

Six battery chargers are located at the southwest end of compartment 11ABE:

- 24V DC charger 2IA-1 (R3200S023A)
- 24V DC charger 2IA-2 (R3200S023B)

- 24V DC standby charger AB (R3200S025)
- 130V DC charger 2A-1 (R3200S020A)
- 130V DC charger 2A-2 (R3200S020B)
- 130V DC charger 2A-1-2 (R3200S020C)

The 24V DC chargers are small units that are located five (5) feet or more from any other important equipment (e.g., the 130V DC chargers are 5 feet across the corridor). There are no cable trays overhead. In addition, as discussed above, the 24V DC standby charger is eliminated from further analysis as a potential fire source. A postulated fire in either of the other two 24V DC chargers would result in only consuming and failing the individual charger. Conservatively, it is assumed here that a fire in either of the two chargers would disable all three 24V DC battery chargers. As the 24V DC system supplies power for primarily neutron monitoring, it is assumed that failure of the chargers would lead to a plant trip with the loss of the main condenser.

The 130V DC chargers, as stated above, are located approximately five (5) feet across the corridor from the 24V DC chargers. The standby charger, 2A-1-2, is deengergized and must be manually aligned. Therefore, the 130V DC standby charger is eliminated from further analysis as a potential fire source.

The nearest component to the 130V DC chargers is Division I DC distribution cabinet 2PA-2. 2PA-2 is actually two (2) connected steel cabinets located approximately 4' 6" to the west of the nearest charger (R3200S020A). A conservatively high peak fire intensity of 500 Btu/sec. (larger than the peak fire intensity of a 32-gallon trash fire) would be required to achieve a critical radiant flux distance of 4'-6". Therefore, failure of 2PA-2 due to a battery charger fire is not indicated by deterministic fire modeling.

Located directly above the chargers are cable trays 1C-101 and 1P-061/DC1P-061. The 1C-101 cable tray is located approximately 2' 6" above the chargers; 1P-061/DC1P-061 is located approximately 4' above the chargers. Tray 1C-101 can be expected to sustain cable damage due to a postulated high intensity charger fire. The other tray is far enough away such that cable damage is not indicated by fire modeling.

Cable tray 1C-101 contains Division I DC cables. Failure of this tray can be conservatively modeled as leading to failure of all Division I DC power.

Cooling Unit Fires

The primary ignition source for a cooling unit fire is a motor-driven grease lubricated fan (T4100B043). This fan unit is suspended from the ceiling approximately 5 feet away from MCC 2PA-15. There are no cable trays within five (5) feet of this fan motor. As the fan motor fire would be a motor windings fire inside the motor housing, the fan motor fire is best modeled using the FIVE methodology radiant exposure approach. An overly conservative peak fire intensity (approximately 400 Btu/sec) would be required to create a critical radiant flux distance of five feet. Therefore, deterministic fire modeling does not indicate that damage will occur to any non-BOP equipment (BOP targets were not investigated) due to a cooling fan fire.

Division I SRV Cabinet Fire

This cabinet is an enclosed steel cabinet and is expected to contain the effects of a fire. The impact of the fire is expected to result in no more than a trip with loss of the Division I SRV function.

Division II SRV Cabinet Fire

This cabinet is an enclosed steel cabinet and is expected to contain the effects of a fire. The impact of the fire is expected to result in no more than a trip with loss of the Division II SRV function.

MCC 2PA-1 Fire

MCC 2PA-1 is located in the southeast corner of compartment 11ABE. The nearest equipment consists of cable trays, a cooling unit, and a Division II MCC. The cooling unit, as discussed above, is not within a critical distance.

Cable trays 1C-101 and 1P-061/DC1P-061 pass directly overhead of MCC 2PA-1 at a distance of approximately 2' 6" above the top of the MCC. These trays can be expected to sustain cable damage during a postulated fire of MCC 2PA-1. Trays 1C-101 and 1P-061/DC1P-061 contain Division I DC cables. Therefore, damage to these trays can be conservatively modeled as failure of all Division I DC.

Nearby MCC 2PB-1 is approximately 4' 6" to the south of MCC 2PA-1. A radiant heat shield exists between these two MCCs. Without crediting the radiant heat shield, MCC 2PB-1 is sufficiently distant such that it is not expected to be damaged due to a fire in MCC 2PA-1. An overly conservative peak fire intensity (500 Btu/sec) is required to result in a failure of MCC 2PB-1 given a postulated fire in MCC 2PA-1. Therefore, fire modeling does not indicate that damage to MCC 2PB-1 can be expected.

Other nearby targets are cable trays containing Division II cables (2C-075 and 2P-075) and trays containing RCIC cables (1P-060, 1C-075, 1C-122). The Division II cable trays are wrapped in 3-M fire barrier material. This material is not credited as it is not maintained. However, all these cable trays are sufficiently distant (approximately six feet or more) from MCC 2PA-1 such that damage to cables is not expected.

Therefore, the expected impacts from a fire in MCC 2PA-1 can be conservatively bounded by assuming failure of all Division I DC.

MCC 2PB-1 Fire

MCC 2PB-1 is located in the southeast corner of compartment 11ABE. The nearest equipment consists of cable trays and a Division I MCC.

Nearby MCC 2PA-1 is approximately 4' 6" to the north of MCC 2PB-1. The discussion provided above for MCC 2PA-1 applies here and indicates that these two MCCs are sufficiently separated such that fire induced damage to the second MCC is not expected.

Other nearby targets are cable trays containing Division II cables (2C-075 and 2P-075) and trays containing RCIC cables (1P-060, 1C-075, 1C-122). The nearest is tray 1P-061/DC1P-061, at a line of sight distance of approximately 4'. This tray is sufficiently distant such that damage to the contained cables is not expected. Other trays are more distant and, therefore, are also not expected to be damaged.

Therefore, a fire in MCC 2PB-1 is expected to result in failure of just the MCC functions. This can be conservatively modeled by assuming failure of all Division II DC.

Cabinet 2PA-2 Fire

Distribution cabinet 2PA-2 is located 4' 6" to the west of the Division I DC battery chargers, as discussed above. A fire in this cabinet is not expected to result in failure of the nearby chargers. However, failure of a charger and 2PA-2 would result in the same impact as just failure of 2PA-2 (i.e., conservatively assume Division I DC disabled)

Cable tray 1P-061/DC1P-061 passes directly overhead of 2PA-2 at a height of approximately 2' 6" above the top of the cabinet. This tray can be postulated to sustain damage during a cabinet fire in 2PA-2. However, this tray contains the same vulnerabilities as the cabinet itself.

Therefore, a fire in 2PA-2 can be conservatively modeled by assuming failure of all Division I DC.

RPS MG Set Fire

RPS MG sets C7102S001A and C7102S001B are located in separate concrete rooms with steel doors. There are no safe shutdown cables or equipment located in these two concrete rooms. A fire in either of these two rooms is not expected to propagate outside the boundaries of the room. These fires can be appropriately modeled as assuming a plant trip and loss of the main condenser.

Miscellaneous Panel Fires

Fires in the miscellaneous panels are expected to remain confined to these panels. These panel fires can be conservatively modeled as resulting in a plant trip with loss of the main condenser and the hard pipe vent.

Transient Ignited Fires

A walkdown of the area did not identify any transient ignition sources or combustible materials. However, to bound the transient combustible fire scenario, a trash container fire is conservatively postulated in the room. This fire was modeled with the QUICK FIVE software as having a duration of 5 minutes and a peak fire intensity of 380 Btu/sec; this is consistent with fire modeling examples in the FIVE methodology. The height of the point source trash container fire was assumed to be 4 feet, consistent with the FIVE methodology.

The critical height for the Target-In-Plume case extends to the ceiling and the critical radiant flux distance is approximately 3.4 feet. Therefore, the analysis needs to address the likelihood of a transient combustible placed under cable trays or near important equipment.

Hot Work Fires

Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets or transient combustibles to cause ignition.

Accident Sequence Quantification

Battery Charger Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to fixed ignition sources:

$F_{fx} = F_{if} \times P_{fs} \times CCDP$, where:

F_{fx} = core damage frequency (due to fixed sources)

F_{if} = fire ignition source frequency (due to fixed sources)

P_{fs} = probability of fire suppression failure

CCDP = Conditional Core Damage Probability

No credit is taken here for fire suppression or fire brigades.

Fire modeling for the 24V DC chargers indicates that fire induced failure of the chargers is best modeled as leading to a plant trip with loss of the main condenser. The conditional core damage probability for this event is based on an existing run (15AB3, CCDP=3.3E-5 - see Table 4-4).

Therefore, the core damage frequency for the 24V DC chargers is calculated as follows (note that the standby charger is eliminated as it is deenergized):

F = (2/6) x 1.4E-3 x 1.0 x 3.3E-5 = 1.54E-8/yr

Fire modeling for the 130V DC chargers indicates that fire induced failure of the chargers can be conservatively modeled as leading to loss of all Division I DC power. The conditional core damage probability for this event is based on an existing run (10AB1, CCDP=4E-4 - see Table 4-4).

Therefore, the core damage frequency for the 130V DC chargers is calculated as follows (note the standby charger is eliminated as it is deenergized):

F = (2/6) x 1.4E-3 x 1.0 x 4E-4 = 1.86E-7/yr

Cooling Unit Fires

The fire modeling performed for the cooling unit equipment does not indicate damage to any nearby non-BOP cabling or equipment. However, the effect on BOP cabling was not evaluated. Therefore, the core damage frequency estimate for this ignition contributor is conservatively modeled as leading to a plant trip with loss of the main condenser and loss of the hard-piped vent. The conditional core damage probability associated with this damage is conservatively estimated at 2E-4/yr (rather than perform an additional run, this CCDP is conservatively based on an existing run, 14AB2, that includes failure of the main condenser, the hard-piped vent and RCIC).

Therefore, the core damage frequency for this area due to a cooling unit fan is conservatively estimated as follows:

5.7E-5 x 1.0 x 2.0E-4 = 1.14E-8/yr

Division I SRV Cabinet Fire

The fire modeling performed for this panel indicates that this panel fire will not damage other equipment, and can best be modeled as a plant trip with loss of the Division I SRV function. The conditional core damage probability for this fire is based on an existing run (03AB4, CCDP=8.6E-6/yr)

Therefore, the core damage frequency associated with a fire in this cabinet is calculated as follows:

4.4E-5 x 1.0 x 8.6E-6 = 3.78E-10/yr

Division II SRV Cabinet Fire

The fire modeling performed for this panel indicates that this panel fire will not damage other equipment, and can best be modeled as a plant trip with loss of the Division II SRV function. The conditional core damage probability for this fire is based on an existing run (03AB4, CCDP=8.6E-6/yr)

Therefore, the core damage frequency associated with a fire in this cabinet is calculated as follows:

4.4E-5 x 1.0 x 8.6E-6 = 3.78E-10/yr

MCC 2PA-1 Fire

The fire modeling for MCC 2PA-1 indicates that damage to nearby Division I DC trays can be expected due to a postulated fire in 2PA-1. The scenario can be conservatively modeled by assuming failure of all Division I DC. The conditional core damage probability for this fire is based on an existing run (10AB1, CCDP=4E-4 - see Table 4-4).

Therefore, the core damage frequency for this area due to a fire in MCC 2PA-1 is conservatively estimated as follows:

9.3E-4 x 1.0 x 4E-4 = 3.70E-7/yr

MCC 2PB-1 Fire

The fire modeling for MCC 2PB-1 indicates that damage to nearby trays and equipment is not expected due to a postulated fire in 2PB-1. However, the failure of MCC 2PB-1 can, by itself, be conservatively modeled as resulting in failure of all Division II DC. The conditional core damage probability for this fire is based on an existing run (10AB2, CCDP=3.3E-4 - see Table 4-4).

Therefore, the core damage frequency for this area due to a fire in MCC 2PB-1 is conservatively estimated as follows:

1.3E-3 x 1.0 x 3.3E-4 = 4.29E-7/yr

Cabinet 2PA-2 Fire

The fire modeling for cabinet 2PA-2 indicates that a postulated fire in 2PA-2 can be conservatively modeled by assuming failure of all Division I DC. The conditional core damage probability for this fire is based on an existing run (10AB1, CCDP=4E-4 - see Table 4-4).

Therefore, the core damage frequency for this area due to a fire in cabinet 2PA-2 is conservatively estimated as follows:

8.4E-4 x 1.0 x 4E-4 = 3.36E-7/yr

Miscellaneous Panel Fires

Fires in these panels are expected to remain confined within the panels. The panels are quantified collectively with a conditional core damage probability assuming a plant trip with loss of the main condenser and the hard pipe vent. The core damage probability is based on and existing run (14AB2, CCDP=2E-4 - see Table 4-4).

Therefore, the core damage frequency for this area due to a miscellaneous panel fire is conservatively estimated as follows:

2.2E-4 x 1.0 x 2E-4 = 4.40E-8/yr

RPS MG Set Fires

The fire modeling performed for these units indicates that they can be modeled collectively as leading to a plant trip and loss of the main condenser. The conditional core damage probability associated with this damage is based on an existing run (15AB3, CCDP=3.3E-5 - see Table 4-4).

Therefore, the core damage frequency for this area due to an MG set fire is calculated as follows:

5.5E-3 x 1.0 x 3.3E-5 = 1.82E-7/yr

Transient Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to transient combustible sources:

$F_t = F_{it} \times u \times p \times w \times P_{fs} \times CCDP$, where:

F_t	=	core damage frequency (due to transients)
F_{it}	=	fire ignition source frequency (due to transients)
u	=	transient combustible located in the range of target components
p	=	probability of combustible being exposed
w	=	frequency of finding a critical combustible loading versus the frequency of inspecting and removing the transient combustible fire source.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters u, P_{fs} and p are conservatively set to 1.0.

Per the FIVE methodology, w is calculated as follows:

$w = (x/2) \times \ln(1/x)$, where $x = F_{ccl}/F_w$

F_{ccl}	=	The frequency of having a critical combustible loading present. FIVE provides a screening value of 1 event/year/compartment.
F_w	=	Frequency of combustible material inspections that would find the transient combustible fire source before a fire occurred.

The value of 1 event/year is used here for F_{ccl}.

Administrative procedure NPP-FP1-01 specifies that: 1) monthly fire protection inspections are to be performed, and 2) an hourly compensatory fire watch is to tour all areas appearing on the Out of Specification Log Sheet. This same procedure states that the accumulation of trash, oil rags, combustible materials, and similar fire hazards is prohibited. It is judged that either of the above tours would identify and correct a trash container inappropriately placed near a cable tray riser. As the hourly compensatory fire watch may not tour this area hourly or every day, a conservative estimate of one tour per week is assumed (F_w = 52/yr). Therefore, w for this area is estimated as follows:

$w = ((1/52)/2) \times \ln(1/(1/52)) = 3.79E\text{-}2$

A bounding conditional core damage probability, given failure of the suppression system, can be conservatively estimated using the initial screening quantification run (11AB1, CCDP=6.5E-3 - see Table 4-4).

Therefore, the core damage frequency for this area due to transient combustible fires is conservatively estimated as follows:

$F_t = 2.1E\text{-}4 \times 1.0 \times 1.0 \times 3.79E\text{-}2 \times 1.0 \times 6.5E\text{-}3 = 5.17E\text{-}8/yr$

Hot Work Fires

Hot work induced core damage frequency is estimated as follows:

$F_{hw} = F_{ihw} \times u \times HEP_{hw} \times P_{fs} \times CCDP$, where:

F_{hw}	=	core damage frequency (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.
u	=	probability of hot work performed in close enough proximity to important cable targets or transient combustibles such that the possibility exists for ignition
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters P_{fs} and u are conservatively set to 1.0.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

A bounding conditional core damage probability, given failure of the suppression system, can be conservatively estimated using the initial screening quantification runs (11AB1, CCDP=6.5E-3 - see Table 4-4).

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

F_{hw} = 8.4E-4 x 1.0 x 5.0E-2 x 1.0 x 6.5E-3 = 2.73E-7/yr

Summary

Summing the individual fire scenario core damage frequencies, the total core damage frequency due to fire in this area is conservatively estimated at 1.90E-6/yr. This compartment remains above the 1E-6/yr FIVE screening criterion.

4B.2.3.13 Misc. Rooms (Div. II Battery Charger Area) - 11ABW

The initial screening quantification of this compartment resulted in a core damage frequency higher than the 1E-6/yr FIVE screening criterion. Therefore, further analysis was performed to reduce the conservatism in the initial screening quantification.

Ignition Frequency

A review of the base ignition frequency calculation for this area shows the following contributors to the fire ignition frequency:

- Five (5) emergency lights: 2.2E-4/yr
- Two (2) fire protection panels: 2.4E-5/yr
- Five (5) battery chargers: 1.2E-3/yr
- Distribution cabinet 2PB-2: 7.9E-4/yr
- One (1) miscellaneous cabinet: 4.4E-5/yr
- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work 7.2E-4/yr
- Transient ignition sources: 2.1E-4/yr

As discussed earlier, the credible fires in either the fire protection panels or the emergency lights would be small and confined to within the boundaries of the unit. In addition, these postulated fires would quickly extinguish due to limited combustibles and would realistically not result in a plant trip. Therefore, the fire protection panels and the emergency lights are eliminated from further analysis.

Standby battery charger R3200S021C can be eliminated from further analysis as it is deenergized and must be manually aligned.

The remaining ignition sources are maintained for further analysis.

Fire Damage Scenarios

The potential fire damage scenarios, based on review of the ignition sources, as described above, include:

- Battery charger fire
- Distribution cabinet 2PB-2 fire
- Miscellaneous panel fire
- Transient ignited fire
- Cable/transient fire due to hot work

Battery Charger Fire

Six battery chargers are located in compartment 11ABW:

- 24V DC charger 2IB-1 (R3200S024A)
- 24V DC charger 2IB-2 (R3200S024B)
- 130V DC charger 2B-1 (R3200S021A)
- 130V DC charger 2B-2 (R3200S021B)
- 130V DC charger 2B-1-2 (R3200S021C)

The 24V DC chargers are small units that are located five (5) feet or more from any other important equipment (e.g., the 130V DC chargers are 5 feet in front across the corridor). There are no cable trays overhead. In addition, as discussed above, the 24V DC standby charger is eliminated from further analysis as a potential fire source. A postulated fire in either of the other two 24V DC chargers would result in only consuming and failing the individual charger. Conservatively, it is assumed here that a fire in either of the two chargers would disable all three 24V DC chargers. As the 24V DC system supplies power for primarily neutron monitoring, it is assumed that failure of the chargers would lead to a plant trip with the loss of the main condenser.

The 130V DC chargers, as stated above, are located approximately five (5) across the corridor for the 24V DC chargers. The standby charger, 2A-1-2, is deengergized and must be manually aligned. Therefore, the 130V DC standby charger is eliminated from further analysis as a potential fire source.

There are no cable trays overhead or nearby. However, Division II DC distribution cabinet 2PB-2 is located approximately one foot to the east of battery charger R3200S021A. However, failure of both the distribution cabinet and a charger results in the same impact as just failure of a charger (i.e., conservatively assume Division II DC disabled).

Therefore, a 130V DC charger fire can be conservatively modeled by assuming failure of all Division II DC.

Cabinet 2PB-2 Fire

Distribution cabinet 2PB-2 is located approximately one foot to the east of 130V DC battery charger R3200S021A. As discussed above for the chargers, failure of both the distribution cabinet and a charger results in the same impact as failure of just a charger or the distribution cabinet (i.e., assume Division II DC disabled).

Therefore, a fire in 2PB-2 can be conservatively modeled by assuming failure of all Division II DC.

Miscellaneous Panel Fire

This single panel is a fuse panel for the 24V DC battery. Like the 24V DC chargers, this panel can be modeled as a plant trip with loss of the main condenser and the hard pipe vent. Fire in this panel is expected to remain confined within the panel.

Transient Ignited Fires

A walkdown of the area did not identify any transient ignition sources or combustible materials. However, to bound the transient combustible fire scenario, a trash container fire is conservatively postulated in the room. This fire was modeled with the QUICK FIVE software as having a duration of 5 minutes and a peak fire intensity of 380 Btu/sec; this is consistent with fire modeling examples in the FIVE methodology. The height of the point source trash container fire was assumed to be 4 feet above the floor, consistent with the FIVE methodology.

The critical height for the Target-In-Plume case extends to the ceiling and the critical radiant flux distance is approximately 3.4 feet. Therefore, the analysis needs to address the likelihood of a transient combustible placed under cable trays or near important equipment.

Hot Work Fires

Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets or transient combustibles to cause ignition.

Accident Sequence Quantification

Battery Charger Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to fixed ignition sources:

$F_{fx} = F_{if} \times P_{fs} \times CCDP$, where:

F_{fx}	=	core damage frequency (due to fixed sources)
F_{if}	=	fire ignition source frequency (due to fixed sources)
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

This area is not equipped with an automatic fire suppression system. No credit is taken here for fire brigades.

Fire modeling for the 24V DC chargers indicates that fire induced failure of the chargers can be conservatively modeled as leading to a plant trip with loss of the main condenser. The conditional core damage probability for this event is based on an existing run (15AB3, CCDP=3.3E-5 - see Table 4-4).

Therefore, the core damage frequency for the 24V DC chargers is calculated as follows (note that the standby charger is eliminated as it is deenergized):

F = (2/5) x 1.2E-3 x 1.0 x 3.3E-5 = 1.58E-8/yr

Fire modeling for the 130V DC chargers indicates that fire induced failure of the chargers can be conservatively modeled as leading to loss of all Division II DC power. The conditional core damage probability for this event is based on an existing run (10AB2, CCDP=3.3E-4 - see Table 4-4).

Therefore, the core damage frequency for the 130V DC chargers is calculated as follows (note the standby charger is eliminated as it is deenergized):

F = (2/5) x 1.2E-3 x 1.0 x 3.3E-4 = 1.58E-7/yr

Cabinet 2PB-2 Fire

The fire modeling for cabinet 2PB-2 indicates that a postulated fire in 2PB-2 can be conservatively modeled by assuming failure of all Division II DC. The conditional core damage probability for this fire is based on an existing run (10AB2, CCDP=3.3E-4 - see Table 4-4).

Therefore, the core damage frequency for this area due to a fire in cabinet 2PB-2 is conservatively estimated as follows:

7.9E-4 x 1.0 x 3.3E-4 = 2.61E-7/yr

Miscellaneous Panel Fires

The panels are quantified collectively with a conditional core damage probability assuming a plant trip with loss of the main condenser and the hard pipe vent. The core damage probability is based on and existing run (14AB2, CCDP=2E-4 - see Table 4-4).

Therefore, the core damage frequency for this area due to a miscellaneous panel fire is conservatively estimated as follows:

4.4E-5 x 1.0 x 2E-4 = 8.80E-9/yr

Transient Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to transient combustible sources:

$F_t = F_{it} \times u \times p \times w \times P_{fs} \times CCDP$, where:

F_t	=	core damage frequency (due to transients)
F_{it}	=	fire ignition source frequency (due to transients)
u	=	transient combustible located in the range of target components
p	=	probability of combustible being exposed
w	=	frequency of finding a critical combustible loading versus the frequency of inspecting and removing the transient combustible fire source.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters P_{fs} and p are conservatively set to 1.0.

Due to the small size of the room (approximately 10' x 15'), the parameter u is appropriately set to 1.0.

Per the FIVE methodology, w is calculated as follows:

$w = (x/2) \times \ln(1/x)$, where $x = F_{ccl}/F_w$

F_{ccl}	=	The frequency of having a critical combustible loading present. FIVE provides a screening value of 1 event/year/compartment.
F_w	=	Frequency of combustible material inspections that would find the transient combustible fire source before a fire occurred.

The value of 1 event/year is used here for F_{ccl}.

Administrative procedure NPP-FP1-01 specifies that: 1) monthly fire protection inspections are to be performed, and 2) an hourly compensatory fire watch is to tour all areas appearing on the Out of Specification Log Sheet. This same procedure states that the accumulation of trash, oil rags, combustible materials, and similar fire hazards is prohibited. It is judged that either of the above tours would identify and correct a trash container inappropriately placed near a cable tray riser. As the hourly compensatory fire watch may not tour this area hourly or every day, a conservative estimate of one tour per week is assumed (F_w = 52/yr). Therefore, w for this area is estimated as follows:

$w = ((1/52)/2) \times \ln(1/(1/52)) = 3.79E-2$

A bounding conditional core damage probability, given failure of the suppression system, can be conservatively estimated using the initial screening quantification run (11AB3, CCDP=1.3E-3 - see Table 4-4).

Therefore, the core damage frequency for this area due to transient combustible fires is conservatively estimated as follows:

F_t = 2.1E-4 x 1.0 x 1.0 x 3.79E-2 x 1.0 x 1.3E-3 = 1.03E-8/yr

Hot Work Fires

Hot work induced core damage frequency is estimated as follows:

$F_{hw} = F_{ihw}$ x u x HEP_{hw} x P_{fs} x CCDP, where:

F_{hw}	=	core damage frequency (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.
u	=	probability of hot work performed in close enough proximity to important cable targets or transient combustibles such that the possibility exists for ignition
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameter P_{fs} is conservatively set to 1.0.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

Due to the small size of the room, the parameter u is appropriately set to 1.0.

A bounding conditional core damage probability can be conservatively estimated using the initial screening quantification run (11AB3, CCDP=1.3E-3 - see Table 4-4).

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

F_{hw} = 8.4E-4 x 1.0 x 5.0E-2 x 1.0 x 1.3E-3 = 5.46E-8/yr

Summary

Summing the individual fire scenario core damage frequencies, the total core damage frequency due to fire in this area is conservatively estimated at 5.09E-7/yr. This estimate is below the 1E-6/yr FIVE screening criterion. Therefore, this compartment is screened from further analysis.

4B.2.3.14 Division II SWGR Room - 12AB

The initial screening quantification of this compartment resulted in a core damage frequency over an order of magnitude higher than the 1E-6/yr FIVE screening criterion. This initial screening quantification is overly conservative.

Fire Ignition Frequency

A review of the base ignition frequency calculation for this area shows the following contributors to the fire ignition frequency:

- Three (3) fire protection panels: 3.7E-5/yr
- Six (6) emergency lights: 1.1E-4/yr
- Six (6) cooling unit components: 1.7E-4/yr
- 480V 72F bus, 4kV/480V transformer, & voltage regulator: 3.1E-4/yr
- 480V 72E bus, 4kV/480V transformer, & voltage regulator: 3.2E-4/yr
- 4160V 65F bus: 1.3E-4/yr
- 4160V 65E bus: 1.7E-4/yr
- 4160V 65G bus: 5.7E-5/yr
- 65T crosstie breaker cabinet: 1.9E-5/yr
- MPU 2: 4.0E-4/yr
- Two (2) MPU 2 voltage regulators: 1.9E-4/yr
- MCC 72F-2A: 2.7E-4/yr
- Cabinet 72E-4A: 2.5E-4/yr
- 120V distribution cabinet for MCC 2PB-1: 1.2E-4/yr
- 130V DC cabinet 2PB2-15: 2.1E-4/yr
- MCC 72C-F isolating contactor (Div. II): 1.9E-5/yr
- Miscellaneous electrical items: 1.6E-4/yr
- Transient ignition sources: 2.1E-4/yr
- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr

As discussed earlier, the fire protection panels and the emergency lights are eliminated from further analysis. The credible fires in either the fire protection panels or the emergency lights would be confined to within the boundaries of the unit and would realistically not result in a plant trip.

The remaining ignition sources are maintained for further analysis.

Fire Damage Scenarios

The potential fire damage scenarios, based on review of the ignition sources, as described above, include:

- Cooling unit fire
- Bus fire
- MCC/cabinet/panel fire
- Miscellaneous electrical fire
- Transient ignited fire
- Cable/transient fire due to hot work

Each of credible fire scenarios is discussed below.

Cooling Unit Fire

Two cooling unit systems are located in this area, east cooling unit T4100B004 and west cooling unit T4100B005. However, each is located far from safety related cable trays and equipment. The primary ignition source for each cooling unit is a motor windings fire. The motor of the east cooling unit is located far from any equipment or cable trays. The motor of the west cooling unit is located approximately 5 feet (line of sight distance) from the west end of bus 65F. As the motor fire would be a windings fire inside the housing, the fan motor is best modeled using the FIVE methodology radiant exposure case. An overly conservative peak fire intensity (approximately 400 Btu/sec.) would be required to create a critical flux distance of five feet. Therefore, deterministic fire modeling does not indicate that damage will occur to any safety equipment or cables due to a cooling unit fire. BOP cables were not investigated in detail for either motor; however, the motors are small fire sources and are far from any equipment and cables.

Therefore, the cooling unit fire scenarios can be conservatively modeled as either cooling unit leading to a plant trip with availability of all modeled equipment.

480V 72F Bus/Transformer Fire

The discussion provided earlier for 480V Bus 72C (see Section 4B.1.2.3.4) applies similarly to 480V Bus 72F. The credible fires are judged to initiate in the 4kV/480V transformer or a breaker cubicle. However, a major difference in the configuration between the switchgear buses in fire compartment 04ABN and those in compartment 12AB is the existence of overhead cable trays in compartment 04ABN. Fire compartment 12AB does not contain overhead cable trays. Therefore, the assumption used for 04ABN that 10% of the postulated switchgear fires are large fires propagating to cable trays and failing all functions contained in the room is much more conservative for fire compartment 12AB than it is for 04ABN. Therefore, it is assumed here that five percent (5%) of the postulated Bus 72F result in large fires that disable all functions in the room. The remaining ninety-five percent (95%) of the spectrum of 72F fires are modeled as breaker and transformer fires that result in loss of the bus.

480V 72E Bus/Transformer Fire

The discussion provided above for 480V Bus 72F applies equally to 480V Bus 72E. The credible fires are judged to initiate in the 4kV/480V transformer or a breaker cubicle. Ninety-five percent (95%) of the spectrum of 72E fires are modeled as breaker and transformer fires that result in loss of the bus. The remaining five percent (5%) of the postulated 72E fires are modeled as large fires that disable all functions in the room.

4160V 65F Bus Fire

The discussion provided above for 480V Bus 72F applies equally to 4160V Bus 65F. A minor exception to the Bus 72F configuration is that this switchgear bus unit does not contain a large transformer component. The postulated fires are assumed to occur in the breaker cubicles. Each cubicle is analyzed as an electrical panel. Ninety-five percent (95%) of the postulated 65F fires are modeled as resulting in damage internal to the switchgear bus unit. If a fire occurs in a cubicle associated with the offsite power feed to the 65F bus then, in addition to loss of the 65F bus, the offsite feed will be lost to both the 65F and the 65E buses. However, the 65E bus can still be powered by the emergency diesel generators (the 65F bus is the source of the fire and is assumed completely disabled). If a fire occurs in a cubicle not associated with the offsite power feed, then only power on the 65F bus is lost (i.e., the offsite feed to Bus 65E is still available). The remaining five percent (5%) of the postulated 65F fires are modeled as large fires that disable all functions in the room.

4160V 65E Bus Fire

The discussion provided above for 4160V Bus 65F applies equally to 4160V Bus 65E. Ninety-five percent (95%) of the postulated 65E fires are modeled as resulting in damage internal to the switchgear bus unit. If a fire occurs in a cubicle associated with the offsite power feed to the 65E bus then, in addition to loss of the 65E bus, the offsite feed will be lost to both the 65F and the 65E bus. However, the 65F bus can still be powered by the emergency diesel generators (the 65E bus is the source of the fire and is assumed completely disabled). If a fire occurs in a cubicle not associated with the offsite power feed, then only power on the 65E bus is lost (i.e., the offsite feed to Bus 65F is still available). The remaining five percent (5%) of the postulated 65E fires are modeled as large fires that disable all functions in the room.

4160V 65G Bus Fire

Although similar in construction to 4kV buses 65E and 65F, Bus 65G is smaller in size. Bus 65G is located in the southeast corner of the room approximately 4 feet south of Bus 65E. The construction of the unit and the similar spacial characteristics allow treatment of this bus in a similar manner as 65E and 65F. Five percent (5%) of postulated 65G fires are conservatively assumed to result in failure of all functions in the room. The remaining ninety-five percent (95%) are assumed to be breaker cubicle fires that fail the bus. As Bus 65G powers the recirculation pumps, the effect of loss of the 65G bus is a plant trip with all modeled systems available.

65T Breaker Cabinet Fire

The 65T breaker cabinet is a steel switchgear unit located between the 65E and 65F switchgears. The offsite transformer feeds enter the top of the switchgear unit via enclosed, louvered, metal ducts. Due to the similar construction, this unit is evaluated in a similar manner as the 65E & F and 72E & F switchgear units. Ninety-five percent (95%) of the postulated 65T fires are modeled as resulting in damage internal to the unit. Like the breaker cubicle fires of the 65 and 72 buses, using the electrical cabinet fire heat rate suggested by the EPRI Fire PRA Implementation Guide indicates that damage will not occur to nearby equipment. A fire in this unit can be expected to result in loss of the 65 transformer feed to both 4kV buses 65E and 65F. The remaining five percent (5%) of the postulated 65T fires are modeled as large fires that disable all functions in the room.

MPU 2 Fire

Modular Power Unit MPU 2 is located in the northeast corner next to MCC 72F-2A a few feet south of the west cooling unit system. There are no cable trays in the room and no conduit are nearer than approximately 2 feet to the side and 4 feet above. Due to the cubicle construction of the unit the hot gases from the internal fire will not escape directly upward such that the conduit are in the direct plume of the fire. Fire modeling using the FIVE radiant exposure case indicates that a fire in the unit will not damage the conduit overhead or the MCC next to it. As such, a postulated fire in MPU 2 will result in damage only to the functions contained within MPU 2. Based on a load review (DECO DC-5025), failure of MPU 2 will not fail any of the modeled functions in the PSA.

Therefore, a postulated fire in MPU 2 can be best modeled as leading to a plant trip with an assumed loss of the main condenser (review of RPS cabling and their effects on the MSIVs was not performed for this scenario).

MPU 2 Voltage Regulator Fire

The two (2) MPU 2 voltage regulators are located against a concrete wall approximately 4 feet in front of MPU 2. The voltage regulators are steel box units approximately 3 feet high. The top surface is a steel mesh; however, a steel shield is installed over each unit. The fire modeling results discussed above for MPU 2 apply equally to the voltage regulators.

Therefore, postulated fires of the MPU 2 voltage regulators are best modeled collectively as leading to a plant trip with an assumed loss of the main condenser (review of RPS cabling and their effects on the MSIVs was not performed for this scenario).

MCC 72F-2A Fire

A postulated fire in MCC 72F-2A would result in loss of power to the loads supplied by MCC-72B-2A. Using the FIVE radiant exposure case and the heat rate for an electrical panel recommended by the EPRI Fire PRA Implementation Guide, no damage to nearby equipment is indicated. There are no overhead cable trays. Loss of MCC 72F-2A would result in loss of 130V battery charger 2B-2 and swing charger 2B1-2. These effects can be conservatively modeled as loss of Division 2 DC power.

Cabinet 72E-4A Fire

A postulated fire in cabinet 72E-4A would result in loss of power to the loads supplied by this cabinet. Using the FIVE radiant exposure case and the heat rate for an electrical panel recommended by the EPRI Fire PRA Implementation Guide, no damage to nearby equipment is indicated. There are no overhead cable trays. No significant effects to the modeled equipment result from loss of cabinet 72E-4A. Therefore, the effects of a fire in cabinet 72E-4A can best be modeled as a plant trip with availability of all modeled systems.

Fire in 120V Distribution Cabinet for MCC 2PB-1

This cabinet is an enclosed steel cabinet. Using the FIVE radiant exposure case and the heat rate for an electrical panel recommended by the EPRI Fire PRA Implementation Guide, no damage to nearby equipment is indicated. There are no overhead cable trays. The effects of this fire can be conservatively modeled as the same effects as loss of MCC 2PB-1 (i.e., loss of HPCI).

2PB2-15 Fire

Cabinet 2PB2-15 is an enclosed steel cabinet located on the north wall of the room. Cable entries into and out of the cabinet are via conduit. In addition, this cabinet is sufficiently distant from nearby equipment and cables such that a postulated fire in the cabinet will damage only the equipment within the cabinet.

Cabinet 2PB2-15 supplies 130V DC control power to buses 65E and 72E. Although AC power remains available on these buses, control power is not available to operate the breakers. Therefore, the effects of this fire can be conservatively modeled as loss of Bus 65E.

MCC 72C-F Isolating Contactor Fire

MCC 72C-F Isolating Contactor (Div. II) is an enclosed steel panel that is located at the west end of bus 72F. It is separated from bus 72F by approximately 1 1/2 feet. A fire in this panel is expected to remain within the boundaries of the panel and not damage nearby equipment or cables.

This panel contains contacts and circuitry connecting bus 72F to the swing bus 72C-F. Swing bus 72C-F powers all RHR LPCI mode injection valves. Swing bus 72C-F is assumed to be normally fed by bus 72C. Failure of MCC 72C-F Isolating Contactor (Div. II) due to a fire only results in loss of the backup feed to MCC 72C-F. Electrical schematics were reviewed to determine whether the failure of MCC 72C-F Isolating Contactor (Div. II) would effect the operability of the 72C feed to MCC 72C-F. It was determined that the other power feed (i.e., 72F) to the swing bus is an independent circuit routed in a separate area of the plant and, thus, a fire in the 72F feed would not impact the availability of the 72C feed. Therefore, this postulated fire can be modeled as a plant trip with all modeled systems available.

Miscellaneous Electrical Items Fire

Fire compartment 12AB contains a number of small miscellaneous electrical items (i.e., those items not discussed individually above). This miscellaneous items category is comprised of disconnect switches and miscellaneous cabinets. These miscellaneous items were categorized together as the associated fires are judged to be small and would result in, at most, a plant trip. Therefore, this group of miscellaneous electrical items is best modeled collectively as resulting in a plant trip.

Transient Fires

A walkdown of the area did not identify any transient ignition sources or combustible materials. However, to bound the transient combustible fire scenario, a trash container fire is conservatively postulated in the room. This fire was modeled with the QUICK FIVE software as having a duration of 5 minutes and a heat rate of 380 Btu/sec; this is consistent with fire modeling examples in the FIVE methodology. The height of the point source trash container was assumed to be 4 feet above the floor, consistent with the FIVE methodology.

The Target-In-Plume and Radiant Exposure worksheets indicate that the critical radiant flux distance is approximately 3.4 feet. The critical height for the Target-In-Plume case for a trash container extends to the ceiling. However, the only targets in the ceiling are non-divisional conduits whose failure would at most result in a plant trip and loss of the main condenser (no cables critical to the hard pipe vent are located in the room). Therefore, the analysis of this area needs to address the likelihood of a transient combustible source placed under near important equipment (i.e., switchgear), and the possibility of a resulting fire. This bounds the loss of main condenser case.

Hot Work Fires

Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets or transient combustibles to cause ignition. No deterministic fire modeling is performed to define the conditions necessary for accidental hot work induced cable ignition. Rather, the critical distance is based on the trash container fire modeling results discussed above.

Accident Sequence Quantification

Cooling Unit Fire

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to fixed ignition sources:

$F_{fx} = F_{if} \times P_{fs} \times CCDP$, where:

F_{fx}	=	core damage frequency (due to fixed sources)
F_{if}	=	fire ignition source frequency (due to fixed sources)
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

This area is not equipped with an automatic fire suppression system. No credit is taken here for fire brigades.

Fire modeling for the cooling units indicates that fire induced failure of these cooling units is best modeled as leading to a plant trip with availability of all modeled equipment. The conditional core damage probability for this event is conservatively based on an existing run (BASE, CCDP=1.3E-5) that includes failure of the main condenser.

Therefore, the core damage frequency for the cooling units in this area is conservatively calculated as follows:

F = 1.7E-4 x 1.0 x 1.3E-5 = 2.21E-9/yr

480V 72F Bus/Transformer Fire

The fire modeling discussion above indicates that the core damage frequency associated with a postulated fire of the 72F switchgear unit is best modeled as three separate scenarios:

- Scenario 1: Bounding worst case fire. This fire is conservatively assumed to result in loss of all functions in the room.
- Scenario 2: 4kV/480V transformer fire. This fire has been shown to result in loss of power on the entire bus.
- Scenario 3: Circuit breaker cubicle fire. This fire can be conservatively modeled as resulting in loss of power on the entire bus.

The bounding worst case fire is conservatively assumed to comprise 5% of all postulated 72F fires. The remaining 95% of the fires are assumed to be comprised of transformer and circuit breaker cubicle fires. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated Bus 72F fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.7E-2 (run 12AB2 - see Table 4-4). The conditional core damage probability used for Scenarios 2 and 3 is conservatively based on run 12AB11 (CCDP=1.5E-4) which assumes failure of both 480V Bus 72F and associated 4kV Bus 65F.

Scenario 1: F = 0.05 x 3.1E-4 x 1.0 x 3.7E-2 = 5.74E-7/yr

Scenarios 2 & 3: F = 0.95 x 3.1E-4 x 1.0 x 1.5E-4 = 4.41E-8/yr

Therefore, the total core damage frequency associated with postulated fires of 480V Bus 72F is 6.18E-7/yr.

480V 72E Bus/Transformer Fire

The core damage frequency associated with postulated fires of 480V Bus 72E is calculated in the same manner as discussed above for Bus 72F. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated Bus 72E fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.7E-2 (run 12AB2 - see Table 4-4). The conditional core damage probability used for Scenarios 2 and 3 is conservatively based on run 12AB9 (CCDP=1.4E-5 - see Table 4-4) which assumes failure of both 480V Bus 72E and associated 4kV Bus 65E.

Scenario 1: F = 0.05 x 3.2E-4 x 1.0 x 3.7E-2 = 5.92E-7/yr

Scenarios 2 & 3: F = 0.95 x 3.2E-4 x 1.0 x 1.4E-5 = 4.26E-9/yr

Therefore, the total core damage frequency associated with postulated fires of 480V Bus 72E is 5.96E-7/yr.

4160V 65F Bus Fire

The fire modeling discussion above indicates that the core damage frequency associated with a postulated fire of the 65F switchgear unit is best modeled as three separate scenarios:

- Scenario 1: Bounding worst case fire. This fire is conservatively assumed to result in loss of all functions in the room.
- Scenario 2: Circuit breaker cubicle fire resulting in loss of power feed from 65 transformer and loss of the 65F bus. One breaker compartment out of a total of seven can potentially result in this scenario.
- Scenario 3: Circuit breaker cubicle fire resulting in loss of power on the 65F bus. Six breaker compartments out of a total of seven can potentially result in this scenario.

The bounding worst case fires are conservatively assumed to comprise 5% of all postulated 65F fires. The remaining 95% of the fires are assumed to be comprised of breaker compartment fires. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated Bus 65F fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.7E-2 (run 12AB2 - see Table 4-4). The conditional core damage probability used for Scenario 2 is 1.2E-3 (run 12AB10 - see Table 4-4). The conditional core damage probability for Scenario 3 is 1.5E-4 (run 12AB11 - see Table 4-4).

Scenario 1: F = 0.05 x 1.3E-4 x 1.0 x 3.7E-2 = 2.40E-7/yr

Scenario 2: F = 0.95 x (1/7) x 1.3E-4 x 1.0 x 1.2E-3 = 2.12E-8/yr

Scenario 3: F = 0.95 x (6/7) x 1.3E-4 x 1.0 x 1.5E-4 = 1.59E-8/yr

Therefore, the total core damage frequency associated with postulated fires of 4kV Bus 65F is 2.77E-7/yr.

4160V 65E Bus Fire

The fire modeling discussion above indicates that the core damage frequency associated with a postulated fire of the 65E switchgear unit is best modeled as three separate scenarios:

- Scenario 1: Bounding worst case fire. This fire is conservatively assumed to result in loss of all functions in the room.
- Scenario 2: Circuit breaker cubicle fire resulting in loss of power feed from 65 transformer and loss of the 65E bus. One breaker compartment out of a total of nine can potentially result in this scenario.
- Scenario 3: Circuit breaker cubicle fire resulting in loss of power on the 65E bus. Eight breaker compartments out of a total of nine can potentially result in this scenario.

The bounding worst case fire is conservatively assumed to comprise 5% of all postulated 65E fires. The remaining 95% of the fires are assumed to be comprised of breaker compartment fires. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated Bus 65E fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.7E-2 (run 12AB2 - see Table 4-4). The conditional core damage probability used for Scenario 2 is 6.2E-4 (run 12AB8 - see Table 4-4). The conditional core damage probability for Scenario 3 is 1.4E-5 (run 12AB9 - see Table 4-4).

Scenario 1: F = 0.05 x 1.7E-4 x 1.0 x 3.7E-2 = 3.14E-7/yr

Scenario 2: F = 0.95 x (1/9) x 1.7E-4 x 1.0 x 6.2E-4 = 1.11E-8/yr

Scenario 3: F = 0.95 x (8/9) x 1.7E-4 x 1.0 x 1.4E-5 = 2.01E-9/yr

Therefore, the total core damage frequency associated with postulated fires of 4kV Bus 65E is 3.27E-7/yr.

4160V 65G Bus Fire

The fire modeling discussion above indicates that the core damage frequency associated with a postulated fire of the 65G switchgear unit is best modeled as two separate scenarios:

- Scenario 1: Bounding worst case fire. This fire is conservatively assumed to result in loss of all functions in the room.
- Scenario 2: Circuit breaker cubicle fire resulting in loss of power on the bus, plant trip, and availability of all modeled systems.

The bounding worst case fire is conservatively assumed to comprise 5% of all postulated 65G fires. The remaining 95% of the fires are assumed to be comprised of breaker compartment fires. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated Bus 65G fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.7E-2 (run 12AB2 - see Table 4-4). The conditional core damage probability used for Scenario 2 is conservatively based on an existing run (BASE, CCDP=1.3E-5) which includes loss of the main condenser.

Scenario 1: F = 0.05 x 5.7E-5 x 1.0 x 3.7E-2 = 1.05E-7/yr

Scenario 2: F = 0.95 x 5.7E-5 x 1.0 x 1.3E-5 = 7.04E-10/yr

Therefore, the total core damage frequency associated with postulated fires of 4kV Bus 65G is 1.06E-7/yr.

65T Breaker Cabinet Fire

The fire modeling discussion above indicates that the core damage frequency associated with a postulated fire of the 65T breaker cabinet is best modeled as two separate scenarios:

- Scenario 1: Bounding worst case fire. This fire is conservatively assumed to result in loss of all functions in the room.
- Scenario 2: Circuit breaker cubicle fire. As discussed above, this fire can be conservatively modeled as resulting in loss of the 65 transformer feed to both 4kV buses 65E and 65F.

The bounding worst case fires are conservatively assumed to comprise 5% of all postulated 65T fires. The remaining 95% of the fires are assumed to be comprised of circuit breaker cubicle fires. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated 65T breaker cabinet fire scenarios is calculated below. The conditional core damage probability used for Scenario 1 is 3.7E-2 (run 12AB2 - see Table 4-4). The conditional core damage probability used for Scenario 2 is 2.1E-4 (run OFF2 - see Table 4-4).

Scenario 1: F = 0.05 x 1.9E-5 x 1.0 x 3.7E-2 = 3.52E-8/yr

Scenario 2: F = 0.95 x 1.9E-5 x 1.0 x 2.1E-4 = 3.79E-9/yr

Therefore, the total core damage frequency associated with postulated fires of the 65T breaker cabinet is 3.90E-8/yr.

MPU 2 Fire

Fire modeling for MPU 2 indicates that fire induced failure of MPU 2 is best modeled as leading to a plant trip with loss of the main condenser. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated MPU 2 fire scenario is calculated below. The conditional core damage probability for this event is based on an existing run (BASE, CCDP=1.3E-5 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of MPU 2 is calculated as follows:

F = 4.0E-4 x 1.0 x 1.3E-5 = 5.20E-9/yr

MPU 2 Voltage Regulator Fire

Fire modeling for the MPU 2 voltage regulators indicates that fire induced failure of these units are best modeled collectively as leading to a plant trip with loss of the main condenser. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of the postulated MPU 2 voltage regulator fire scenarios is calculated below. The conditional core damage probability for this event is based on an existing run (BASE, CCDP=1.3E-5 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of the MPU 2 voltage regulators is calculated as follows:

F = 1.9E-4 x 1.0 x 1.3E-5 = 2.47E-9/yr

MCC 72F-2A Fire

Fire modeling for MCC 72F-2A indicates that fire induced failure of this unit can be conservatively modeled as loss of Division 2 DC power. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of this postulated MCC fire scenario is calculated below. The conditional core damage probability for this event is based on an existing run (10AB2, CCDP=3.3E-4 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of MCC 72F-2A is calculated as follows:

F = 2.7E-4 x 1.0 x 3.3E-4 = 8.91E-8/yr

Cabinet 72E-4A Fire

Fire modeling for cabinet 72E-4A indicates that fire induced failure of this cabinet is best modeled as a plant trip with availability of all modeled systems. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of this postulated cabinet fire scenario is calculated below. The conditional core damage probability for this event is conservatively based on an existing run (BASE, CCDP=1.3E-5) which includes loss of the main condenser.

Therefore, the core damage frequency for a postulated fire of cabinet 72E-4A is conservatively calculated as follows:

F = 2.5E-4 x 1.0 x 1.3E-5 = 3.25E-9/yr

Fire in 120V Distribution Cabinet for MCC 2PB-1

Fire modeling for this cabinet indicates that the fire induced effects are best modeled as a plant trip with loss of the HPCI system. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of this postulated cabinet fire scenario is calculated below. The conditional core damage probability for this event is based on an existing run (3AB25, CCDP=2.9E-5 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of this cabinet is conservatively calculated as follows:

F = 1.2E-4 x 1.0 x 2.9E-5 = 3.48E-9/yr

2PB2-15 Fire

Fire modeling for cabinet 2PB2-15 indicates that fire induced failure of this cabinet can be conservatively modeled as a plant trip with loss of Bus 65E. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of this postulated cabinet fire scenario is calculated below. The conditional core damage probability for this event is conservatively based on an existing run (12AB9, CCDP=1.4E-5 - see Table 4-4).

Therefore, the core damage frequency for a postulated fire of cabinet 2PB2-15 is conservatively calculated as follows:

F = 2.1E-4 x 1.0 x 1.4E-5 = 2.94E-9/yr

MCC 72C-F Isolating Contactor Fire

Fire modeling for this panel indicates that fire induced failure of the panel is best modeled as a plant trip with availability of all modeled systems. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of this postulated cabinet fire scenario is calculated below. The conditional core damage probability for this event is conservatively based on an existing run (BASE, CCDP=1.3E-5) which includes loss of the main condenser.

Therefore, the core damage frequency for a postulated fire of this cabinet is conservatively calculated as follows:

F = 1.9E-5 x 1.0 x 1.3E-5 = 2.47E-10/yr

Miscellaneous Electrical Items Fire

Fire modeling of the remaining miscellaneous electrical items indicates that fire induced failure of this group of components can be best modeled collectively as a leading to a plant trip with availability of all modeled systems. Therefore, using the FIVE equation for fixed ignition sources, the core damage frequency of these postulated fire scenarios is calculated below. The conditional core damage probability for this event is conservatively based on an existing run (BASE, CCDP=1.3E-5) which assumes loss of the main condenser.

Therefore, the core damage frequency for these postulated fires is conservatively calculated as follows:

F = 1.6E-4 x 1.0 x 1.3E-5 = 2.08E-9/yr

Transient Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to transient combustible sources (note that fixed combustible fire sources are not included in this formula, for the reasons discussed above):

$F_t = F_{it} \times u \times p \times w \times P_{fs} \times CCDP$, where:

F_t	=	core damage frequency (due to transients)
F_{it}	=	fire ignition source frequency (due to transients)
u	=	transient combustible located in the range of target components
p	=	probability of combustible being exposed
w	=	frequency of finding a critical combustible loading versus the frequency of inspecting and removing the transient combustible fire source.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters P_{fs} and p are conservatively set to 1.0.

Per the FIVE methodology, u is calculated as follows:

$u = (A_s + A_{sr})$ / Net Area, where:

A_s	=	Exposed surface area of targets facing floor (e.g., width x length of horizontal cable trays)
A_{sr}	=	Area around radiant target determined by the critical separation distance (determined to be 3.4 ft., as described above)

A review of cable tray drawings shows that an approximate value of u is 0.25. This estimate employed conservative estimates for A_s and A_{sr} to simplify the analysis.

Per the FIVE methodology, w is calculated as follows:

w = (x/2) x ln(1/x), where x = F_{ccl}/F_w

F_{ccl}	=	The frequency of having a critical combustible loading present. FIVE provides a screening value of 1 event/year/compartment.
F_w	=	Frequency of combustible material inspections that would find the transient combustible fire source before a fire occurred.

The value of 1 event/year is used here for F_{ccl}.

Administrative procedure NPP-FP1-01 specifies that: 1) monthly fire protection inspections are to be performed, and 2) an hourly compensatory fire watch is to tour all areas appearing on the Out of Specification Log Sheet. This same procedure states that the accumulation of trash, oil rags, combustible materials, and similar fire hazards is prohibited. It is judged that either of the above tours would identify and correct a trash container inappropriately placed near a cable tray riser. As the hourly compensatory fire watch may not tour this area hourly or every day, a conservative estimate of one tour per week is assumed (F_w = 52/yr). Therefore, w for this area is estimated as follows:

w = ((1/52)/2) x ln(1/(1/52)) = 3.79E-2

A bounding conditional core damage probability of the above fire can be conservatively estimated using the initial screening quantification run (12AB2, CCDP=3.7E-2 - see Table 4-4).

Therefore, the core damage frequency for this area due to transient combustible fires is conservatively estimated as follows:

F_t = 2.1E-4 x 0.25 x 1.0 x 3.79E-2 x 1.0 x 3.7E-2 = 7.36E-8/yr

Hot Work Fires

Hot work induced core damage frequency is estimated as follows:

F_{hw} = F_{ihw} x u x HEP_{hw} x P_{fs} x CCDP, where:

F_{hw}	=	core damage frequency (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.
u	=	probability of hot work performed in close enough proximity to important cable targets or transient combustibles such that the possibility exists for ignition
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{fs}	=	probability of fire suppression failure

CCDP = Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameter P_{fs} is conservatively set to 1.0.

The parameter u is estimated with a value of 0.25. This value is the same as that used in the transient combustible case and is based on a 5 min. trash container fire with a heat rate of 380 Btu/sec.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

A bounding conditional core damage probability of the above fire can be conservatively estimated using the initial screening quantification run (12AB2, CCDP=3.7E-2 - see Table 4-4).

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

F_{hw} = 8.4E-4 x 0.25 x 5.0E-2 x 1.0 x 3.7E-2 = 3.88E-7/yr

Summary

The total core damage frequency due to fire in this area is calculated by summing the estimated core damage frequencies of the above fire scenarios. The core damage frequency for this area is estimated at 2.54E-6/yr. This estimate is above the 1E-6/yr FIVE screening criterion. The FIVE methodology used in this analysis does not support a core damage frequency estimate due to fire in this area less than the 1E-6/yr FIVE screening criterion.

4B.2.3.15 Ventilation Equipment Area - 13AB

The initial screening quantification of the ventilation equipment area resulted in a core damage frequency higher than the 1E-6/yr FIVE screening criterion. Further analysis was performed to address the conservatism in the initial screening analysis.

Ignition Frequency

A review of the base ignition frequency calculation for this area shows the following contributors to the fire ignition frequency:

- Four (4) Div. I ECCS trip unit cabinets: 1.7E-4/yr
- Four (4) Div. II ECCS trip unit cabinets: 1.7E-4/yr
- MCC 72C-3B: 4.8E-4/yr
- MCC 72E-5B: 6.1E-4/yr
- MCC 72F-5B: 4.8E-4/yr
- Six (6) miscellaneous panels: 2.6E-4/yr
- Seven (7) cooling unit components: 2.0E-4/yr
- Two (2) fire protection panels: 2.4E-5/yr
- Three (3) emergency lights: 1.3E-4/yr

- Cable fires due to hot work: 1.2E-4/yr
- Transient combustible fires due to hot work: 7.2E-4/yr
- Transient ignition sources: 2.1E-4/yr

As discussed earlier, the credible fires in either the fire protection panels or the emergency lights would be small and confined to within the boundaries of the unit. In addition, these postulated fires would quickly extinguish due to limited combustibles and would realistically not result in a plant trip. Therefore, the fire protection panels and the emergency lights are eliminated from further analysis.

The remaining ignition sources are maintained for further analysis.

Fire Damage Scenarios

The potential fire damage scenarios, based on review of the ignition sources, as described above, include:

- Div. I ECCS trip unit fire
- Div. II ECCS trip unit fire
- MCC fire
- Miscellaneous panel fire
- Cooling unit fire
- Transient ignited fire
- Cable/transient fire due to hot work

Div. I ECCS Trip Unit Fire

A bank of four Division I ECCS panels (H21-P080, -P082, -P084, -P086) are located in the northwest corner of compartment 13AB. Each is a separate steel cubical. Postulated failure of any of the four panels due to fire will result in failure of a number of ECCS systems. However, there is no combustible material near these units (all nearby cable is enclosed in conduit) and there are no cable trays over head. In addition, each unit in the bank is enclosed and expected to confine the fire within the boundaries of the cabinet. The steel boundaries between the cabinets will prevent a fire damaging the adjacent cabinet. The locations of the trip units are such that a fire in each will only damage equipment and functions within the cabinet.

Div. II ECCS Trip Unit Fire

A bank of four Division II ECCS panels (H21-P081, -P083, -P085, -P087) are located in the southwest corner of compartment 13AB. The discussion above for the Division I ECCS trip units applies here also; the configuration is the same and no combustibles are near the cabinets.

MCC Fire

Three (3) MCCs are located in compartment 13AB (MCC 72C-3B, MCC 72E-5B, and MCC 72F-5B). Each is located in an open area such that radiant heat from the MCC would not result in damage to other components modeled in the fire PSA. Neither MCC is located close enough to the ECCS trip units to cause fire induced damage. Therefore, a fire in either of the three MCCs will result only in failing the functions supported by the MCC.

Miscellaneous Panel Fires

Fires in the miscellaneous panels are expected to remain confined to the panel.

Cooling Unit Fire

Two compressors and four ventilation fans are located in this area. However, each is located far away from the ECCS trip units. In addition, there are no important cable trays in the vicinity of these items. A search of cables that would disable the main condenser was not performed. However, it was determined that there are no cables in the room whose failure would directly disable the hard pipe vent. Therefore, a fire of any one of these components can be conservatively modeled by assuming a plant trip occurs with loss of the main condenser.

Transient Ignited Fires

A walkdown of the area did not identify any transient ignition sources or combustible materials. However, to bound the transient combustible fire scenario, a trash container fire is conservatively postulated in the room. This fire was modeled with the QUICK FIVE software as having a duration of 5 minutes and a peak fire intensity of 380 Btu/sec; this is consistent with fire modeling examples in the FIVE methodology. The height of the point source trash container fire was assumed to be 4 feet, consistent with the FIVE methodology.

The critical height for the Target-In-Plume case is approximately 17 feet above the floor if the container is located in a corner (14 feet if against a wall and 12 feet if in center of room), and the critical radiant flux distance is approximately 3.4 feet. This area does not contain important cable trays; however, the analysis needs to address the likelihood of a transient combustible placed near the ECCS trip cabinets and the possibility of a resulting fire.

Hot Work Fires

Hot work is assumed to result in cable fires and cable damage if procedural controls are not followed and the work is being performed in close enough proximity to important cable targets or transient combustibles to cause ignition.

Accident Sequence Quantification

Division I ECCS Trip Unit Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to fixed ignition sources:

$F_{fx} = F_{if} \times P_{fs} \times CCDP$, where:

F_{fx}	=	core damage frequency (due to fixed sources)
F_{if}	=	fire ignition source frequency (due to fixed sources)
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

This area is not equipped with an automatic fire suppression system. No credit is taken here for fire brigades.

The fire damage discussion above states that a fire in any one of these panels will not damage other equipment (i.e., it will result only in failures of functions contained within the panel).

Review of the cable database information compiled in support of this analysis shows that individual failure of each of the four trip panels results in a similar list of damaged equipment: Div. I RHR, DIV. I LPCS, HPCI, RCIC. Therefore, the Div. I ECCS trip unit fire scenario is quantified using an ignition frequency equal to the sum of the individual panel frequencies and a conditional core damage probability equal to 2.2E-3 (run 13AB3, - see Table 4-4).

Therefore, the core damage frequency for this fire scenario is conservatively estimated as follows:

1.7E-4 x 1.0 x 2.2E-3 = 3.74E-7/yr

Division II ECCS Trip Unit Fires

The fire damage scenario for the Division II ECCS trip units are calculated in the same manner as the Division I ECCS trip units. The conditional core damage probability is 1.2E-3 (run 13AB4 - see Table 4-4).

Therefore, the core damage frequency for this fire scenario is conservatively estimated as follows:

1.7E-4 x 1.0 x 1.2E-3 = 2.04E-7/yr

MCC Fires

The fire modeling for the three MCCs does not indicate fire damage beyond the MCC involved in the fire. Therefore, the three MCC fires are analyzed collectively by using the sum of the individual fire ignition frequencies and the conditional core damage probability associated with the MCC with the biggest impact on plant safety.

Based on a review of the MCC loads, MCC 72E-5B contains functions resulting in the largest impact on plant safety (i.e., among these three MCCs). Failure of MCC 72E-5B can be expected to result in loss of the main condenser. The other two MCCs supply power to primarily hydrogen recombiner equipment. Therefore, the conditional core damage probability for the MCC fire scenario is based on run 15AB3 (CCDP=3.3E-5 - see Table 4-4).

Therefore, the core damage frequency for this area due to an MCC fire is conservatively estimated as follows:

1.57E-3 x 1.0 x 3.3E-5 = 5.18E-8/yr

Miscellaneous Panel Fires

As discussed above, fires in these panels are expected to remain confined within the panels. Based on review of the panel functions, failure of each panel would result in failure of no modeled equipment. Therefore, the panels are quantified collectively with a conditional core damage probability based on just a plant trip (15AB2, CCDP=3.4E-5 - see Table 4-4).

Therefore, the core damage frequency for this area due to a miscellaneous panel fire is conservatively estimated as follows:

2.6E-4 x 1.0 x 3.4E-5 = 8.84E-9/yr

Cooling Unit Fires

The fire modeling performed for the cooling unit equipment does not indicate damage to any nearby non-BOP cabling or equipment. The effect on BOP cabling that may support the main condenser was not evaluated. However, it was determined that a fire in the room would not directly disable the hard pipe vent. Therefore, the core damage frequency estimate for this ignition contributor is conservatively modeled as leading to a plant trip with loss of the main condenser. The conditional core damage probability for the MCC fire scenario is based on run 15AB3 (CCDP=3.3E-5 - see Table 4-4).

Therefore, the core damage frequency for this area due to a cooling unit fan is conservatively estimated as follows:

2.0E-4 x 1.0 x 3.3E-5 = 6.60E-9/yr

Transient Fires

The FIVE methodology recommends the following general formula for estimating the fire induced core damage frequency due to transient combustible sources:

$F_t = F_{it} \times u \times p \times w \times P_{fs} \times CCDP$, where:

F_t	=	core damage frequency (due to transients)
F_{it}	=	fire ignition source frequency (due to transients)
u	=	transient combustible located in the range of target components
p	=	probability of combustible being exposed
w	=	frequency of finding a critical combustible loading versus the frequency of inspecting and removing the transient combustible fire source.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters u, P_{fs} and p are conservatively set to 1.0.

Per the FIVE methodology, w is calculated as follows:

$w = (x/2) \times \ln(1/x)$, where $x = F_{ccl}/F_w$

F_{ccl}	=	The frequency of having a critical combustible loading present. FIVE provides a screening value of 1 event/year/compartment.
F_w	=	Frequency of combustible material inspections that would find the transient combustible fire source before a fire occurred.

The value of 1 event/year is used here for F_{ccl}.

Administrative procedure NPP-FP1-01 specifies that: 1) monthly fire protection inspections are to be performed, and 2) an hourly compensatory fire watch is to tour all areas appearing on the Out of Specification Log Sheet. This same procedure states that the accumulation of trash, oil rags, combustible materials, and similar fire hazards is prohibited. It is judged that either of the above tours would identify and correct a trash container inappropriately placed near a cable tray riser. As the hourly compensatory fire watch may not tour this area hourly or every day, a conservative estimate of one tour per week is assumed (F_w = 52/yr). Therefore, w for this area is estimated as follows:

$w = ((1/52)/2) \times \ln(1/(1/52)) = 3.79E\text{-}2$

A bounding conditional core damage probability, given failure of the suppression system, can be conservatively estimated using the more conservative of the quantification runs for the ECCS trip units (13AB3, CCDP=2.2E-3 - see Table 4-4).

Therefore, the core damage frequency for this area due to transient combustible fires is conservatively estimated as follows:

$F_t = 2.1E\text{-}4 \times 1.0 \times 1.0 \times 3.79E\text{-}2 \times 1.0 \times 2.2E\text{-}3 = 1.75E\text{-}8/yr$

Hot Work Fires

Hot work induced core damage frequency is estimated as follows:

$F_{hw} = F_{ihw} \times u \times HEP_{hw} \times P_{fs} \times CCDP$, where:

F_{hw}	=	core damage frequency (due to hot work)
F_{ihw}	=	fire ignition source frequency (due to hot work). Includes both cable and transient combustible fires due to hot work.
u	=	probability of hot work performed in close enough proximity to important cable targets or transient combustibles such that the possibility exists for ignition
HEP_{hw}	=	human error probability for failure to properly implement procedure NPP-FP1-01 regarding hot work. The assumption is that if the procedure is properly implemented that stray ignitions will not occur or that they will be immediately extinguished.
P_{fs}	=	probability of fire suppression failure
CCDP	=	Conditional Core Damage Probability

To simplify the analysis and to enhance the bounding nature of this analysis, parameters P_{fs} and u are conservatively set to 1.0.

A value of 5E-2 is used to estimate HEP_{hw}. A formal human reliability analysis was not performed. An applicable and typical screening human error probability for failing to follow a procedure in a non-stressful situation would be 0.1. The value of 5E-2 is judged to be more realistic.

A bounding conditional core damage probability, given failure of the suppression system, can be conservatively estimated using the more conservative of the quantification runs for the ECCS trip units (13AB3, CCDP=2.2E-3 - see Table 4-4).

Therefore, the core damage frequency for this area due to hot work is conservatively estimated as follows:

$F_{hw} = 8.4E\text{-}4 \times 1.0 \times 5.0E\text{-}2 \times 1.0 \times 2.2E\text{-}3 = 9.24E\text{-}8/yr$

Summary

Summing the individual fire scenario core damage frequencies, the total core damage frequency due to fire in this area is conservatively estimated at 7.55E-7/yr. This compartment remains above the 1E-6/yr FIVE screening criterion.

4B.2.3.16 CCHV Equipment/SGTS Area - 14AB

The initial screening quantification of compartment 14AB resulted in a core damage frequency higher than the 1E-6/yr FIVE screening criterion. This area is analyzed further to consider more realistic fire effects.

As the initial screening quantification for this compartment is conservative and results in a core damage frequency of 1.12E-6/yr (very close to the screening criterion), rather than perform detailed fire modeling, credit is taken here for fire brigade response. A conservative failure probability of 0.75 is used here to model failure of timely fire brigade response. Two scenarios result:

- successful suppression
- unsuccessful suppression

With successful suppression, a conditional core damage probability lower than the initial screening value would be appropriate. However, to simplify the analysis and keeping with the bounding nature of this analysis, the same initial screening value is used for both cases.

Therefore, the core damage frequency for this compartment is conservatively estimated as follows:

F = 5.62E-3 x 0.75 x 2E-4 = 8.43E-7/yr

This estimate is below the 1E-6/yr FIVE screening criterion. Therefore, this area is screened from further analysis.

4B.2.3.17 FIVE Phase II Summary

A detailed summary of the FIVE Phase II analysis for the Fermi 2 Auxiliary Building is provided in Table 4-13. A total of nineteen (19) Auxiliary Building fire compartments were identified for Phase II analysis. Three (3) of these 19 compartments screened in the initial screening quantifications. An additional eleven (11) compartments screened following detailed fire modeling. The following five (5) Auxiliary Building compartment remained unscreened at the end of the FIVE analysis:

- 03AB, Relay Room
- 04ABN, Major Division I Portion of Div. I SWGR Room
- 09AB, Control Room Complex
- 11ABE, Miscellaneous Rooms - Majority of Area (Div. I)
- 12AB, Division II SWGR Room

Refer to Section 4B.0 for a discussion of the dominant accident sequences associated with a postulated fire in each of these five areas.

4B.3 REVIEW OF PLANT INFORMATION AND WALKDOWNS

A variety of information sources were reviewed and a number of walkdowns were performed in support of the fire analysis for the auxiliary building. These are discussed below.

4B.3.1 Review of Plant Information

The following are the primary information sources reviewed as part of the internal fire analysis:

- Fire Protection Analysis
- Fermi 2 CECO database
- Fermi 2 cable database
- Plant arrangement drawings
- FHA drawings
- Electrical drawings
- Fermi 2 PSA REBECA documentation
- Plant procedures
- Design Calculations

The Fire Protection Analysis is a detailed information source that provided numerous inputs into the fire analysis, such as:

- Equipment in each fire zone
- Combustible loading in each fire zone (type and quantity)
- Fire protection equipment in each fire zone
- Consequences of design basis fire in each fire zone
- Consequences of fire suppression system actuation in each fire zone
- Design basis information concerning fire suppression systems
- Design information concerning fire barriers
- Design information concerning alternate shutdown capability

The accuracy of the cable routing information is of paramount importance since it is this that provides much of the basis for the fire analysis. For the Appendix R equipment, the associated raceways and their location within specific fire areas were validated as part of the Appendix R program. Summaries of this information are contained in the Fire Protection Analysis in the form of drawings and text. More detailed information, such as conduit, cable tray, and cable numbers were obtained from the CECO database and the DECO cable database.

Plant arrangement drawings were used, as necessary, in conjunction with CECO database and DECO cable database information to locate equipment credited in the fire analysis. The arrangement drawings were also reviewed throughout the internal fire assessment, as necessary, to locate equipment and to assess the potential for fire spread.

The FHA drawings were also a well-used source that aided in the determination of the potential for fire spread. The FHA drawings indicated the following information directly on the drawings:

- Fire zone boundaries and zone IDs
- Fire doors
- Fire dampers
- Raceway fire wrap
- Fire detection equipment
- Suppression systems and coverage areas

Conduit and cable tray drawings were used when it was necessary to accurately locate specific cabling (either for fire damage modeling or due to some ambiguity in another information source). one-line electrical drawings were used as a reference source in the determination of power supply vulnerabilities.

The Fermi 2 PSA REBECA documentation was used as an information source in the determination of equipment vulnerabilities. The REBECA fault trees provided a quick reference source for determining the impact on systems modeled in the fire analysis due to fire induced damage of individual components or subsystems. This was necessary to support the modeling of system train failures in the RISKMAN PSA model.

Plant procedures were reviewed as part of the assessment to: 1) verify that all fire barriers credited in the fire analysis are covered under a surveillance program, 2) verify that ignition sources and combustibles are covered by procedural requirements, and 3) determine whether a fire initiated event occurs for a fire in a given area. The procedures reviewed included the following:

- Abnormal Operating Procedures (AOPs)
- Administrative Procedures
- Fire Protection Procedures
- Fire Pre-Plan Procedures

A number of Fermi 2 design calculations were used to support the determination of equipment vulnerabilities:

- DC-5024, "Load List and Loss of Power Impact - MPU #1"
- DC-5025, "Loss of Power Impact on MPU #2"
- DC-5026, "Loss of Power Impact for MPU #3"
- DC-5027, "Loss of Power Impact for MPU #4"
- DC-5028, "Loss of Power Impact for MPU #5"
- DC-5029, "Loss of Power Impact for MPU #6"
- DC-5702, "Fire Loading Calculation"

In addition, the Fermi 2 "Cable Routing Equipment - From and To By Equipment" database was used to support the determination of equipment vulnerabilities.

4B.3.2 Walkdowns

Phase III of the FIVE methodology requires performance of walkdowns and verification activities for two purposes:

- to gather data and information to perform Phases I and II, and
- as a final check to verify assumptions and modeling approaches in the analysis.

Such walkdowns may be performed as necessary during the analysis and after the analysis to verify assumptions. The walkdowns performed were confirmatory in nature. These walkdowns were intended to provide confirmation of equipment location and area dimensions needed for fire damage analyses and to confirm information in the screening analysis files.

The first fire IPEEE walkdown of the Auxiliary Building was performed on July 25-26, 1995. The walkdown group consisted of one Fermi 2 engineer from the Fire Protection Group and two engineers from ERIN Engineering. The purpose of this walkdown was to review the Auxiliary Building fire compartment boundaries and general layout to support the Phase I analysis.

A second fire IPEEE walkdown of the Auxiliary Building was performed on August 30, 1995. The walkdown group consisted of one Fermi 2 engineer from the Fire Protection Group and two engineers from ERIN engineering. The purpose of this walkdown was to review fire source and target spacial details to support fire modeling in the Phase II analysis.

A third fire IPEEE walkdown of the Auxiliary Building was performed on December 5-7, 1995. The purpose of this walkdown was to obtain detailed spacial information and construction details of the switchgear to facilitate the fire modeling of the divisional switchgear rooms. The Control Room was also visited to verify the existence of fire bulkheads between the control board panels and to look for open topped cabinets.

Other short and informal walkdowns were performed by Fermi 2 fire protection personnel during the fire hazards evaluation.

4B.4 FIRE GROWTH AND PROPAGATION

The treatment used in this analysis for fire growth and propagation follows the FIVE methodology. As part of the fire growth and propagation analysis, calculation spreadsheets were developed for different fire types using the QUICK FIVE computer code.

Consistent with the requests of NUREG-1407, the following issues are discussed below:

- Fire size and duration
- Cross-zone fire spread
- Spread of hot gases and smoke

4B.4.1 Fire Size and Duration

The recommendations in the FIVE methodology are used to estimate the fire size and duration.

The fire size and duration depends on the type and amount of combustibles available. The FIVE methodology classifies combustibles into two types:

- Fixed combustibles
- Transient combustibles

Examples of fixed combustibles are cables, pump/motor lubricating oil, electrical cabinets, batteries and filtration media (e.g., carbon). Examples of transient combustibles are rags, anti-contamination clothing, cleaning solvents, and trash barrels. Per guidance in the FIVE methodology, transient combustibles that need not be considered include:

- Flammable and combustible liquids stored in approved containers
- Flammable and combustible liquids stored in approved storage cabinets
- Combustible liquids stored in sealed 55 gallon drums
- Clothing and other incidental combustibles kept in closed metal cabinets
- Clothing and trash kept in closed non-combustible containers

The duration and size of the fire is determined by the amount of combustibles in the area (determined from the Fermi 2 Fire Protection Analysis and walkdowns) and the heat release rate of the combustible type as defined in the FIVE methodology. Guidance provided in the FIVE methodology, in the form of examples and test data, was used to assign heat release rates and damage thresholds for equipment for which heat release rate and damage threshold data could not be located (e.g., switchgear fires).

4B.4.2 Cross Zone Fire Spread

The assessment of the potential of fire spread from one compartment to adjacent compartments is performed in the Fire Compartment Interaction Analysis of Phase I. The FIVE methodology provides boundary criteria (see Section 4B.0.1.1.6) to be used in the determination of the potential for fire spread. If the boundary criteria are met, then the analysis assumes that any postulated fire in the area will remain within the confines of the fire compartment boundaries. If the boundary criteria are not met for an adjacent compartment then the analysis assumes that fire may spread to that adjacent compartment. As such, adjacent fire compartments with unscreened fire barriers are combined into single fire compartments at the end of Phase I.

4B.4.3 Spread of Hot Gases and Smoke

The FIVE methodology conservatively assumes at the initial stage that any fire in a compartment will result in damage to all equipment within the compartment. For compartments that do not screen out with this assumption, the methodology provides a detailed process of fire damage analysis. This process evaluates targets in relation to the fire plume, hot gas layer, and thermal radiation.

The modeling of fire growth and propagation uses the algorithms and look-up tables contained in the FIVE methodology.

4B.5 EVALUATION OF COMPONENT FRAGILITIES AND FAILURE MODES

4B.5.1 Evaluation of Fire-Induced Failures

Information from the Fermi 2 CECO and cable databases and drawings (conduit, cable tray, isometric, and arrangement) was used to determine the vulnerabilities in a compartment. Database searches by compartment provided location information for the following components:

- cable trays
- conduit
- cables
- equipment

For many compartments in the Auxiliary Building, assuming that all cable and equipment in a compartment resulted in failure of the associated systems would result in a core damage frequency many orders of magnitude above the FIVE screening criterion of 1E-6/yr. As such, the above information was reviewed in detail to determine realistic functional and systemic vulnerabilities in the compartment. This investigation typically involved noting whether a cable was a control, power, or instrumentation cable, and which equipment the cable connected. Using systems interaction knowledge, a determination was then made as to whether failure of a specific cable resulted in failure of a modeled function or system. Detailed electrical circuit analysis was not performed. Although this investigation was conservative (e.g., when a circuit function was unclear, it was assumed to cause system failure), this investigation did result in deleting a number of obvious non-system failures, that would have otherwise been considered failures, from the initial screening quantification runs.

This approach was used primarily in the initial screening quantifications of the Phase II analysis. Once detailed fire modeling began, the analysis focused on damage thresholds, critical fire distances, realistic fire scenarios, and fire suppression. When equipment was determined to be damaged due to fire, the information from the equipment vulnerability investigation was used to make new quantification runs or use existing runs.

4B.5.2 Fire Damage Modeling Approach

The deterministic fire modeling performed for the compartments that did not screen in the initial screening quantifications began with a realistic review of the ignition sources in the compartment. This review eliminated ignition sources that were determined not to exist or not to apply to the compartment (e.g., heater in the cable spreading room). In addition, ignition sources such as emergency lights and fire protection panels were generally dismissed as credible fire sources that would result in a core damage accident.

Once a list of realistic ignition sources was determined, the fire damage modeling defined fire scenarios and important target sets. For every identified target set, it was necessary to determine the geometric relationship between potential targets and fire sources. Three general types of fire scenarios were considered.

- Targets located in the plume, directly above the fire source.
- Targets located in the hot gas layer (outside the plume, but possibly in the ceiling jet)

- Targets exposed to heating by thermal radiation, located next to the fire source.
- The FIVE methodology was used to evaluate fire growth and propagation. The target temperature rises determined from the FIVE algorithms and look up tables were then compared with target damage threshold criteria (temperature or heat flux) and if the criteria were not exceeded, the specified fire was screened from further analysis. (Note that this was performed using the automated QUICK FIVE worksheets.) If the damage threshold was exceeded, then the target was assumed failed and a PSA quantification run was performed to determine the conditional core damage probability given the associated component damage of this fire scenario. This analysis required collection of data for the following parameters.
- Location of targets relative to a potential fire source
- Damage threshold criteria for targets
- The exposure fire peak intensity and total energy content
- The fire enclosure volume and heat loss fraction
- FIVE fire location factor (4 for corner, 2 for against wall, and 1 for center of room)

In the case of fire suppression, credit was taken for fire suppression when an automatic system existed in the compartment. Even with successful suppression, the fire damage modeling included conservative assessments of equipment failure due to initial fire damage. The failure probabilities for the various fire suppression systems were taken from the FIVE methodology.

Credit was conservatively not taken for fire brigades in the Auxiliary Building, with one exception - compartment 14AB. This area exceeded the FIVE screening criterion by a small margin in the initial screening quantification. A conservative failure probability of 0.5 for fire brigade suppression reduced the core damage frequency for the compartment below the FIVE screening criterion. This is appropriate, as detailed fire modeling for this area would have otherwise shown the area to be not a significant fire risk area (i.e., CDF below 1E-6/yr).

4B.5.3 Damage Threshold Criteria

This analysis used basic FIVE methodology damage threshold criteria. The key criteria are repeated below.

A temperature of 700°F was used, per the FIVE methodology, as the failure temperature criterion for IEEE-383 qualified cables. The FIVE methodology suggests a temperature of 425°F for non-qualified cable. This value was not used as all cabling considered in this analysis is IEEE-383 qualified.

In the case of radiant heat flux, the FIVE methodology prescribes a representative value of 1 Btu/sec/ft^2 for qualified cables. A value of 0.5 Btu/sec/ft^2 is suggested in the FIVE methodology as a screening value for non-qualified cable. This screening value and an intermediate value of 0.75 were used for equipment other than cables (e.g., switchgear, MCCs).

4B.5.4 Fermi 2 Fire Damage Modeling Cases

Fire damage modeling was performed for all areas that did not screen in the initial screening quantifications of Phase II.

4B.6 ANALYSIS OF PLANT SYSTEMS, SEQUENCES, AND PLANT RESPONSE

The calculation of the conditional safe shutdown failure probability given a fire initiating event was performed by using the Level 1 Fermi 2 PSA Models. All available mitigation paths modeled in the Fermi 2 PSA are considered in this analysis. The equipment not directly affected by a fire can represent a significant benefit in preventing core damage, but may also have a high conditional failure probability. Therefore, the frequency of core damage due to a fire must account for this additional random failure probability of other equipment to make an appropriate assessment of the fire risk.

The PSA logic models are used to represent the equipment remaining after a fire affects a fire compartment. An existing PSA event tree structure and the associated logic models are used for analysis to obtain the conditional core damage probability. The combination of this conditional probability with the ignition frequency, yields a core damage frequency for comparison to the FIVE screening value of 1.0E-6 per year. Below this value, the fire compartment does not need to be considered further in the IPEEE fire assessment.

Details of this risk analysis are given in Section 4.3. The quantification runs were performed using the RISKMAN code. The runs performed for the Auxiliary Building are summarized in Table 4-4.

4B.6.1 Dominant Core Damage Sequences

The FIVE methodology is a screening approach that is not focused toward developing realistic core damage sequence frequencies for all postulated fire scenarios in the plant. The overwhelming majority of postulated fire scenarios are screened from further detailed analysis prior to calculating reportable core frequencies (i.e., conservative quantifications are initially employed).

However, the following five Auxiliary Building compartments remain unscreened at the end of the FIVE analysis:

- 03AB 2.77E-6/yr
- 04ABN 4.51E-6/yr
- 09AB 4.27E-6/yr
- 11ABE 1.90E-6/yr
- 12AB 2.54E-6/yr

4.3 ANALYSIS OF PLANT SYSTEMS, SEQUENCES, AND PLANT RESPONSE

4.3.1 Calculation Of Conditional Core Damage Frequency

4.3.1.1 Baseline Fire Model

This section describes the baseline Fermi 2 PSA plant model which was used as the starting point of all conditional core damage probability (CCDP) calculations given a specific fire scenario. (The terms conditional core damage probability, CCDP, and conditional core damage frequency, CCDF, are used interchangeably in Section 4.) The baseline model was developed from the Fermi 2 IPE plant model **[4.9]**. The IPE model employed a linked event tree approach utilizing the RISKMAN code. Two types of event trees were included in the IPE model, the general transient event trees, and the special event trees, such as LOCA and Break Outside Containment Event Trees. The general transient event trees consist of the Front-line System Event Tree (FLT), the Recovery Event Tree (RCVRY), and the General Transient Late Event Tree (GTLAT). The general transient event trees are most suitable for modeling the impacts to the plant systems by a fire and hence were selected as a basis for developing the baseline plant model in the IPEEE submittal. In addition, the Fermi 2 IPE included two support system event trees, the Electric Power Event Tree (EPT), and the Mechanical System Event Tree (MST). These support system event trees were also used in the IPEEE baseline plant model. The resulting baseline plant model consists of five linked event trees. These are shown below in the order that the event tree appears in the PSA model:

- Electric Power Event Tree (EPT)
- Mechanical System Event Tree (MST)
- Front Line System Event Tree (FLT)
- Recovery Event Tree (RCVRY)
- General Transient Late Event Tree (GTLAT)

It is assumed that any fire in the power plant causes a reactor trip and hence is treated as a general transient. For the purpose of calculating CCDP, the frequency of such a fire is assumed to be 1 (per year). The impacts of a fire are modeled by setting the respective top event in the appropriate event tree to guaranteed failure. If a fire caused partial failure of a system, or it only fails one division of the systems, the split fraction rules were modified so that the appropriate split fractions were chosen to represent the impact. The impacts of fires include damaging the power buses, cables (power, control, and instrument cables), and equipment included in the system models. Hence, for any fire scenario, spatial information needs to be obtained to assess the damage to the equipment in that fire zone and the impact that the equipment damage has on the respective top event. Further, cables should be traced to assess the impact of a fire at any location, because equipment at other locations may be unavailable due to cable damage. Tracing all the cables would require a tremendous amount of effort and was deemed not to be necessary. As long as the plant model conservatively assumes that systems with non-traced cables are guaranteed failed by any fire in the plant and the conservatism induced by such assumptions does not predict unrealistically high CCDP, then a reduced set of traced cables is adequate. The baseline model assumes a generic fire in the power plant, which leads to a demand for a SCRAM and which can be in any location. Following are the impacts of the generic fire:

- The main condenser is not available, but the condensate pumps are available for makeup to the condensate storage tank (CST)
- Fire Causes all MSIVs to fail close
- Feedwater system is not tracked - assume not available
- Heater Feed pumps are not tracked - assume not available
- Credit is not taken for off-site grid recovery once a fire damages an off-site power bus
- Inter divisional cross ties of ESF buses are assumed to be a guaranteed failure
- No recovery of the main condenser is allowed due to fire
- Late boron injection is not tracked - assume not available
- For external injection, only standby feedwater is tracked

The above set of impacts were included in developing the base model used for any fire in the plant. This set generally represents the least impact of a fire in the Fermi 2 Nuclear Power plant. For a specific fire scenario, equipment evaluated to be disabled by that specific fire is added to the above generic equipment impact list. However, for some fire scenarios, it was determined that some of the equipment not tracked would not be impacted by the fire. As an example, unless directly damaging cables leading to the MSIVs or the condenser, fires in the relay room and the control room were assumed not to affect the operability of the MSIVs or the condenser. Therefore, for those events where it could be determined that the MSIVs and condenser were not affected by the fire, a challenge to these systems was placed back into the PSA evaluation of the event.

4.3.1.2 Conditional Core Damage Frequency of Individual Fire Scenarios

A CCDP was calculated for each of the postulated fire scenarios. Depending on the assumptions employed, the impacts of a fire may be categorized into one of two groups:

a) All of the equipment and cables in the zone where the fire is located are damaged, or,

b) A limited set of equipment or cables in the location is damaged.

In the latter case, the set of equipment could be one division, due to the fact that divisional separation is required by the Appendix R analysis, or it could be any set of equipment that is spatially adjacent to each other or a set of system functions that are disabled due to the damage of control cables passing through the fire location. For the purpose of calculating a CCDP each fire scenario is assumed to occur which is numerically equivalent to having an initiation frequency of once per year. The set of equipment and cables damaged by the fire scenario combined with the risk impacts as calculated in the PSA model yield the total impact of the corresponding fire scenario. The equipment and cable impacts are modeled through modifications of the split fraction rules in the linked event trees listed in section 4.3.1.1. Table 4-4 shows the designator, the main top events impact, and the CCDP of each fire scenario.

4.3.1.3 Summary of Plant Response

The plant response to the fire is modeled through guaranteed or partial failure of the top events in the PSA event tree model. The main top events impacted by a fire are listed in Table 4-4. The following is a discussion of the impacts associated with the top events in the PSA model:

OS:

Top events O1 and O2 model the unavailability of power from the off-site grid, including failure of the 120-kv mat (01) and the 345 KV mat (02) switchyards. The major pieces of equipment modeled in these top events are the breakers, transformers, and buses that connect the off-site grid to the engineering safeguards feature buses. If a fire damages any of this equipment, the corresponding top event (O1 and/or O2) is assumed guaranteed failed. Credit is not taken for recovery from loss of off-site power if loss of the grid is due to a fire.

DA and DB:

These two top events model the unavailability of essential DC power from the station batteries, with DA representing Division I and DB representing Division II. Failure of DA will prevent starting of emergency Diesel Generators (EDG) 11 and 12, and failure of DB disables EDG 13 and 14. The major equipment modeled in these two top events include the batteries, battery chargers, and the DC bus. A fire could damage the DC power equipment and cause failure of top events DA and/or DB.

EDGs:

Top events G1, G2, G3, and G4 model the unavailability of the emergency diesel generators 11, 12, 13, and 14, respectively. The major equipment modeled in these top events include the diesel generators, the diesel output breaker, and the diesel service water and fuel oil transfer systems. A fire could damage this equipment and fail the diesel generator top events.

BUSES:

Top events B4, C4, E5, and F5 model power unavailability due to faults or bus maintenance on the respective 4160 and 480V buses. The major equipment modeled in these top events includes the 4160V AC buses (e.g., 4160V bus 64B and EDG bus 11A for top event B4), 480V AC buses (e.g., bus 72B and 72EA for top event B4), and the associated breakers and transformers. A fire may damage this equipment and disable the corresponding buses.

CF:

Top event CF models the unavailability of 480V AC power at swing bus 72CF due to bus fault or maintenance. Major equipment includes swing bus 72CF, and the relays and circuitry necessary to cause transfer of swing bus 72CF.

B1, B2, and DC:

Top events B1, B2, and DC model the unavailability of power to the balance of plant (BOP) distribution. The major equipment modeled in top event DC includes the BOP battery, battery chargers, and the bus required to supply DC power. Major equipment modeled in B1 and B2 include the breakers, transformers, and the buses required to supply BOP AC power. A fire damaging this equipment disables the corresponding BOP top events.

NP

Top event NP models the failure of the analog transmitter trip system (ATTS) to generate a low reactor pressure signal. Equipment modeled in this top include the pressure transmitters, bistables and output relays associated with reactor pressure. A fire may fail the top NP by damaging any of the above equipment.

LV:

Top event LV models the failure of the ATTS to generate a reactor low water level signal from either level 1 or level 2. Equipment modeled includes the level transmitters, bistables, and relays associated with reactor water level.

TB:

Top event TB models the failure of the Turbine Building Closed Cooling Water (TBCCW) system to supply cooling water to the station air compressors, and the feed and condensate system pumps. The equipment modeled includes the TBCCW pumps and valves. A fire damaging this equipment will disable top event TB.

CC:

Top event CC models the failure of the Reactor Building Closed Cooling Water (RBCCW) system to supply cooling to the essential reactor equipment. Equipment modeled includes the RBCCW pumps and valves.

C1/C2:

Top events C1 and C2 model the division I and II of Emergency Equipment Cooling Water/Emergency Equipment Service Water (EECW/EESW). The two top events include the automatic initiation signal, pumps, pipes, and valves necessary to provide cooling water for the specified emergency equipment.

SA:

Top event SA models station air and the interruptible air system. Equipment modeled includes the station air compressors and associated valves and piping, and the interruptible air header.

A1/A2:

Top events A1 and A2 model the two divisions of the noninterruptible air system. Equipment modeled includes the piping and valves needed to maintain supply of air from the station air system (normal supply), and the compressors, valves, and piping needed to maintain a supply of air from the NIAS compressors (alternate supply). A fire damaging this equipment may disable division I and/or II of NIAS.

PT:

Top event PT models failure of recirculation pump trip. This top event is only questioned when reactor scram fails and addresses the automatic ATWS mitigation function of tripping off both recirculation pumps on high reactor vessel pressure or low-low vessel water level. Fire damage of the recirculation pumps is assumed to result in a hot short and continued operation of the pumps.

VS:

Top event VS models the opening and closing of the safety relief valves for pressure control in the safety and low-low set modes of operation.

V3:

Top event V3 questions whether more than two SRVs are stuck open, given that at least one SRV fails to reseat. This top event is only asked in sequences involving failure of top event VS, which models failure of the SRVs to reseat. Failure of event V3 is assumed to result in a rapid depressurization of the reactor vessel to below the shutoff head of low pressure injection systems and failure of steam driven turbines. The injection requirements for such a condition are similar to those for a large (and medium) LOCA. Failure of VS and success of V3 is approximately equivalent to a small LOCA

CT:

Top event CT models the unavailability of the Condensate Storage Tank (CST) as a water source for operation of the standby feedwater system, control rod drive hydraulic system, HPCI, and RCIC. This top event also models the makeup of water to the CST for the long-term operation of the standby feedwater system or control rod drive hydraulic system. A fire which disables the top event CT will result in the unavailability of the standby feedwater system and control rod drive hydraulic system.

SF:

This top event models the unavailability of Standby Feedwater System to control reactor water level. This top event is questioned in sequences in which main feedwater is unavailable. The success criteria of this top event is that the operator manually starts the standby feedwater pumps, the standby feedwater system remains available for 24 hours following reactor scram, and the operator manually throttles standby feedwater flow to control the vessel water level. Failure of this top event leads to sequences in which the high pressure injection systems (RCIC or HPCI) or manual depressurization and the low pressure injection systems are questioned.

RC:

Top event RC models the unavailability of Reactor Core Isolation Cooling (RCIC) System as a short-term high pressure water injection and cooling system. The success criteria is that the RCIC system either starts automatically, on low-low vessel level, or manually to provide flow to the reactor vessel.

HP:

Top event HP models the unavailability of the high pressure injection system (HPCI) as a short-term high pressure water injection and cooling system. Success requires that the HPCI system either starts automatically or manually on low-low vessel level or high drywell pressure or manually to provide flow to the reactor vessel.

LA and LB:

Top event LA and LB model the unavailability of RHR pump trains in loop A and B, respectively. Major equipment includes the RHR pumps and the suction valves from the suppression pool. In non-ATWS events, pump operation is required for LPCI and/or shutdown and suppression pool cooling. Failure of top event LA and/or LB disables the corresponding loop for Low Pressure Core injection (LPCI), shutdown cooling, and suppression pool cooling models of operation.

CS:

Top event CS questions whether the core spray system injects water into the reactor vessel after the reactor is depressurized. The success criteria is that the suppression pool water is available to the core spray pump suction, at least one core pump starts and runs for 24 hours, and the injection path into the reactor vessel is available.

OV:

Top event OV questions whether human actions and hardware components are successful in venting the torus airspace or drywell to remove decay heat. For non-ATWS with loss of decay heat removal sequences the operator has about 19 hours to vent before containment integrity is challenged. Venting is guaranteed failed for ATWS sequences when other ATWS recovery options have also failed such as SLCS injection and recirculation pump trip, or with three or more stuck open SRVs.

4.3.1.4 Dominant Accident Sequences from Fire

All potential fire sources were identified. The frequency of a fire from each source was estimated along with the probability that the fire would lead to a core damage event. Combining the initiating fire event frequency with the Conditional Core Damage Probability (CCDP) leads to the core damage frequency (CDF) for a given fire initiator in a given compartment. Six fire compartments exceeded the 1.0E-06/yr core damage frequency screening criterion in the FIVE methodology. The resulting CDFs for these unscreened compartments are given in Table 4-14. These compartments were the relay room, the division 1 and division 2 switchgear rooms, the control room, a room on the 3rd floor of the auxiliary building and the 2nd floor of the reactor building. The following sections contain a brief discussion of the major contributions to the CCDP for each of the six compartments.

4.3.1.4.1 Relay Room

The most severe complications in the relay room were due to fires in panels P623, P854 and P855. A fire in panel P623 or P854 was assumed to cause depressurization and to fail operation of the hardened vent plus one division of RHR for heat removal. HPCI is also tendered inoperable, but that failure is inconsequential due to the vessel depressurization. The main condenser for heat removal was also assumed to fail given a fire in either panel P623 or P854. This would leave only one division of RHR for heat removal. Given a random failure of the available division of RHR, the conditional core damage frequency for a fire in panel P623 or P854 is about once in every 27 events, dominated by loss of decay heat removal sequences.

The consequences of a fire in panel P855 are similar to the consequences of a fire in panel P623 or P854. A fire in P855 would not take out HPCI but would add failure of a division 2 ESF, division 2 EDGs, and division 2 core spray. However, these differences have very little impact on CCDF due principally to the redundant impacts of the failures that are in common.

4.3.1.4.2 Division 1 and Division 2 Switchgear Rooms

The dominant core damage contributors to a fire in the division 1 switchgear room are fire events 4AB02 and 4AB04. Both of these events are assumed to result in a severe degradation of the decay heat removal capability. Given either event, only division 2 of RHR is assumed to be available for decay heat removal. In addition, it is assumed that the events would result in a guaranteed depressurization of the RPV. Only division 2 of Core Spray and division 2 of RHR are available for injection. Random failures of the heat removal capability or the injection function would result in core damage. The conditional core damage frequency given event 4AB02 or 4AB04 is about once in every 29 events, dominated by loss of injection at low pressure sequences.

Of all the fire sources in the division 2 switchgear room, fire event 12AB02 is most apt to result in core damage. This event assumes that the reactor is depressurized due to failure of the SRVs to reseat as a direct result of the fire. HPCI and RCIC are both available but are assumed to be inoperable due to the depressurization. Operation of the standby feedwater system is also assumed to be degraded by the fire and must be remotely started at the dedicated shutdown panel. Random failure of the available low pressure injection systems is assumed to result in core damage. The conditional core damage frequency given fire event 12AB02 is about once in every 27 events, dominated by loss of injection at low pressure sequences.

4.3.1.4.3 Control Room

The most severe fire in the control room would be a fire in control panel H11P602. A fire in this panel is assumed to result in the failure of the main condenser, RBCCW and all division 2 ESF systems. The RPV is assumed to depressurize due to the failure of SRVs to reseat. Random failure of standby feedwater and the division 1 low pressure injection systems is assumed to result in core damage. The conditional core damage frequency for fire event 9AB01 is about once in every 100 events, dominated by loss of injection at low pressure sequences.

4.3.1.4.4 Third Floor Auxiliary Building

Outside of the division 2 switchgear room is a room containing division 1 and division 2 MCCs plus the three battery chargers for the 125 volt division 1 batteries. A widespread fire in this room that might be caused by hotwork is assumed to fail division 1 DC power and RCIC. In addition, it is assumed that a fire would render the hardened vent and the division 1 SRVs inoperable. Core damage could occur from random failures of the available division 2 equipment leading to either a loss of injection or inadequate decay heat removal. The conditional core damage frequency for fire event 11AB01 is about once in every 153 events, dominated by loss of decay heat removal and loss of injection at low pressure sequences. However, it should be noted that a more likely fire would be initiated in the immediate vicinity of the battery chargers themselves. However, the consequences would be limited to failure of the corresponding division of DC power. The conditional core damage frequencies for fire events 10AB1 and 10AB2 (division 1 or 2 DC, respectively) are about once in every 2500 events and once in every 3030 events, respectively. The former is dominated by loss of decay heat removal and loss of injection at low pressure sequences, and the latter is dominated by loss of injection at both high and low pressure sequences.

4.3.1.4.5 Second Floor Reactor building

MCC 72C-F swing bus is located on the second floor of the reactor building. The primary loads fed from the swing bus, with regard to risk significance, are the LPCI injection valves. Fire modeling has found that the fire with the most severe consequences, on the second floor of the reactor building, involves a localized fire in MCC 72C-F. Fire event RB6S05 evaluates the effects of a fire in the MCC 72C-F swing bus. Such a fire would impact not only LPCI injection, but the decay heat removal capability via the shutdown cooling mode. It is conservatively assumed that given fire event RB6S05 that Station Air and the hardened vent would not be available. Failure of Station Air would close the MSIVs making the condenser unavailable for heat removal. The conditional core damage frequency given fire event RB6S05 is dominated by a random failure of the torus cooling injection valves. Such a failure would result in the total loss of decay heat removal given the assumed unavailability of the condenser and the hardened vent. Given these assumptions, core damage would be expected about once in every 1449 occurrences of fire event RB6S05, dominated by loss of decay heat removal sequences.

4.4 ANALYSIS OF CONTAINMENT PERFORMANCE

The FIVE evaluation of Fermi 2 did not consider fires inside the containment because the containment is inerted with nitrogen during operation. This nitrogen will prevent any fire from propagating beyond the specific source of ignition. In addition, a review of the equipment inside the containment determined that there are no significant sources with the capability to generate anything other than a small portion of the heat required to generate a hot gas layer (HGL) inside the containment which could cause damage to cables or equipment required to achieve safe shutdown.

Another issue addressed is whether a single fire could damage redundant equipment by including both trains in a single fire plume. The review of the cable routing and equipment location databases and plant drawings demonstrated that there is sufficient spatial separation between equipment to prevent a single fire from affecting both trains of safe shutdown equipment.

The area between the steel drywell and the shield wall is filled with polyurethane foam. The drywell gap analysis, design calculation DC-5241 **[4.69]**, analyzed this area to determine the effect of a fire in the foam on the ability to safely shutdown the plant. In addition, DC-5241 showed that reactor hot shutdown can be achieved from the main control room without repair procedures, and that cold shutdown can be achieved with manual operation of several motor operated valves given the worst case scenario of a fire in the drywell gap.

A walkdown of Fermi 2 was performed to verify that the drywell access hatches and drywell head were not at significant risk of damage due to a fire. This walkdown confirmed that the hatches and drywell head are located in areas with low combustible loading and that there are no ignition sources in their vicinity that would impair the ability of these structures to maintain containment integrity or cause damage to components located inside the drywell. Moreover, all equipment and personnel hatches as well as the penetrations and penetration seals are passive; i.e., they do not relay on pneumatic pressure or electricity to function. Thus, vulnerability of these support systems to fire is not an issue with regard to the integrity of these passive containment components

Finally, the overall challenge to containment by the core damage sequences that do not screen out through the FIVE methodology is addressed. It first needs to be restated in dealing with the absolute magnitude of core damage sequences in this context that there is considerable conservatism in the FIVE methodology that leads to unrealistically high values for such sequences. Principal among these assumptions are:

- Use of fire initiation frequencies based on the FIVE fire events data base is conservative since some of the recorded fires are very small and others self-extinguish. This conservatism is only partially offset by the use of fire modeling.
- Systems for which cabling has not been tracked or definitely precluded from fire damage is assumed to be damaged by a fire
- Little credit is taken for detection and manual suppression

Notwithstanding these conservatisms, only six compartments do not screen out against the FIVE <1.0E-6/yr CDF criterion. The largest calculated CDF was 4.5E-6/yr, and two values were barely in excess of the screening limit. See Table 4-14. The PSA evaluation shows that about one third of the functional contribution to the CDF for the unscreened compartments is due to loss of decay heat removal (DHR) sequences. These DHR sequences do not lead to early and high radionuclide releases, which is the release category of principal interest in assessing a "large" release. See Section 3.16. The remainder of the fire induced unscreened CDF sequences are due to loss of injection, principally at low pressure. Only a portion of these would lead to early and high releases through the mechanisms described in Section 4 of the Fermi 2 IPE **[4.9]**. Note that most of these scenarios have been conservatively modeled to include three or more SRVs stuck open (due to hot shorts) leading to conditions of a large LOCA. Moreover, the Appendix R analysis DC-4921 **[4.6]** demonstrated that there are no high/low pressure interface concerns due to fire that would result in loss of containment integrity.

Based on the considerations discussed above, it is not expected that containment failure modes due to fire induced accident sequences will differ significantly from those found in the internal events evaluation **[4.9]** nor will the frequency of early and high radionuclide release.

4.5 TREATMENT OF FIRE RISK SCOPING STUDY ISSUES

4.5.1 Seismic/Fire Interactions

Generic Letter 88-20, Supplement 4 **[4.0]**, requires licensees to address issues raised by SANDIA National Laboratories (SNL) in the Fire Risk Scoping Study (FRSS) **[4.11]**. Within the scope of the FRSS are issues related to seismic/fire interaction. Specifically, three primary areas of concern were identified: (1) Seismically induced fires, (2) Seismic actuation of fire suppression systems, and (3) Seismic degradation of suppression systems. These issues were addressed in the Fermi 2 IPEEE program as described below **[4.12]**.

4.5.1.1 Seismically Induced Fires

This issue addresses the concern over potential breakage of flammable liquid or gas vessels during a seismic event that could create fire hazards to the plant. To address this issue, the Seismic Review Team (SRT) performed a seismic screening evaluation of storage vessels containing flammable or combustible materials. This review concentrated on those vessels located in Category I areas of the plant containing safe shutdown equipment. The only flammable liquids in significant quantities in Category I plant areas are diesel fuel and lubricating oil. The following components were considered as part of this review:

- Lube oil (200 gal. total) for the HPCI and CRD pumps in the HPCI pump room in the reactor building;
- Diesel fuel storage tanks (42,000 gal.), day tanks (550 gal.), and lube oil tanks (275 gal.) in each of four tank rooms in the RHR complex;
- Lube oil sump (495 gal.) in each of four diesel generator rooms in the RHR complex; and,
- Fuel oil piping from the tank rooms to the diesel generators in the RHR complex.

Components associated with the first three items were included on the Seismic Margin Assessment (SMA) safe shutdown equipment list (SSEL). Therefore, these items were evaluated by the SRT as part of the seismic screening walkdowns discussed in Section 3.0. All of the items were found to be seismically rugged. The fuel oil piping was also reviewed by the SRT, and it was found to be well supported.

No flammable gases are stored in Category I areas.

Fires at power and industrial facilities can also occur due to the toppling of energized electrical equipment. At Fermi 2, all of the safety related equipment and their anchorages are seismically qualified. Each electrical component on the SSEL was walked down by the SRT and generic anchorage calculations were performed to demonstrate adequate seismic capacity. In addition, the SRT observed non-safety related electrical cabinets in proximity to safety related electrical cabinets. The non safety related cabinets were all positively anchored to preclude toppling.

Based on the results of this review, it is concluded that seismically induced fires with the potential to affect safe shutdown equipment are not likely.

4.5.1.2 Seismic Actuation of Fire Suppression Systems

The effects of inadvertent actuation of suppression systems were previously considered as part of the internal flooding analysis in response to NRC Information Notice 83-41 **[4.13]**. However, some plant equipment credited as being part of the seismic safe shutdown path in the SMA may not have been reviewed. To address this issue for Fermi 2, plant areas with automatic suppression systems were identified.

The automatic suppression systems at Fermi 2 consist of wet pipe sprinkler systems with fusible links, Halon systems, and carbon dioxide systems. Wet pipe sprinkler systems are not used in electrical equipment rooms. Extinguishing materials used in the fire protection system are selected to be compatible with the equipment in the area.

For each plant area with an automatic suppression system, the equipment on the SMA SSEL was compared to that on the Fire Hazards Analysis (FHA) documented in UFSAR Section 9.5. Where differences existed, the components were reviewed to determine their susceptibility to the particular suppression agent (water, Halon, or CO_2). In some instances, functionally redundant equipment was not affected by the same suppression system. Therefore, although actuation may affect equipment of a specific divisional train, redundant equipment would be available for plant shutdown. No cases were found where actuation of the suppression system would cause a loss of safe shutdown capability.

Seismic actuation of the wet pipe sprinkler system could occur only as a result of failure of the fusible link, sprinkler heads, or piping. The system is not actuated electrically. In addition to the review above, the SRT performed a walkdown of all the sprinkler piping and found that an interaction failure is not likely. The piping is well supported and generous clearance is provided around the sprinkler heads to allow them to perform their function.

The Halon and CO_2 systems are controlled electrically; therefore, actuation could occur as the result of relay chatter. A review of the fire protection system control circuitry was conducted to determine if any low ruggedness (bad actor) relays are installed. Results of this review determined that there are only two Westinghouse Type SG relays installed in the fire protection and miscellaneous relay cabinet. Failure of either (or both) of these relays would cause a spurious fire alarm in the fire control panels in the relay room and control room. No other systems or components would be affected. No other low ruggedness relays were found in the fire protection system. In addition, the review included a search for mercury switches; none were found.

Based on the results of this review, seismic actuation of the fire suppression systems at Fermi 2 is not a concern.

4.5.1.3 Seismic Degradation of Suppression Systems

Most of the fire suppression systems in the Category I buildings are wet pipe systems that are actuated by fusible links. Although this system cannot actuate due to relay chatter, the piping can break causing spraying or flooding of safety related equipment. As a result, the SRT reviewed Category I plant areas with wet fire protection systems. This review included both the sprinkler distribution piping as well as the header piping. Headers were followed through various areas of the plant. Similarly, fire protection piping associated with the stand pipes (local hose stations) was reviewed.

During the course of the walkdowns, the SRT noted the following:

- The fire protection piping is well supported. None of the piping spans observed would be characterized as excessive;
- Almost none of the fire protection piping is rod hung. Supports are typically rigid type supports consisting of angles and tube steel members with U-bolts or boxed-in framing;
- Throughout the plant, the piping was noted to be well supported in the transverse lateral direction. Adequate support was provided in the longitudinal direction by transverse supports at changes in piping direction. All of the sprinkler system piping and adjacent headers were noted to have longitudinal support via three-way restraints. The restraints were provided on both horizontal and vertical runs of pipe;
- No instances were noted where cast iron pipe was installed;
- Due to the overall good support, no instances were noted where small, stiff pipes are attached to large, flexible pipes;
- Some of the fire protection piping uses mechanical type fittings (Victaulic couplings or similar). The mechanical fittings are typically used on smaller pipes (6" diameter NPS and under); and
- Some of the fire protection piping uses threaded connections. The threaded connections are typically limited to smaller pipes (2" diameter NPS and under).

Based on the observations of the SRT, gross seismic degradation of the suppression piping is not considered a credible failure that could lead to incapacitation of safety related equipment. Even though the piping uses both mechanical and threaded fittings, the seismic support of the piping would prohibit significant displacement which could lead to gross failure. It was obvious from the SRT review that the fire protection piping is seismically supported to prevent failure and falling, as required by its classification in the UFSAR.

In addition to the wet pipe, the SRT walked down piping associated with the Halon and CO_2 systems. Like the wet pipe, the Halon and CO_2 piping were found to be well supported. Storage tanks and cylinders associated with the Halon and CO_2 systems were all found to be positively anchored or supported.

Based on this review, seismic degradation of the suppression systems at Fermi 2 does not pose a threat to safety-related equipment required for safe shutdown.

4.5.2 Fire Barrier Qualifications

The Fermi 2 Fire Hazards Analysis identified the fire barriers and determined the barrier requirements for the floors, walls, and ceilings enclosing separate fire areas and for the doors and other penetrations through these barriers **[4.3]**. For fire areas not having a three-hour fire rated assembly, the Fermi 2 SSER 5 **[4.64]** analyzed each one individually with respect to its fuel [illegible], its fire suppression and detection systems, and its proximity to safe shutdown equipment and concluded that the fire-rated assemblies provided were adequate for the areas affected and satisfied the guidelines in Section D.1.d and D.1.j of Appendix A to BTP APCSB 9.5-1, and were, therefore, acceptable.

In general, all doorway openings to areas containing safe shutdown equipment or circuits were provided with fire doors with ratings commensurate with the fire ratings of the respective walls, with the following exception. A few doors of special blast-resistant or water-tight construction, or of fire-rated construction less than the respective wall, were evaluated in Fermi 2 SSER 5 and 6 **[4.64, 4.65]** to provide adequate protection in the event of a fire based upon the design and construction of the subject doors and their respective locations within the plant.

The reliability of fire doors at Fermi 2 is ensured by frequent inspections controlled by plant procedures. Many fire doors are provided with electrical supervision of position, and these doors as well as the electrical supervision are verified operable every 31 days by procedures 28.507.02 **[4.41]** and 28.507.03 **[4.42]**. Surveillance procedures 24.000.02 **[4.47]** and 24.000.03 **[4.48]** verify the closed position of unlocked fire doors without electrical supervision once every 24 hours and the locked-closed position of locked fire doors without electrical supervision every 7 days. Procedures 28.507.02 and 28.507.03 verify all fire doors operable by performance of a full functional test every 31 days although the UFSAR Section 9A requirements are only once per 6 months.

In general, the Fermi 2 SSER 5 confirmed that three-hour fire-rated penetration seals were provided for all penetrations of fire-rated walls and floors/ceilings tested in accordance with ASTM E-119 except for penetration seals in the relay room stairwell, which were evaluated to be acceptable based on analysis. Penetration seals were specified by Design Specification 3071-198 **[4.49]** to be qualified by fire tests, and a specific seal type was identified for each penetration on plant drawings. In general, plant procedure 28.507.05 **[4.44]** provides for visual inspection of at least 10% of each type of penetration fire seal once per 18 months with each seal being inspected at least once per 15 years. As a result of the evaluation performed for Supplement 1 to NRC Information Notice 88-04 **[4.55]**, a group of penetration fire seals exposed to a high temperature environment are now specifically inspected in total every 18 months to ensure their operability. Reviews performed for NRC Information Notices 88-04, 88-56 and 94-28 **[4.54, 4.56, 4.57]** concluded that the subject problem conditions had not occurred at Fermi 2 based on the completeness of the Fermi 2 original design specification requirements, the quality of the contractor's installation program, the review and approval of the seal fire tests by the insurance authority, and the continuing seal surveillance program required by the UFSAR.

In general UL listed 3 hour rated fire dampers have been installed in the fire walls at Fermi 2, with certain exceptions. Several 3 hour fire rated barriers contain a single 1-1/2 hour rated fire damper in the duct penetration of the fire barriers, and a few barriers contain two 1-1/2 hour rated fire dampers in series. The Fermi 2 SSER 5 **[4.64]** evaluated these fire barrier arrangements to be acceptable based on the negligible fuel load on either side of the barrier, and the early warning fire detection on each side of the subject barriers which would assure that a fire would be discovered in its incipient stage and be extinguished by the fire brigade within a short time span.

The concern identified in NRC Information Notice 89-52 **[4.53]** involving closure of fire dampers against system air flow was previously resolved for most fire dampers at Fermi 2 by the addition of higher capacity closure springs to the dampers. For several fire damper installation locations, instructions were added to the appropriate fire brigade fire pre-plans to direct manual shutdown of the respective ventilation system in the event of a fire of large magnitude occurring in the area, in order to resolve the fire damper closure concern.

A number of deficiencies had been identified in Fermi 2 fire damper installations including lack of specified expansion gaps for thermal expansion, dampers located outside the plane of the barrier, and other deviations from the fire damper manufacturer's required installation details. Some of these fire damper deficiencies were resolved by new fire tests of similar installation arrangements, some deficiencies were resolved by engineering evaluations which determined that fire loadings of less than 30 minutes equivalent on both sides of a respective barrier did not require fire damper use based on NFPA 90A only requiring dampers in ducts penetrating barriers rated for 2 hours or more, and some damper deficiencies were resolved to be acceptable based on the new fire test results in combination with additional engineering analysis. Some field inspection work remains in order to determine that the fusible link holding straps are properly positioned for certain dampers. These specific fire dampers requiring field inspection are still being considered inoperable until the plant operating mode permits the subject inspection to be completed.

Fermi 2 Procedure 28.507.04 **[4.43]** provides for a visual inspection of all fire dampers every 18 months, and, in addition, provides for a functional test of approximately 10% of the fire dampers by disconnecting the fusible link and confirming the proper closure of the respective fire damper. A continuous or hourly roving fire watch as required is provided to compensate for any fire damper found to be inoperable.

4.5.3 Manual Fire Fighting Effectiveness

4.5.3.1 Fire Reporting and First Aid

At Fermi 2 the initial responses to fire discovery are controlled with plant Procedure 20.000.22, "Plant Fires". The Control Room is alerted to an actual or possible fire condition by observing fire detection or fire suppression system alarms annunciated in the Control Room or by verbal report from person(s) discovering the fire. The plant is provided with a separate independent paging and communication system as well as the site telephone system, both of which can be readily utilized to rapidly report a fire condition to the Control Room. Portable fire extinguishers of appropriate and effective types are located throughout the plant buildings and can be utilized for initial first aid use by plant personnel discovering the fire or by fire brigade personnel during the fire brigade fire fighting response phase of the fire incident. Practical hands-on training with portable fire extinguishers is given to all fire brigade trained personnel as well as all fire watch qualified personnel. Fire brigade trained personnel are included in the operations, security, and fire protection groups.

4.5.3.2 Fire Brigade

A five member Fire Brigade of qualified, trained personnel is designated and maintained at all times, and is separate from the minimum shift crew necessary for safe shutdown of the unit. Fire Brigade members are designated from the operations, security, and fire protection groups, with a typical shift Fire Brigade consisting of one fire brigade leader, two operators, one fire protection inspector, and one nuclear security officer. Satisfactory completion of a special periodic physical examination is required for all qualified brigade members. The Fire Brigade Leader must also hold a Reactor Operator or Senior Reactor Operator License or must possess extensive knowledge of plant systems and their impact on reactor safety, as well as successfully completing a leadership training course.

The Fire Brigade members are each provided with personal protective equipment, including helmets, bunker pants and coats, boots, gloves, self-contained breathing apparatus (SCBA), and flashlights and radios. In addition, extra portable equipment is provided for their use as necessary, including fire hose, nozzles, hose tools, portable extinguishers, foam equipment, smoke ejection equipment poles and axes, and emergency ventilation fans.

4.5.3.3 Fire Brigade Training

All Fire Brigade members receive initial training and continued training every quarter. Initial training includes the following topics:

- Fire Brigade duties, responsibilities, and methodology of response.
- Fire concepts, fire behavior, and methods of extinguishment.
- Use of water, hose and nozzles, and foam.
- Use of portable fire extinguishers used at Fermi.
- Use of SCBA and SCBA communication devices.
- Use of personal protective equipment and fire scene safety practices.
- Radiological hazards in fire emergencies.
- Fire scene ventilation.
- Hazardous materials at Fermi.
- Search and rescue operations.
- Salvage and overhaul practices.
- Use of portable brigade equipment.
- Radio communications.
- Fixed Fire Protection Systems at Fermi including plant tour of these systems and components.
- Fire attack strategy and tactics including use of preplans.
- Fire watch duties.
- Hands-on fireground evolutions including structural fire fighting with full gear and SCBA's

A yearly practice (training) session is also held for each shift Fire Brigade to provide all members with hands-on live fire training in full gear with SCBA's and interior structural fire fighting.

4.5.3.4 Drills And Records

Fire drills, both announced and unannounced, are preplanned and held at regular intervals, with one drill for each shift at least once every 3 months to allow each Fire Brigade to practice as a team. At least one drill is performed each year on a back shift for each shift Fire Brigade, and no less than one drill is unannounced for each shift Fire Brigade per year. All drills are thoroughly critiqued, and at 3-year intervals a randomly selected unannounced drill is evaluated by other qualified individuals as part of the Nuclear Quality Assurance triennial fire protection audit.

Pre-fire plans have been developed for all main plant areas, are required and scheduled to be reviewed on a periodic basis, and are updated as necessary to reflect plant changes. The pre-fire plans are used as a part of the Fire Brigade training.

Records are maintained for all personnel training and for all fire drills in accordance with plant procedures, in order to demonstrate the training level of the Fire Brigade.

All Fire Brigade equipment is inventoried and confirmed to be available at its designated location by procedure on a monthly basis.

4.5.4 Total Environment Equipment Survival

4.5.4.1 Potential Adverse Effects on Plant Equipment By Combustion Products

The detrimental short-term effects of non-thermal combustion products (smoke) on the safe shutdown equipment and circuits required in the event of any specific fire are not believed to be significant for the safe shutdown time lines used in the Fermi 2 analysis. Long-term effects of smoke from a substantial fire condition would not be relevant during the safe shutdown process time lines on which the Fermi 2 analysis is based.

4.5.4.2 Spurious or Inadvertent Fire Suppression Actuation

The subject of seismically induced inadvertent actuations of fire suppression systems has been discussed in section 4.5.1. This section will only discuss other possible actuations of the Fermi 2 fire suppression systems.

Generically, the fire protection systems had been designed for Fermi 2 with the philosophy of minimizing water near electrical equipment. When water has been used indoors in safety related areas, sprinkler systems are employed rather than deluge systems. Each sprinkler is an automatic device and will only discharge water when heated to its rated actuation temperature. An automatic sprinkler has an extremely low failure rate, and even then water would only be discharged from a single sprinkler, affecting only a very limited area with the discharged water being readily accepted by the plant drain system. As a result of a review of a Significant Event Report from another plant, the design of several Fermi 2 sprinkler systems was modified to increase the sprinkler actuation temperatures, and reduce the possibility of sprinkler actuation caused by a steam line break or abnormal room temperature.

Gaseous type fire suppression systems (CO_2 or Halon) are provided for several areas and equipment types, including emergency diesel generators, relay room, cable tunnel, cable spreading room, cable tray area on 630 ft. elevation of the auxiliary bldg., computer rooms, and standby gas treatment system filters. In general, inadvertent actuation of any of these gaseous systems will not cause loss of function of Class 1E equipment since the equipment can operate in a gaseous environment. Closing of some HVAC dampers, upon system actuation, will result in loss of cooling to respective areas, until manual actions are taken to re-open dampers and establish air flow. Inadvertent actuation of a CO_2 system into an emergency diesel generator (EDG) room will not affect the operation of the EDG, since separate combustion air is provided for the engine by direct connection to the outside.

4.5.4.3 Operator Action Effectiveness

Abnormal Operating Procedure 20.000.18 **[4.38]** would be utilized by the operating personnel in the event that a fire in the Main Control Room or other designated fire zones results in the need to control and shutdown the plant from the Dedicated Shutdown Panel.

If a significant fire occurred in any other area of the plant, control and shutdown would be performed from the Main Control Room utilizing normal and abnormal operating procedures as required. Operations personnel are trained on all operating procedures as required by their positions and responsibilities.

4.5.5 Control Systems Interactions

The alternative (dedicated) shutdown system at Fermi 2 has been designed and installed to meet the technical requirements of 10CFR50, Appendix R, Sections III.G.3 and III.L. This system provides safe-shutdown capability separate and remote from the Control Room complex and certain other designated plant fire zones. The system would be used when a fire within the Control Room complex or the other designated fire zones is determined to have significantly damaged the safe-shutdown equipment/cabling within these zones.

The alternative shutdown system consists of a Dedicated Shutdown Panel (also referred to as the 3L panel) and associated instrumentation, one combustion turbine generator (CTG), the standby feedwater (SBFW) system, and Division I portions of the following systems:

- RHR
- RHRSW
- Emergency Equipment Cooling Water (EECW)
- Emergency Equipment Service Water (EESW)

The Dedicated Shutdown Panel is supplemented by local manual operator actions to achieve hot or cold shutdown. The Dedicated Shutdown Panel is a local operation station, remote from the fire areas of concern including the Main Control Room, with instrumentation and control switches and transfer switches necessary for operating the SBFW system required to keep the reactor core covered with water.

Hot and cold shutdown can be achieved from the Dedicated Shutdown Panel with manual operator action required locally in the Reactor/Auxiliary Building and RHR Complex.

Local operation includes controlling equipment at local panels, switchgear, MCC's, distribution panels, and valves. In the event of a serious Control Room fire, or fire in certain other designated fire zones that would significantly affect the controls in the Main Control Room, Shutdown operations would be controlled by use of Abnormal Operating Procedure 20.000.18 **[4.38]**.

Calculation No. 1020-014-300 titled "Dedicated Shutdown Design Review" **[4.50]**, performed by Impell Corp., evaluated the equipment, circuits, cable routing, and transfer schemes for the Fermi 2 dedicated (alternative) shutdown system and concluded that it was in complete compliance with 10CFR50, Appendix R. The methodology for this calculation review identified the circuits required to be functional as well as the adequacy of the transfer schemes, and verified that the essential cable routes were independent of the Main Control Room and other designated alternate shutdown areas by location or protection.

4.6 USI A-45 AND OTHER SAFETY ISSUES

4.6.1 Unresolved Safety Issue A-45

USI A-45, "Shutdown Decay Heat Removal Requirements", was completed as part of the Fermi 2 IPE submittal **[4.9]**. The documentation of this resolution is provided in sections 3.4.3 and 7.1.3 of the IPE submittal.

The FIVE analysis considered loss of decay heat removal capability through the PSA evaluation of shutdown capability with equipment not damaged by the fire. Of the six compartments that did not meet the screening criteria of <1E-06/yr, (see Table 4-14) two compartments had core damage frequencies (CDFs) dominated by loss of decay heat removal (RB06 and 04ABN), and one compartment (11ABE) was about equally divided between loss of decay heat removal and loss of injection at low pressure. However, the dominant scenarios for all six compartments were evaluated with regard to the absolute contribution to CDF due to loss of decay heat removal sequences using the end state results produced by the PSA model. The resulting CDF from the dominant fire scenarios for actual loss of decay heat removal sequences (DHR category in the Fermi 2 IPE or TW in WASH-1400) is 5.4E-6/yr. Adding this value to the internal events DHR contribution of 1.7E-6/yr for the Fermi 2 IPE brings the total DHR contribution to 7.1E-6/yr.

However , it is recognized that the NRC staff expanded the functional definition of decay heat removal for BWRs in A-45 to also include loss of injection (except for large LOCA sequences) and station blackout scenarios (see discussion in Section 3.4.3 of the Fermi 2 IPE report **[4.9]**). The major impact of the use of this expanded definition on the fire study results is the addition of loss of injection at low reactor pressure sequences since two of the unscreened compartments are dominated by such sequences. The resulting CDF for the dominant fire scenarios using the expanded NRC definition of loss of decay heat removal for fire induced transients is 1.6E-5/yr. Adding the corresponding contribution from the Fermi 2 IPE of 3.6E-6/yr brings the total CDF for the expanded NRC DHR loss contribution to 2.0E-5/yr.

Both of these CDF values remain below the Category 1 criteria ("acceptably small or reducible to an acceptable level by simple improvements") of less than 3.0E-5/yr cited in NUREG-1289 **[4.66]** and in Figure 3.4-8 of the Fermi 2 IPE **[4.9]**. Moreover, the larger value of 1.6E-5/yr associated with the expanded definition of decay heat removal is only slightly greater than the 1.0E-5/yr value stipulated in NUREG-1289 as the interim quantitative design objective. If the conservative assumptions inherent in the FIVE methodology were modified to be approximately consistent with the best estimate approach typically utilized in PSAs to assess risk, there is little doubt that the fire related NRC staff definition of loss of decay heat removal contribution to CDF would drop below the 1.0E-5/yr criterion.

It is thus concluded that the risk of loss of decay removal as a result of potential internal fires is not a vulnerability, and thus the decay heat removal capability following fire induced transients is adequate.

4.6.2 Generic Safety Issue 57

The issues resulting from GI-57 **[4.14]** addressed in section 4.8 of this submittal are as follow:

1. Mercury relays in fire protection control systems that could potentially disable safety related equipment or impair the operators' ability to perform corrective action.

2. Cabinets or components required for safe shutdown being impaired due to water intrusion following actuation of a sprinkler or deluge system.

3. Loss of fire water suppressant capabilities following an earthquake due to loss of offsite power and failure of the diesel fire pump batteries.

4. Seismic/fire interaction in switchgear rooms damaging safety related cables routed over or near non-seismically designed switchgear cabinets.

5. Seismic/fire interaction in cable spreading rooms due to non-seismically designed electrical cabinets.

It was determined that there are no significant design problems in the fire protection system that would result in impairment to safety-related equipment.

In addition to the review for seismic/fire interactions, the FIVE analysis considered potential damage to cables and equipment routed over or near switchgear and other electrical cabinets.

The switchgear rooms were analyzed to determine the probability of an internal fire affecting cables. Sections 4B.2.3.4 and 4B.2.3.14 addressed this issue and determined that a fire would be contained within the switchgear and that no cables routed above or near the switchgear would be damaged. Therefore, the risk of damage to safety-related equipment or cables was determined to be negligible.

The cable spreading room and similar areas were analyzed to determine the probability of fires damaging cables. The design of these areas resulted in few, if any, electrical cabinets being installed. The electrical cabinets in these areas were fire protection panels, Sections 4B.2.3.9 and 4B.2.3.10 and were determined to be insignificant ignition sources. Therefore, the risk of damage to safety-related equipment or cables was determined to be negligible.

Based on the above, no vulnerabilities or additional insights related to Generic Issue 57 from the FIVE assessment have been found that require incorporation in plant procedures .

4.7 INSIGHTS AND RECOMMENDATIONS

4.7.1 Reactor Building

All compartments of the Reactor Building met the screening criteria of <1.0E-6/yr except for the second floor (06RB). This compartment's final screening value is 1.00E-6/yr, which is right at a value that could be assumed to screen. As discussed in section 4A.3.4.2.5 the conservative assumption of failing all equipment powered from these cabinets could have been resolved by additional analysis and would most likely have allowed this area to screen.

The principal insights gained from a review of this compartment is that the dominating fire sequences are for fires in cabinets that are part of the Dedicated Shutdown System. Thus, they have the potential for isolating equipment from their primary control location, the main control room. These cabinets are thus more than just "backup devices" in terms of the impact of their loss.

Discussions were held between the Fire Task Manager and the Operations Training Supervisor to determine whether the Plant Operators were aware of the potential effects of loss of these cabinets. The result of this meeting satisfied both the Fire Protection Engineer and the Operation Training Supervisor that the Operator training was adequate.

A discussion between the Fire Task Manager and the Fire Brigade Training Instructor resulted in a decision to perform additional drills in the vicinity of these electrical cabinets in order to increase the awareness of the fire brigade members to the need to quickly isolate and extinguish fires that may damage the cabinets. These additional drills will be performed following development of the training plan. This plan will be developed following a walkdown of the area by the Fire Protection Engineer and the Instructor to clarify the content of the training plan.

4.7.2 Auxiliary Building

Five compartments in the Auxiliary Building did not meet the FIVE screening criteria of <1.0E-6/yr. These compartments are the Control Room (CDF 4.27E-6/yr), Division 1 Switchgear (CDF 4.51E-6/yr), Relay Room (CDF 2.77E-6/yr), Division 2 Switchgear (CDF 2.54E-6/yr) and the Division 1 portion of the Miscellaneous Room on Elevation 643'-6" (CDF 1.9E-6/yr).

No new insights were gained from the evaluation of these compartments. These compartments were already recognized as having a major role in the prevention of core damage, and the magnitude of the CDF in excess of the screening criterion is judged to be modest. These compartments contain all the cables and equipment needed to achieve normal shutdown. Because they contain these cables and equipment, all cables supporting equipment required to achieve shutdown using the dedicated shutdown procedure **[4.67]** were routed independent of these areas.

Therefore, there are no additional actions required for these unscreened Auxiliary Building compartments and no further analysis will be performed.

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Table 4- 1 Fire Areas

FIRE AREA	DESCRIPTION	ELEVATION
01AB	Aux. Bldg. Basement	551' - 0" and 562' - 0"
02AB	Mezzanine and Cable Tray Area	583' - 6" and 603' - 6"
03AB	Relay Room	613' - 6"
04AB	Division 1 Switchgear Room	613' - 8 "
05ABE	Division 1 Cable Tunnel	613" - 6"
05ABW	Division 2 Cable Tunnel	613" - 6"
06AB	Miscellaneous Rooms	613" - 6"
07AB	Cable Spreading Room	630" - 6"
08AB	Cable Tray Area	631' - 0"
09AB	Control Room	643' - 6" and 655' - 6"
10ABE	Division 1 Battery Room	643' - 6"
10ABW	Division 2 Battery Room	643' - 6"
11AB	Miscellaneous Rooms	643' - 6"
12AB	Division 2 Switchgear Room	643' - 6"
13AB	Ventilation Equipment Area	650' - 6"
14AB	Control Center Ventilation Equipment Rooms and Standby Gas Treatment Rooms	677' - 6"
15AB	Ventilation Equipment Area	677' - 6"
01RB	Torus Room	540' - 0"
02RBNE	Basement Corner Room	540' - 0" and 562' - 0"
02RBNW	Basement Corner Room	540' - 0" and 562' - 0"
02RBSE	Basement Corner Room	540' - 0" and 562' - 0"
02RBSW	Basement Corner Room	540' - 0" and 562' - 0"
03RB	HPCI pump and Turbine and CRD pump rooms	540' - 0" and 562' - 0"
04RB	Corridor zone	562' - 0" and 564' - 0"
05RB	First Floor	583' - 6"
06RB	Second Floor	613' - 6"
07RB	Third Floor	641' - 6"
08RB	Fourth Floor	659' - 6"
09RB	Fifth Floor	684' - 6"
10RB	Drywell	562' - 0" to 684' - 6"
11RHR	Division 1 RHR	554' - 3" to 617' -0"
12RHR	Division 1 RHR	554' - 3" to 617' -0"
13RHR	Division 2 RHR	554' - 3" to 617' -0"
14RHR	Division 2 RHR	554' - 3" to 617' -0"
ABFST	Aux. Boiler Fuel Oil Storage Tank	treated as part of YARD
ABH	Aux. Boiler House	treated as part of YARD
CST	Condensate Storage Tank	treated as part of YARD
EF1	Fermi 1	n/a
GSWPH	General Service Water Pump House	n/a
HSF	Hydrogen Storage Facility	treated as part of YARD

OBA & TSC	Office Building Annex & Technical Support Center	n/a
ONSB	Onsite Storage Building	n/a
OSB	Office and Service Building	n/a
RW	Rad Waste Building	
TB	Turbine Building	
TRANS	Transformers	n/a
USRCD	Underground Safety Related Conduit Duct	n/a
YARD	Yard	n/a

Table 4- 2 FIRE COMPARTMENTS

FIRE COMPARTMENT	DESCRIPTION	ELEVATION
01AB	Aux. Bldg. Basement	551' - 0" and 562' - 0"
02AB	Mezzanine and Cable Tray Area	583' - 6" and 603' - 6"
03AB	Relay Room	613' - 6"
04ABN	Division 1 Switchgear Room	613' - 8 "
04ABS	Division 2 Cable Chase in Div. 1 Switchgear Room	613' - 8 "
05ABE	Division 1 Cable Tunnel	613" - 6"
05ABW	Division 2 Cable Tunnel	613" - 6"
06AB	Miscellaneous Rooms	613" - 6"
07AB	Cable Spreading Room	630" - 6"
08AB	Cable Tray Area	631' - 0"
09AB	Control Room	643' - 6" and 655' - 6"
10ABE	Division 1 Battery Room	643' - 6"
10ABW	Division 2 Battery Room	643' - 6"
11ABE	Miscellaneous Rooms - Majority of Area Division 1	643' - 6"
11ABW	Miscellaneous Rooms - Division 2 Battery Charger Area	643' - 6"
12AB	Division 2 Switchgear Room	643' - 6"
13AB	Ventilation Equipment Area	650' - 6"
14AB	Control Center Ventilation Equipment Rooms and Standby Gas Treatment Rooms	677' - 6"
15AB	Ventilation Equipment Area	677' - 6"
01RB	Torus Room	540' - 0"
02RBNE	Basement Corner Room	540' - 0" and 562' - 0"
02RBNW	Basement Corner Room	540' - 0" and 562' - 0"
02RBSE	Basement Corner Room	540' - 0" and 562' - 0"
02RBSW	Basement Corner Room	540' - 0" and 562' - 0"
03RB	HPCI pump and Turbine and CRD pump rooms	540' - 0" and 562' - 0"
04RB	Corridor zone	562' - 0" and 564' - 0"

Table 4- 2 FIRE COMPARTMENTS

FIRE COMPARTMENT	DESCRIPTION	ELEVATION
05RB	First Floor	583' - 6"
06RB	Second Floor	613' - 6"
07RB	Third Floor	641' - 6"
08RB	Fourth Floor	659' - 6"
09RB	Fifth Floor	684' - 6"
10RB	Drywell	562' - 0" to 684' - 6"
RHR1	Division 1 RHR - EDG compartment and Pump Rooms	554' - 3" to 617' -0"
RHR1SG	Division 1 RHR - Switchgear	554' - 3" to 617' -0"
RHR2	Division 2 RHR - EDG Compartment and Pump Rooms	554' - 3" to 617' -0"
RHR2SG	Division 2 RHR - Switchgear	554' - 3" to 617' -0"
ABFST	Aux. Boiler Fuel Oil Storage Tank	treated as part of YARD
ABH	Aux. Boiler House	treated as part of YARD
CST	Condensate Storage Tank	treated as part of YARD
EF1	Fermi 1	n/a
GSWPH	General Service Water Pump House	n/a
HSF	Hydrogen Storage Facility	treated as part of YARD
OBA & TSC	Office Building Annex & Technical Support Center	n/a
ONSB	Onsite Storage Building	n/a
OSB	Office and Service Building	n/a
RW	Rad Waste Building	
TB	Turbine Building	divided into four compartments: Basement, 1st floor, 2nd floor, 3rd floor, Waste Oil Room.
TRANS	Transformers	n/a
USRCD	Underground Safety Related Conduit Duct	n/a

Table 4- 2 FIRE COMPARTMENTS

FIRE COMPARTMENT	DESCRIPTION	ELEVATION
YARD	Yard	n/a

Table 4- 3 CREDITED BALANCE OF PLANT SYSTEMS

SYSTEM NUMBER	DESCRIPTION
C41	STANDBY LIQUID CONTROL SYSTEM
C71	REACTOR PROTECTION SYSTEM (PORTIONS ONLY)
E11	RHR SERVICE WATER
G3352	RWCU OUTBOARD CONTAINMENT ISOLATION VALVES
H11	MISCELLANEOUS INSTRUMENT RACKS
N20	CONDENSATE SYSTEM
N21	FEEDWATER SYSTEM
N3021	MAIN STEAM BYPASS VALVES
N71	CIRCULATING WATER
P11	CONDENSATE STORAGE AND TRANSFER
P41	GENERAL SERVICE WATER SYSTEM
P42	RBCCW
P43	TBCCW
P50	STATION AIR
R14	MISCELLANEOUS SWITCHGEAR
T46	STANDBY GAS TREATMENT
T49	PRIMARY CONTAINMENT PNEUMATIC SUPPLY
V41	RADWASTE BUILDING CHILLER

Table 4- 4 CONDITIONAL CORE DAMAGE ANALYSES TABULATION

Fire Zone	OS	DA	DB	EDGs	BUSES	CF	B1	B2	DC	NP	LV	TB	CC	C1	C2	SA	A1	A2	PT	V3	CT	SF	RC	HP	LA	LB	CS	OV	CCDF
BASE																													1.3E-05
10AB1		X																											4.0E-04
10AB2			X																										3.3E-04
11AB1		X																		X			X					X	6.5E-03
11AB2			X																X				X	X		X		X	2.2E-03
11AB3			X																X		X	0.1							1.3E-03
12AB1			X	3, 4											X			X		X						X	2		8.6E-04
12AB2	2		X	3,4											X			X		X		X				X			3.7E-02
12AB3					F																	0.1							6.6E-04
12ABC					E																	0.1							3.3E-05
12AB4	2			3	E																								6.2E-04
12AB5				3	E																								1.7E-05
12AB6	2			4	F																								1.2E-03
12AB7				4	F																								4.1E-04
12AB8	2			3	E																								6.2E-04
12AB9				3	E																								1.4E-05
12A10	2			4	F																								1.2E-03
12A11				4	F																								1.5E-04
13AB1																			X	X			X	X	X		1	X	3.0E-03
13AB2																			X	X			X	X		X	1	X	5.2E-03
13AB3																			X	X			X	X	X		1		2.2E-03
13AB4																			X	X			X	X		X	1		1.2E-03
14AB1					C, F																		X					X	1.0E+00
14AB2																						0.1	X					X	2.0E-04
15AB2																						X							3.4E-05
15AB3																						0.1							3.3E-05
1AB01				1, 2										X			X	X		X		X	X	X	X		1	X	3.0E-03
1AB02				3, 4														X				X		X		X	2	X	3.9E-03
F1AB01				1, 2										X			X	X		X		0.1	X	X	X		1	X	3.0E-03
F1AB02				3, 4														X				0.1		X		X	2	X	3.9E-03
1AB3																	X					0.1							3.3E-05
1AB4																		X				0.1	X	X					6.1E-04

Table 4- 4 CONDITIONAL CORE DAMAGE ANALYSES TABULATION

Fire Zone	OS	DA	DB	EDGs	BUSES	CF	B1	B2	DC	NP	LV	TB	CC	C1	C2	SA	A1	A2	PT	V3	CT	SF	RC	HP	LA	LB	CS	OV	CCDF
2AB01				3, 4	E,F							X	X		X	X		X	X	X			X	X		X			3.5E-02
2AB02				1, 2	B,C							X	X	X		X	X		X	X			X	X	X				3.3E-02
3AB01																									X				2.4E-05
F3AB01																						X			X				2.7E-04
3AB02																							X						4.5E-05
3AB03																											1		9.0E-06
3AB04																				AD									8.6E-06
3AB05				1, 2																									9.1E-06
3AB06																										X			3.6E-05
3AB07																											2		9.0E-06
3AB08				3, 4																									9.4E-06
3AB10		X												X			X	X	X	X			X	X	X		1		4.4E-03
F3AB10		X												X			X	X	X	X		X	X	X	X		1		4.4E-03
3AB11														X			X	X		X			X		X		1		3.0E-03
F3AB11		X												X			X	X		X		X	X		X		1		3.0E-03
3AB12		X												X															2.4E-04
3AB13		X												X			X	X	X	X			X		X		1		4.4E-03
F3AB13		X												X			X	X	X	X		X	X		X		1		4.4E-03
3AB14														X			X	X		X			X	X	X		1		3.0E-03
F3AB14														X			X	X		X		X	X	X	X		1		3.0E-03
3AB15		X		1, 2	B, C									X			X	X	X	X				X	X		1	X	4.4E-03
3AB16					C																								1.3E-04
3AB17					C									X			X	X		X			X	X	X		1	X	3.1E-03
F3AB17					C									X			X	X		X		X	X	X	X		1	X	3.1E-03
3AB19				3, 4	F								X		X			X	X	X						X	2	X	3.5E-02
3AB20					F													X								X		X	1.5E-04
3AB21													X		X					X				X		X	2	X	3.6E-02
3AB23													X		X			X	X	X				X		X	2	X	3.6E-02
3AB24													X		X			X	X								2		4.7E-05
3AB25																								X					2.9E-05
3AB26												X	X			X	X					X							4.6E-04
4AB01			X	3, 4									X		X	X		X						X		X	2	X	2.1E-03

Table 4- 4 CONDITIONAL CORE DAMAGE ANALYSES TABULATION

Fire Zone	OS	DA	DB	EDGs	BUSES	CF	B1	B2	DC	NP	LV	TB	CC	C1	C2	SA	A1	A2	PT	V3	CT	SF	RC	HP	LA	LB	CS	OV	CCDF
4AB02		X		1, 2									X	X		X	X	X	X	X			X	X	X		1	X	3.4E-02
4AB03			C	3,4											X	X						0.1		X		X	2	X	2.0E-03
4AB04	1	X		1,2									X	X		X	X		X	X		0.1	X		X		1	X	3.4E-02
4AB05					C																	0.1							1.3E-03
4AB06					B																	0.1							7.0E-04
4AB07						C																0.1							3.5E-05
4AB08	1			1	B																								3.6E-04
4AB09				1	B																								4.4E-04
4AB10	1			2	C																								1.9E-03
4AB11				2	C																								8.1E-04
4AB12	1			1	B																								3.6E-04
4AB13				1	B																								1.4E-04
4AB14	1			2	C																								1.9E-03
4AB15				2	C																								1.6E-04
5AB01		X		1, 2										X					X	X			X		X		1	X	4.4E-03
5AB02			X	3, 4									X		X				X	X				X		X	2	X	3.7E-02
5AB03	1			1,2										X					X	X		0.1	X		X		1	X	3.3E-02
5AB04	2			3,4									X		X				X	X		0.1		X		X	2	X	3.5E-02
6AB01		X		1, 2									X						X	X			X		X	X	1	X	1.0E+00
6AB02		X		1,2		X							X	X					X	X		0.1	X				1	X	6.3E-02
9AB01													X		X					X				X		X	2		1.0E-02
9AB02												X							X		X								2.4E-04
9AB03																													1.3E-05
9AB04					B, C																								4.3E-03
9AB05					E, F																								2.9E-03
9AB06	1,2																												8.0E-04
9AB07																X		X							X				2.5E-05
9AB08																												X	1.2E-05
ABH																					X								2.1E-04
ABH1																					X								7.2E-06
EF1					B							X	X																2.5E-03
GSWPH												X	X																5.1E-05

Table 4- 4 CONDITIONAL CORE DAMAGE ANALYSES TABULATION																													
Fire Zone	OS	DA	DB	BDGs	BUSES	CF	B1	B2	DC	NP	LV	TB	CC	C1	C2	SA	A1	A2	PT	V3	CT	SF	RC	HP	LA	LB	CS	OV	CCDF
RADWST	1,2						X	X	X	1	1	X	X			X				X	X	X							1.4E-02
RW1							X					X	X			X					X	X							4.6E-04
RW2								X				X	X			X					X	X							4.6E-04
RW3							X	X				X	X			X					X								5.1E-04
RW4							X	X	X			X	X			X					X	X							4.6E-04
RW5							X					X	X			X					X								4.6E-04
RW6								X	X			X	X			X					X	X							5.1E-04
RW7									X			X	X			X					X								5.2E-04
RW8												X	X			X					X							X	4.7E-04
RW9												X	X			X					X	X							4.7E-04
RB01		X	X		B, C, E, F	X								X	X		X	X					X	X	X	X	X		1.0E+00
RB01N		X			B, C	X								X			X						X	X	X		1		1.5E-02
RB01A			X		E,F									X	X		X	X						X		X	X		4.4E-03
RB01B		X			B,C									X	X		X	X					X	X	X		X		4.5E-03
RB01C						X																	X	X			X	X	6.3E-02
RB02NEa		X			B, C												X						X	X	X		1		4.5E-03
RB02NE																	X						X	X	X		1		4.2E-04
RB2NE1																	X						X	X	X		1		7.8E-05
RB02NWa					B, C									X					X					X	X				4.2E-03
RB02NW														X					X					X	X				2.7E-04
RB02SEa					E, F																		X	X			2		4.5E-03
RB02SE																							X	X			2		1.8E04
RB02SWa					E, F														X					X		X	2		4.4E-03
RB02SW																			X					X		X			3.1E-04
RB03			X		E																			X			2		3.5E-04
RB04D1		X			B, C									X			X						X		X		1		4.2E-03
RB04D2			X		E, F													X						X		X	2		4.4E-03
RB05D1		X		1, 2	B, C	X				1	1		X	X		X	X		X	X			X		X		1		7.0E-02
RB05D2			X		E, F	X				2	2		X		X	X		X	X	X				X		X	2		7.1E-02
RB05SG																X												X	9.0E-05
RB05S1																X							X					X	1.2E-04
RB05S2															X	X												X	9.0E-05

Table 4- 4 CONDITIONAL CORE DAMAGE ANALYSES TABULATION

Fire Zone	OS	DA	DB	EDGs	BUSES	CF	B1	B2	DC	NP	LV	TB	CC	C1	C2	SA	A1	A2	PT	V3	CT	SF	RC	HP	LA	LB	CS	OV	CCDF
RB05S3																X							X				1	X	1.2E-04
RB05S4														X		X												X	8.9E-05
RB05S5					E											X												X	1.2E-04
RB05S6					B											X												X	4.3E-03
RB5S6A					B											X												X	4.2E-03
RB05S7																X						X						X	2.3E-04
RB05S8																X						X	X				1	X	9.4E-04
RB05S9																X									X			X	3.7E-03
RB5S10			C													X										X	2	X	3.7E-03
RB5S11																X		X							X		1	X	3.7E-03
S002B																	X										1		2.7E-06
MS002B																											1		2.7E-06
RB06D1		X		1, 2	B, C	X				1	1		X	X		X				X			X		X		1		7.0E-02
RB06D2			X		E, F	X				2	2		X		X	X				X				X		X	2		7.1E-02
RB6S01														X		X												X	8.9E-05
RB6S02															X	X												X	8.9E-05
RB6S03					C											X												X	7.9E-03
RB6S04																X												X	8.9E-05
RB6S05						X										X												X	6.9E-04
RB6S06																X				X					X	X	1	X	1.0E+00
RB6S07							X									X												X	3.2E-04
RB6S08										1	1					X											1	X	1.5E-04
RB6S09										1	1					X												X	1.5E-04
RB6S10										2	2					X												X	1.6E-04
RB6S11					E, F										X	X												X	4.4E-03
RB6S12					F											X												X	4.3E-03
RB6S13					E								X		X	X										2	2	X	4.4E-03
RB6S14													X			X									2		2	X	6.1E-03
S003B														X										X	X		1	X	7.9E-05
MS003B														X										X	X		1	X	3.9E-03
S005C															X								X			X	2		6.5E-06
MS005C															X								X			X	2		6.9E-05

Table 4- 4 CONDITIONAL CORE DAMAGE ANALYSES TABULATION

Fire Zone	OS	DA	DB	EDGs	BUSES	CF	B1	B2	DC	NP	LV	TB	CC	C1	C2	SA	A1	A2	PT	V3	CT	SF	RC	HP	LA	LB	CS	OV	CCDF
RB07			X																	X			X	X					4.7E-03
RB7S01																													2.2E-05
RB7S02																								X			1		4.0E-05
RB08																							X						5.5E-05
RB09					F					X	X								X				X	X		X	2		5.3E-04
RB09A																													2.2E-05
RB10																													2.2E-05
RHR1		X		1, 2	B, C				X					X															2.9E-02
RHR1S1				1,2	B,C																								4.3E-03
RHR1S2				1,2	B,C									X															4.3E-03
R1S01				1,2																									1.3E-05
R1S02				1,2	B																								3.3E-04
R1S03				1,2	C																								6.1E-04
R1S04				1,2	B,C																								4.3E-03
R1S05																													1.5E-04
R1S06				2	C																								6.1E-04
R1S07														X															1.3E-05
R1S08				1	B																								3.3E-04
R1S09				2	C																								6.1E-04
RHR2			X	3, 4	E, F										X														4.4E-03
R2S01				4	F																								2.9E-04
R2S02				3	E																								1.3E-05
R2S03				3,4																									1.3E-05
R2S04																													2.8E-04
R2S05			X																										3.3E-04
TB		X	X		B, C				X	1	1	X	X			X				X	X	X	X	X					1.0E+00
TB1												X	X			X					X		X	X					4.1E-03
TB2												X	X			X					X	X							4.7E-04
TB3					B							X	X			X					X	X							4.3E-03
TB3A												X	X			X					X	X							9.3E-04
TB3B	1,2											X	X			X					X	X							6.1E-03
TB3B1	1,2																					A							6.1E-03

Table 4- 4 CONDITIONAL CORE DAMAGE ANALYSES TABULATION

Fire Zone	OS	DA	DB	EDGs	BUSES	CF	B1	B2	DC	NP	LV	TB	CC	C1	C2	SA	A1	A2	PT	V3	CT	SF	RC	HP	LA	LB	CS	OV	CCDF
TB4									X			X	X			X					X	X							5.2E-04
TB5									X			X	X			X					X								5.2E-04
TB6		X	X		B,C				X	1	1	X	X			X				X	X	X							1.0E+00
TB7					B,C				X			X	X			X					X	X							5.2E-03
TB8	1						X																						9.3E-04
TB9																						X							4.7E-04
TB10																						2							4.7E-04
TB11	1						X															2							9.3E-04
TB12	1						X		X																				9.9E-04
TB13	1						X															X							9.3E-04
TRANS	1						X																						2.0E-04
OFF2	2							X																					2.1E-04
USRCD		X	X	1, 2, 3 ,4	B, C, E ,F				X																				1.0E+00
USRCD1		X		1,2	B,C				X																				2.9E-02
USRCD2			X	3,4	E,F				X																				5.2E-03
YARD	1,2				B, C		X	X				X	X								X								6.5E-02
YARD1	2				C			X				X	X								X								6.9E-02
YARD2	1				B,C		X					X	X								X								5.2E-03
YARD3					B							X	X								X								4.5E-03
YARD4												X	X								X	X							4.6E-04
YARD5												X	X								X								4.6E-04

NOTES:	
X	Corresponding top event failed
1,2,3,4 (EDGs)	Diesel 11, 12, 13, 14, respectively
1,2 (OS)	Division 1 and 2, respectively, of offsite power failed
1,2 (CS)	Division 1 and 2, respectively, of core spray failed
B,C,E,F (BUSES)	Top events B4, C4, E5, F5, respectively, failed
0,1 (SBFW)	Remote start of SBFW required with additional human error probability of 0.1 per demand

Table 4- 5 Initial Phase II Results - all areas except Aux. Bldg.				
FIRE COMPARTMENT	**COMPARTMENT FIRE FREQUENCY (F1)**	**CCDF (P2)**	**CDF (F_2) (F1 x P2)**	**SCREEN**
ABFST	5.68E-2	1.3E-5	7.38E-7	YES
ABH	8.72E-3	7.2E-6	6.28E-8	YES
CST	5.68E-2	1.3E-5	7.38E-7	YES
EF1	1.76E-3	2.5E-3	4.40E-6	NO
GSWPH	6.00E-3	5.1E-5	3.06E-7	YES
HSF	5.68E-2	1.3E-5	7.38E-7	YES
OBA&TSC	5.68E-2	1.3E-5	7.38E-7	YES
ONSB	5.68E-2	1.3E-5	7.38E-7	YES
OSB	5.68E-2	1.3E-5	7.38E-7	YES
RB01	1.44E-3	1.0	1.44E-3	NO
02RBNE	2.97E-3	7.8E-5	2.3E-7	YES
02RBNW	2.98E-3	2.7E-4	8.05E-7	YES
02RBSE	3.81E-3	1.8E-4	6.86E-7	YES
02RBSW	2.74E-3	3.1E-4	8.49E-7	YES
RB03	3.39E-3	3.5E-4	1.19E-6	NO
RB04	1.23E-3	8.5E-3	1.05E-5	NO
RB05	1.27E-2	2.8E-2	3.56E-4	NO
RB06	1.02E-2	2.9E-2	2.96E-4	NO
RB07	9.20E-2	5.3E-3	4.88E-4	NO
RB08	7.03E-3	5.5E-5	3.87E-7	YES
RB09	3.45E-3	5.3E-4	1.83E-6	NO
RB10	4.33E-3	2.2E-5	9.53E-8	YES
RHR1	3.66E-2	2.9E-2	1.06E-3	NO
RHR2	3.71E-2	4.4E-3	1.63E-4	NO
RW	1.15E-2	1.4E-2	1.61E-4	NO
TB	5.68E-2	1.0	5.68E-2	NO
TRANS	2.51E-3	2.0E-4	5.02E-7	YES
USRCD	9.00E-4	1.0	9.00E-4	NO
YARD	3.75E-3	6.5E-2	2.44E-4	NO

Table 4- 6 Final Phase II Results - all areas except Aux. Bldg.						
FIRE COMPARTMENT	**COMPARTMENT FIRE FREQUENCY (F1)**	**INITIAL CCDF (P2)**	**INITIAL CDF (F_2) (F1 x P2)**	**SCREEN**	**REVISED TOTAL CDF (F_2)**	**SCREEN**
ABFST	5.79E-2	1.3E-5	7.38E-7	YES		
ABH	8.72E-3	7.2E-6	6.28E-8	YES		
CST	5.79E-2	1.3E-5	7.38E-7	YES		
EF1	1.76E-3	2.5E-3	4.40E-6	NO	1.80E-7	YES
GSWPH	6.00E-3	5.1E-5	3.06E-7	YES		
HSF	5.79E-2	1.3E-5	7.38E-7	YES		
OBA&TSC	5.79E-2	1.3E-5	7.38E-7	YES		
ONSB	5.79E-2	1.3E-5	7.38E-7	YES		
OSB	5.79E-2	1.3E-5	7.38E-7	YES		
RB01	1.44E-3	1.0	1.44E-3	NO	4.6E-7	YES
02RBNE	2.97E-3	7.8E-5	2.3E-7	YES		
02RBNW	2.98E-3	2.7E-4	8.05E-7	YES		
02RBSE	3.81E-3	1.8E-4	6.86E-7	YES		
02RBSW	2.74E-3	3.1E-4	8.49E-7	YES		
RB03	3.39E-3	3.5E-4	1.19E-6	NO	7.5E-7	YES
RB04	1.23E-3	8.5E-3	1.05E-5	NO	2.78E-7	YES
RB05	1.26E-2	2.8E-2	3.56E-4	NO	6.6E-7	YES
RB06	1.02E-2	2.9E-2	2.96E-4	NO	8.26E-6	NO
RB07	9.20E-2	5.3E-3	4.88E-4	NO	8.15E-7	YES
RB08	7.03E-3	5.5E-5	3.87E-7	YES		
RB09	3.45E-3	5.3E-4	1.83E-6	NO	6.50E-7	YES
RB10	4.33E-3	2.2E-5	9.53E-8	YES		
RHR1	3.4E-2	2.9E-2	1.06E-3	NO	6.6E-7	YES
RHR1SG					5.9E-7	YES
RHR2	3.57E-2	4.4E-3	1.63E-4	NO	7.3E-7	YES
RHR2SG					3.3E-7	YES
RW	1.25E-2	1.4E-2	1.61E-4	NO	9.57E-7	YES
TB	5.79E-2	1.0	5.68E-2	NO		
TBb					6.6E-7	YES
TB1					8.7E-7	YES
TB2					3.8E-7	YES
TB3					1.1E-8	YES
TBWOR					8.0E-7	YES
TRANS	2.51E-3	2.0E-4	5.02E-7	YES		
USRCD	9.00E-4	1.0	9.00E-4	NO	1.92E-7	YES
YARD	3.75E-3	6.5E-2	2.44E-4	NO	9.4E-7	YES

Table 4- 7 List of Appendix R Safe Shutdown Systems		
Code	System	Function
C11	CRD Hydraulic Control Units	REQUIRED FOR HOT AND COLD SHUTDOWN
B21	MSIVs (manual closure only)	
T50	Suppression Pool Lvl/Tmp Monitoring	
B21	RPV Press. and Temp. Monitoring	
T41	Control Center HVAC and ESF fan-coil units	
P44	Emergency Equipment Cooling Water (EECW)	
P45	Emergency Equipment Service Water (EESW)	
R30-01	EDGs and Auxiliaries	
X41-03	EDG and Switchgear Room HVAC	
R32	ESF DC Power	
R30,R14, R16	ESF AC Power for Shutdown Equipment	
E11-51	RHR Service Water	
E11-56	RHR Cooling Towers	
P50-02	Control Air	
E51	Reactor Core Isolation Cooling (RCIC)	REQUIRED FOR HOT SHUTDOWN ONLY
B21	Safety Relief Valves (SRVs)	
E41	High Pressure Coolant Injection (HPCI)	
E11	RHR, Containment Cooling and LPCI Modes	
E11	RHR, Shutdown Cooling Mode	REQUIRED FOR COLD SHUTDOWN ONLY
B31	Recirculation (Discharge Valves only)	

Table 4- 8 Auxiliary Building Fire Zones

ZONE	DESCRIPTION
01AB	Aux. Bldg. Basement (el. 551' 0")
02AB	Aux. Bldg. Mezzanine/Cable Tray Area - includes stairwell up to 03AB (el. 583' 6" & 603' 6")
03AB	Aux. Bldg. Relay Room (el. 613' 6")
04AB	Aux. Bldg. Switchgear Room Division I (el. 613' 8 1/2")
05ABE	Aux. Bldg. Cable Tunnel Division I (el. 613' 6")
05ABW	Aux. Bldg. Cable Tunnel Division II (el. 613' 6")
06AB	Aux. Bldg. 2nd Floor Miscellaneous Rooms (el. 613' 6")
07AB	Aux. Bldg. Cable Spreading Room (el. 630' 6")
08AB	Aux. Bldg. Cable Tray Area (El 631' 0")
09AB	Aux. Bldg. Control Complex - includes stairwell down to 07AB (el. 643" 6' & 655' 6")
10ABE	Aux. Bldg. Battery Room Division I (el. 643' 6")
10ABW	Aux. Bldg. Battery Room Division II (el. 643' 6")
11AB	Aux. Bldg. Miscellaneous Rooms (el. 643' 6")
12AB	Aux. Bldg. Switchgear Room Division II (el. 643' 6")
13AB	Aux. Bldg. Ventilation Area - includes stairwell down to el. 643' 6" (El 659' 6")
14AB	Aux. Bldg. CCHV and SGTS Rooms (el. 677' 6")
15AB	Aux. Bldg. Ventilation Equipment Area (el. 677' 6")

Table 4- 9 Sample FCIA Summary - Auxiliary Building

COMPARTMENT	EXPCOMP	PFS	COMMENT
01AB	01RBNE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	01RBSE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	02AB	N	FIVE Criteria #2. Intervening concrete floor of 02AB inspected per NPP 28.507.01
	02RBNE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door RB-8 inspected per NPP 28.507.02.
	02RBSE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	03RB	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01. "A" fire door, RB-3, inspected per NPP 28.507.02.
	04RBN	N	FIVE Criteria #2. Intervening concrete floor of 04RBN inspected per NPP 28.507.01
	04RBS	N	FIVE Criteria #2. Intervening concrete floor of 04RBS inspected per NPP 28.507.01
	TB	N	FIVE Criteria #2. Intervening concrete wall inspected per 28.507.01
02AB	05RBN	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	TB	N	FIVE Criteria #2. Intervening walls inspected per NPP 28.507.01. "A" fire door R1-13 inspected per NPP 28.507.02.
	03AB	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01. "A" fire door R2-16 inspected per NPP 28.507.02.
	06AB	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	05RBS	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	03RB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	01AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	04ABN	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	04ABS	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
03AB	06RBN	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	07AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	05RBN	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	02AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01. "A" fire door R2-16 inspected per NPP 28.507.02.
	TB	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01. "A" fire door R2-13 inspected per NPP 28.507.02.
	05ABE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	05ABW	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R2-12 inspected per NPP 28.507.02.
04ABN	06RBS	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	06AB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R2-15 inspected per NPP 28.507.01.
	08AB	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	02AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	05RBS	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	TB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	04ABS	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door inspected per NPP 28.507.02.
04ABS	04ABN	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door inspected per NPP 28.507.02.
	02AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	08AB	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
05ABE	03AB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	07AB	N	FIVE Criteria #2. Intervening concrete floor of 07AB inspected per NPP 28.507.01
	TB	N	FIVE Criteria #2. Intervening concrete wall, floor and ceiling inspected per NPP 28.507.01
	06AB	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01
	08AB	N	FIVE Criteria #2. Intervening concrete floor of 08AB inspected per NPP 28.507.01
	05ABW	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire doors R2-22, R2-23, and R2-24 inspected per NPP 28.507.02.
05ABW	03AB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R2-12 inspected per NPP 28.507.02.
	07AB	N	FIVE Criteria #2. Intervening concrete floor of 07AB inspected per NPP 28.507.01
	05ABE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire doors R2-22, R2-23, and R2-24 inspected per NPP 28.507.02.
	TB	N	FIVE Criteria #2. Intervening concrete wall, floor, and ceiling inspected per NPP 28.507.01

Table 4- 9 Sample FCIA Summary - Auxiliary Building

COMPARTMENT	EXPCOMP	PFS	COMMENT
	06AB	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01. "A" fire door R2-11 inspected per NPP 28.507.02.
	08AB	N	FIVE Criteria #2. Intervening concrete floor of 08AB inspected per NPP 28.507.01
06AB	TB	N	FIVE Criteria #2. Intervening concrete walls, floor and ceiling inspected per NPP 28.507.01. "A" fire door R2-14 inspected per NPP 28.507.02.
	02AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	05ABE	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01
	05ABW	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01. "A" fire door R2-11 inspected per NPP 28.507.02.
	06RBS	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	05RBS	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	04ABN	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R2-15 inspected per NPP 28.507.02.
07AB	06RBN	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	03AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01.
	09AB	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01. "A" fire doors RM-2 and RM-3 inspected per NPP 28.507.01.
	05ABE	N	FIVE Criteria #2. Intervening concrete floor is inspected per NPP 28.507.01
	05ABW	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	TB	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01
08AB	05ABW	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	05ABE	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	TB	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01
	10ABE	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	10ABW	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	06RBS	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	04ABN	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01.
	04ABS	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01.
	06AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	11ABE	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	11ABW	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	12AB	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01.
09AB	13AB	N	FIVE Criteria #2. Stairwell opposite wall directly connects with zone 13AB. Intervening concrete wall inspected per NPP 28.507.01.
	14AB	N	FIVE Criteria #2. Intervening concrete floor of 14AB inspected per NPP 28.507.01
	07AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01. "A" fire doors RM2-2 and RM2-3 inspected per NPP 28.507.02.
	TB	N	FIVE Criteria #3. Intervening concrete wall inspected per NPP 28.507.01. Fire door R3-13, although inspected per NPP 28.507.02, is 1.5 hrs - but, zone combustible loading is 55,000 BTU/ft^2.
	07RBN	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	10ABW	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	10ABE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
10ABE	09AB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	10ABW	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	13AB	N	FIVE Criteria #2. Intervening concrete wall and ceiling inspected per NPP 28.507.01
	11ABE	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01. "A" fire door R3-6 inspected per NPP 28.507.02.
	TB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	08AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
10ABW	09AB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	13AB	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	10ABE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	07RBN	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	07RBS	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	11ABW	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R3-21 inspected per NPP 28.507.02.
	08AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01

Table 4- 9 Sample FCIA Summary - Auxiliary Building

COMPARTMENT	EXPCOMP	PFS	COMMENT
	TB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
11ABE	10ABE	N	FIVE Criteria #2. Intervening concrete walls inspected per NPP 28.507.01. "A" fire door R3-6 inspected per NPP 28.507.02.
	13AB	N	FIVE Criteria #2. Intervening concrete wall and ceiling inspected per NPP 28.507.01
	TB	N	FIVE Criteria #2. Intervening concrete floor and walls inspected per NPP 28.507.01. "A" fire door R3-12 inspected per NPP 28.507.02.
	08AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	12AB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R3-7 inspected per NPP 28.507.02.
	11ABW	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door inspected per NPP 28.507.02.
11ABW	07RBS	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	10ABW	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R3-21 inspected per NPP 28.507.02.
	12AB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R3-11 inspected per NPP 28.507.02.
	11ABE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door inspected per NPP 28.507.02.
	13AB	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	08AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
12AB	11ABE	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R3-7 inspected per NPP 28.507.02.
	11ABW	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R3-11 inspected per NPP 28.507.02.
	TB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	07RBS	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	13AB	N	FIVE Criteria #2. Intervening concrete ceiling inspected per NPP 28.507.01
	08AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01.
13AB	09AB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	08RBN	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	08RBS	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	TB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01. "A" fire door R3-4 inspected per NPP 28.507.02.
	10ABE	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	10ABW	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	12AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	11ABE	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	11ABW	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	15AB	N	FIVE Criteria #4. Intervening concrete ceiling not inspected per NPP 28.507.01. However, zone equipped with automatic fire detection and zone combustible loading is 1190 Btu/ft^2.
14AB	15AB	N	FIVE Criteria #4. Intervening concrete wall inspected per NPP 28.507.01. Rating of door between 14AB and 15AB unidentified and door is not listed for inspection in NPP 28.507.02. However, auto. fire detection avail. and combustible loading is 7600 Btu/ft^2.
	09RBN	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	09AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	13AB	N	FIVE Criteria #2. Intervening concrete floor inspected per NPP 28.507.01
	TB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
15AB	14AB	N	FIVE Criteria #4. Intervening concrete wall inspected per NPP 28.507.01. Rating of door between 14AB and 15AB unidentified and door is not listed for inspection in NPP 28.507.02. However, auto. fire detection avail. and combustible loading is 2200 Btu/ft^2.
	TB	N	FIVE Criteria #2. Intervening concrete wall inspected per NPP 28.507.01
	13AB	N	FIVE Criteria #4. NPP 28.507.01 does not list the intervening concrete floor as a barrier to be inspected. However, zone equipped with automatic fire detection and zone combustible loading is 2200 Btu/ft^2.

Table 4- 10 FIVE Phase I Screening Summary - Auxiliary Building

NO.	FIRE AREA	COMP ID	COMPARTMENT DESCRIPTION	PHASE I SCREEN	INITIAL PHASE II SCREENING CDF	INITIAL PHASE II SCREEN	PHASE II FIRE MODELING CDF	FINAL PHASE II SCREEN
1	Auxiliary Bldg.	01AB	Aux. Bldg. Basement (el. 551')	NO	2.23E-05	NO	2.80E-07	YES
2	Auxiliary Bldg.	02AB	Aux. Bldg. Mezzanine and Cable Tray Area (el. 583' and 603' 6")	NO	8.36E-05	NO	4.37E-07	YES
3	Auxiliary Bldg.	03AB	Aux. Bldg. Relay Room (el. 613' 6")	NO	7.48E-06	NO	2.77E-06	NO
4	Auxiliary Bldg.	04ABN	Aux. Bldg. Switchgear Division I (el. 613' 8 1/2") - Major Div. I Portion	NO	1.55E-04	NO	4.31E-06	NO
5	Auxiliary Bldg.	04ABS	Aux. Bldg. Switchgear Division I (el. 613' 8 1/2") - Major Div. II Portion	NO	2.14E-06	NO	2.45E-07	YES
6	Auxiliary Bldg.	05ABE	Aux. Bldg. Cable Tunnel Division I (el. 613' 6")	NO	3.56E-05	NO	1.98E-07	YES
7	Auxiliary Bldg.	05ABW	Aux. Bldg. Cable Tunnel Division II (el. 613' 6")	NO	3.78E-05	NO	2.10E-07	YES
8	Auxiliary Bldg.	06AB	Aux. Bldg. 2nd Floor Misc. Rooms (el. 613' 6")	NO	8.38E-05	NO	6.35E-07	YES
9	Auxiliary Bldg.	07AB	Aux. Bldg. Cable Spreading Room (630' 6")	NO	> 1E-6	NO	1.05E-07	YES
10	Auxiliary Bldg.	08AB	Aux. Bldg. Cable Tray Area (631' 0")	NO	> 1E-6	NO	1.05E-07	YES
11	Auxiliary Bldg.	09AB	Aux. Bldg. Control Complex (el. 643' 6" and 655' 6")	NO	> 1E-6	NO	4.27E-06	NO
12	Auxiliary Bldg.	10ABE	Aux. Bldg. Battery Room Division I (el. 643' 6")	NO	8.68E-07	YES	---	YES
13	Auxiliary Bldg.	10ABW	Aux. Bldg. Battery Room Division II (el. 643' 6")	NO	7.16E-07	YES	---	YES
14	Auxiliary Bldg.	11ABE	Aux. Bldg. Misc Rooms (el. 643' 6") - Major Div. I Portion	NO	7.80E-05	NO	1.90E-06	NO
15	Auxiliary Bldg.	11ABW	Aux. Bldg. Misc Rooms (el. 643' 6") - Major Div. II Portion	NO	4.46E-06	NO	5.09E-07	YES
16	Auxiliary Bldg.	12AB	Aux. Bldg. Switchgear Room Division II (el. 643' 6")	NO	1.64E-04	NO	2.54E-06	NO
17	Auxiliary Bldg.	13AB	Aux. Bldg. Ventilation Area (el. 659' 6")	NO	2.97E-05	NO	7.55E-07	YES
18	Auxiliary Bldg.	14AB	Aux. Bldg. CCHV and SGTS Rooms (el. 677' 6")	NO	1.12E-06	NO	8.43E-07	YES
19	Auxiliary Bldg.	15AB	Aux. Bldg. Ventilation Equipment Area (el. 677' 6")	NO	4.92E-08	YES	---	YES

Table 4- 11 Auxiliary Building Fire Ignition Frequency Estimates

Fire Compartment	Fire Ignition Frequency (Per Yr)
01AB	3.23E-03
02AB	1.23E-03
03AB	1.32E-02
04ABN	4.57E-03
04ABS	1.05E-03
05ABE	1.08E-03
05ABW	1.08E-03
06AB	1.33E-03
07AB	1.08E-03
08AB	1.05E-03
09AB	1.09E-02
10ABE	2.17E-03
10ABW	2.17E-03
11ABE	1.20E-02
11ABW	3.43E-03
12AB	4.44E-03
13AB	3.62E-03
14AB	5.62E-03
15AB	1.49E-03

Table 4- 12 Initial Phase II Results - Auxiliary Building

COMPARTMENT	RUN DESCRIPTION	RUN ID (1)	Fi	CCDP	CDF	COMMENTS
01AB	Basement Div. I	1AB01	3.23E-03	3.00E-03	9.69E-06	
	Basement Div. II	1AB02	3.23E-03	3.90E-03	1.26E-05	
	Total of 1AB Fire Scenarios:	---	---	---	**2.23E-05**	
02AB	Mezzanine Div. I	2AB01	1.23E-03	3.50E-02	4.31E-05	
	Mezzanine Div. II	2AB02	1.23E-03	3.30E-02	4.06E-05	
	Total of 2AB Fire Scenarios:	---	---	---	**8.36E-05**	
03AB	Panel -P617	3AB01	4.47E-05	2.40E-05	1.07E-09	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P621	3AB02	4.47E-05	4.50E-05	2.01E-09	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P626	3AB03	4.47E-05	9.00E-06	4.03E-10	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P628	3AB04	4.47E-05	8.60E-06	3.85E-10	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P898A	3AB05	4.47E-05	9.10E-06	4.07E-10	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P618	3AB06	4.47E-05	3.60E-05	1.61E-09	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P627	3AB07	4.47E-05	9.00E-06	4.03E-10	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P898B	3AB08	4.47E-05	9.40E-06	4.21E-10	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P613	3AB10	4.47E-05	4.40E-03	1.97E-07	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P622	3AB11	4.47E-05	3.00E-03	1.34E-07	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P857	3AB12	4.47E-05	2.40E-04	1.07E-08	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P822	3AB13	4.47E-05	4.40E-03	1.97E-07	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P823	3AB14	4.47E-05	3.00E-03	1.34E-07	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P868	3AB15	4.47E-05	4.40E-03	1.97E-07	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P889	3AB16	4.47E-05	1.30E-04	5.82E-09	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P891	3AB17	4.47E-05	3.10E-03	1.39E-07	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P855	3AB19	4.47E-05	3.50E-02	1.57E-06	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P853	3AB20	4.47E-05	1.50E-04	6.71E-09	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P854	3AB21	4.47E-05	3.60E-02	1.61E-06	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P623	3AB23	4.47E-05	3.60E-02	1.61E-06	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P870	3AB24	4.47E-05	4.70E-05	2.10E-09	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P620	3AB25	4.47E-05	2.90E-05	1.30E-09	Impacts based on review of DECO To-From database. Results may be conservative.
	Panel -P609	BASE	4.47E-05	1.30E-05	5.82E-10	Panel contains RPS, assume MC=1.0. Use BASE run.
	Panel -P610	BASE	4.47E-05	1.30E-05	5.82E-10	Panel contains RPS, assume MC=1.0. Use BASE run.
	Panel -P611	BASE	4.47E-05	1.30E-05	5.82E-10	Panel contains RPS, assume MC=1.0. Use BASE run.
	Panel -P820	4AB01	4.47E-05	2.10E-03	9.40E-08	RCIC and Div. II CS, EECW, RHR disabled. Conservatively use Run 4AB01.
	Panel -P856	3AB26	4.47E-05	4.60E-04	2.06E-08	MC, SA, SBFW, RBCCW, TBCCW, and NIAS I disabled.
	Panel -P862	RB04D1	4.47E-05	4.20E-03	1.88E-07	Conservatively assume all Div. II equipment disabled and use Run RB04D1.
	Panel -P869	RB04D1	4.47E-05	4.20E-03	1.88E-07	Conservatively assume all Div. I equipment disabled and use Run RB04D1.
	Panel -P888	3AB15	4.47E-05	4.40E-03	1.97E-07	Vent and Div. I buses, RHR, NIAS disabled. Conservatively use Run 3AB15.
	Panel -P900	9AB04	4.47E-05	4.30E-03	1.92E-07	Assume disables buses 64B and C. Use run 9AB04.
	Panel -P901	9AB04	4.47E-05	4.30E-03	1.92E-07	Assume disables buses 64B and C. Use run 9AB04.
	Panel -P902	9AB05	4.47E-05	2.90E-03	1.30E-07	Assume disables buses 65E and F. Use run 9AB05.

Table 4- 12 Initial Phase II Results - Auxiliary Building

COMPARTMENT	RUN DESCRIPTION	RUN ID (1)	Fi	CCDP	CDF	COMMENTS
	Panel -P903	9AB05	4.47E-05	2.90E-03	1.30E-07	Assume disables buses 65E and F. Use run 9AB05.
	Panel -P923	3AB02	4.47E-05	4.50E-05	2.01E-09	RCIC rack disabled. Use Run 3AB02.
	Panel -P929	3AB25	4.47E-05	2.90E-05	1.30E-09	HPCI rack disabled. Use Run 3AB25.
	Panel -P821	15AB3	4.47E-05	3.30E-05	1.48E-09	Panel fire results in manual shutdown and loss of MC (w/ 0.1 HEP for SBFW)
	Panel -P837	15AB2	4.47E-05	3.40E-05	1.52E-09	Panel fire results in manual shutdown (w/ 0.1 HEP for SBFW)
	Panel -P838	15AB2	4.47E-05	3.40E-05	1.52E-09	Panel fire results in manual shutdown (w/ 0.1 HEP for SBFW)
	Panel -P839	15AB2	4.47E-05	3.40E-05	1.52E-09	Panel fire results in manual shutdown (w/ 0.1 HEP for SBFW)
	Panel -P840	15AB2	4.47E-05	3.40E-05	1.52E-09	Panel fire results in manual shutdown (w/ 0.1 HEP for SBFW)
	Panel -P861	15AB3	4.47E-05	3.30E-05	1.48E-09	Panel fire results in manual shutdown and loss of MC (w/ 0.1 HEP for SBFW)
	Panel -P866	15AB3	4.47E-05	3.30E-05	1.48E-09	Panel fire results in manual shutdown and loss of MC (w/ 0.1 HEP for SBFW)
	Panel -P877	15AB3	4.47E-05	3.30E-05	1.48E-09	Panel fire results in manual shutdown and loss of MC (w/ 0.1 HEP for SBFW)
	Panel -P878	15AB3	4.47E-05	3.30E-05	1.48E-09	Panel fire results in manual shutdown and loss of MC (w/ 0.1 HEP for SBFW)
	Panel -P879	15AB3	4.47E-05	3.30E-05	1.48E-09	Panel fire results in manual shutdown and loss of MC (w/ 0.1 HEP for SBFW)
	Panel -P880	15AB3	4.47E-05	3.30E-05	1.48E-09	Panel fire results in manual shutdown and loss of MC (w/ 0.1 HEP for SBFW)
	Cabinet 2PA2-5 (all 17 "buckets")	BASE	7.61E-04	1.30E-05	9.89E-09	Impacts based on review of PSA fault trees. Results in loss of MC.
	Cabinet 2PA2-6 (all 17 "buckets")	3AB03	7.61E-04	9.00E-06	6.85E-09	Impacts based on review of PSA fault trees. Results in loss of Div. I LPCS.
	Cabinet 2PB2-5 (all 17 "buckets")	3AB07	7.61E-04	9.00E-06	6.85E-09	Impacts based on review of PSA fault trees. Results in loss of Div. II LPCS.
	Cabinet 2PB2-6 (all 17 "buckets")	BASE	7.61E-04	1.30E-05	9.89E-09	Impacts based on review of PSA fault trees. Results in loss of MC.
	96 Misc. Panels - MC available	15AB2	4.30E-03	3.40E-05	1.46E-07	Each panel fire results in manual shutdown (w/ 0.1 HEP for SBFW)
	84 Misc. Panels - MC unavailable	15AB3	3.76E-03	3.30E-05	1.24E-07	Each panel fire results in manual shutdown and loss of MC (w/ 0.1 HEP for SBFW)
	Total of 3AB Fire Scenarios:	---	---	---	**7.48E-06**	
04ABN	Major Division I Portion of 04AB	4AB04	4.57E-03	3.40E-02	**1.55E-04**	
04ABS	Division II Cable Chase of 04AB	4AB03	1.05E-03	2.04E-03	**2.14E-06**	
05ABE	East Cable Tunnel - Division I	5AB03	1.08E-03	3.30E-02	**3.56E-05**	
05ABW	West Cable Tunnel - Division II	5AB04	1.08E-03	3.50E-02	**3.78E-05**	
06AB	2nd Floor Miscellaneous Rooms	6AB02	1.33E-03	6.30E-02	**8.38E-05**	
07AB	Cable Spreading Room	N/A	1.08E-03	N/A	**> 1E-6**	Initial screening runs not performed.
08AB	Cable Tray Area	N/A	1.05E-03	N/A	**> 1E-6**	Initial screening runs not performed.
09AB	Control Room Complex	N/A	1.09E-02	N/A	**> 1E-5**	Initial screening runs not performed.
10ABE	East Battery Room - Division I	10AB1	2.17E-03	4.00E-04	**8.68E-07**	

Table 4- 12 Initial Phase II Results - Auxiliary Building

COMPARTMENT	RUN DESCRIPTION	RUN ID (1)	Fi	CCDP	CDF	COMMENTS
10ABW	West Battery Room - Division II	10AB2	2.17E-03	3.30E-04	**7.16E-07**	
11ABE	Misc. Rooms - Majority of Area (Div. I)	11AB1	1.20E-02	6.50E-03	**7.80E-05**	
11ABW	Misc. Rooms - Div. II Battery Chargers	11AB3	3.43E-03	1.30E-03	**4.46E-06**	
12AB	Division II SWGR Room	12AB2	4.44E-03	3.70E-02	**1.64E-04**	
13AB	Ventilation Equip. Area - Division I	13AB1	3.62E-03	3.00E-03	1.09E-05	
	Ventilation Equip. Area - Division II	13AB2	3.62E-03	5.20E-03	1.88E-05	
	Total of 13AB Fire Scenarios:	---	---	---	**2.97E-05**	
14AB	CCHV Equip./SGTS Area	14AB2	5.62E-03	2.00E-04	**1.12E-06**	
15AB	Ventilation Equipment Area	15AB3	1.49E-03	3.30E-05	**4.92E-08**	

Table 4- 13 Final Phase II Results - Auxiliary Building

NO.	FIRE AREA	COMP ID	COMPARTMENT DESCRIPTION	PHASE I SCREEN	INITIAL PHASE II SCREENING CDF	INITIAL PHASE II SCREEN	PHASE II FIRE MODELING CDF	FINAL PHASE II SCREEN
1	Auxiliary Bldg.	01AB	Aux. Bldg. Basement (el. 551')	NO	2.23E-05	NO	2.80E-07	YES
2	Auxiliary Bldg.	02AB	Aux. Bldg. Mezzanine and Cable Tray Area (el. 583' and 603' 6")	NO	8.36E-05	NO	4.37E-07	YES
3	Auxiliary Bldg.	03AB	Aux. Bldg. Relay Room (el. 613' 6")	NO	7.48E-06	NO	2.77E-06	NO
4	Auxiliary Bldg.	04ABN	Aux. Bldg. Switchgear Division I (el. 613' 8 1/2") - Major Div. I Portion	NO	1.55E-04	NO	4.51E-06	NO
5	Auxiliary Bldg.	04ABS	Aux. Bldg. Switchgear Division I (el. 613' 8 1/2") - Major Div. II Portion	NO	2.14E-06	NO	2.45E-07	YES
6	Auxiliary Bldg.	05ABE	Aux. Bldg. Cable Tunnel Division I (el. 613' 6")	NO	3.56E-05	NO	1.98E-07	YES
7	Auxiliary Bldg.	05ABW	Aux. Bldg. Cable Tunnel Division II (el. 613' 6")	NO	3.78E-05	NO	2.10E-07	YES
8	Auxiliary Bldg.	06AB	Aux. Bldg. 2nd Floor Misc. Rooms (el. 613' 6")	NO	8.38E-05	NO	6.35E-07	YES
9	Auxiliary Bldg.	07AB	Aux. Bldg. Cable Spreading Room (630' 6")	NO	> 1E-6	NO	1.05E-07	YES
10	Auxiliary Bldg.	08AB	Aux. Bldg. Cable Tray Area (631' 0")	NO	> 1E-6	NO	1.05E-07	YES
11	Auxiliary Bldg.	09AB	Aux. Bldg. Control Complex (el. 643' 6" and 655' 6")	NO	> 1E-6	NO	4.27E-06	NO
12	Auxiliary Bldg.	10ABE	Aux. Bldg. Battery Room Division I (el. 643' 6")	NO	8.68E-07	YES	---	YES
13	Auxiliary Bldg.	10ABW	Aux. Bldg. Battery Room Division II (el. 643' 6")	NO	7.16E-07	YES	---	YES
14	Auxiliary Bldg.	11ABE	Aux. Bldg. Misc Rooms (el. 643' 6") - Major Div. I Portion	NO	7.80E-05	NO	1.90E-06	NO
15	Auxiliary Bldg.	11ABW	Aux. Bldg. Misc Rooms (el. 643' 6") - Major Div. II Portion	NO	4.46E-06	NO	5.09E-07	YES
16	Auxiliary Bldg.	12AB	Aux. Bldg. Switchgear Room Division II (el. 643' 6")	NO	1.64E-04	NO	2.54E-06	NO
17	Auxiliary Bldg.	13AB	Aux. Bldg. Ventilation Area (el. 659' 6")	NO	2.97E-05	NO	7.55E-07	YES
18	Auxiliary Bldg.	14AB	Aux. Bldg. CCHV and SGTS Rooms (el. 677' 6")	NO	1.12E-06	NO	8.43E-07	YES
19	Auxiliary Bldg.	15AB	Aux. Bldg. Ventilation Equipment Area (el. 677' 6")	NO	4.92E-08	YES	---	YES

Table 4- 14 UNSCREENED COMPARTMENTS		
FIRE AREA	**FIRE COMPARTMENT**	**CDF (F_2) (F1 x P2)**
Reactor Building 2nd Floor	RB06	1.00E-6
Relay Room	03AB	2.77E-6
Div. 1 Switchgear	04ABN	4.51E-6
Control Room	09AB	4.27E-6
Div. 1 portion Miscellaneous Room	11ABE	1. 9E-6
Div. 2 Switchgear	12AB	2.54E-6

Table 4- 15 Example Ignition Source Data Sheet (ISDS)

Fire Compartment Ignition Source Data Sheet (ISDS)

Compartment Description

Fire Area			
Fire Compartment		15AB	

Compartment Fire Ignition Frequency

Stop 1.1	Plant Location ID	RB	
	Plant Location Description	Reactor Building (BAR)	
Step 1.2 (WFL)	Location Weighting Factor	1.00E+00	Plant Location
(WFL)	Location Weighting Factor	1.00E+00	Plant Wide Components

Compartment Ignition Sources (FIF)

Fire Ignition/Fuel Source	Sources in Compartment (A)	Sources in Plant Location (B)	Weighting Factor WFLS = (A)/(B)	Fire Frequency (FF)	Ignition Source Frequency (FIF)
1. Electrical Cabinets	3	1.1 3E + 03	2.66E-03	5.0E-02	1.3E-0[illegible]
2. Pumps	0	5.40E + 0 1	0.00E+00	2.5E-02	0.00E + 00
3.	0	0.00E+00	0.00E + 00	0.00E+00	0.00E+00
4.	0	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5.	0	0.00E+00	0.00E + 00	0.00E+00	0.00E + 00
6.	0	0.00E+00	0.00E + 00	0.00E+00	0.00E + 00
7.	0	0.00E+00	0.00E + 00	0.00E+00	0.0E+00
Plant Wide Ignition Sources	(A)	(C)	(A)/(C)		*Reference 1. 2*
Fire Protection Panels	1	1.97E+02	5.08E-03	2.4E-03	1.2E-05
RPS MG Sets	0	2.00E+00	0.00E + 00	5.5E-03	0.00E + 00
Non-qualified Cable Run	0	0.00E+00	0.00E+00	6.3E-03	0.00E+00
Junction box/Splice in Non-qualified Cable	0	0.00E+00	0.00E + 00	1.6E-03	0.00E + 00
Junction Box in Qualified Cable	0	0.00E + 00	0.00E + 00	1.6E-03	0.00E + 00
Transformers	1	8.10E+01	1.23E-02	7.9E-03	9.8E-05
Battery Chargers	0	1.70E+01	0.00E + 00	4.0E-03	0.00E+00
Off-gas/Hydrogen Recombiner (BWR)	0	2.00E+00	0.00E + 00	8.6E-02	0.00E + 00
Hydrogen Tanks	0	4.00E+00	0.00E + 00	3.2E-03	0.00E+00
Miscellaneous Hydrogen Fires (Y/N)	N	4.30E+01	0.00E + 00	3.2E-03	0.00E + 00
Gas Turbines	0	4.00E+00	0.00E + 00	3.1 E-02	0.00E +00
Air Compressors	0	9.00E+00	0.00E + 00	4.7E-03	0.00E + 00
Ventilation Subsystems	7	3.33E+02	2.1 0E-02	9.5E-03	2.0E-04
Elevator Motors	0	4.00E+00	0.00E+00	6.3E-03	0.00E + 00
Dryers	0	8.00E+00	0.00E+00	8,7E-03	0.00E + 00
Cable Fires Caused by Welding	N/A	4.30E+01	2.33E-02	5.1 E-03	1.2E-04
Transient Fires Caused by Welding/Cutting	N/A	4.30E+01	2.33E-02	3.1 E-02	7.2 E-04
Transients;	(Y/N)	7		1.3E-03	2.1 E-04
Cigarette Smoking	N	0			
Extension Cords	Y	4			
Heater	Y	3			
Candle	N	0			
Overheating	N	0			
Hot Pipe	N	0			

Compartment Fire Frequency (FL)

Step 1.4 Compartment Fire Frequency	**1.49E-03**
Fire Compartment CSDS Required	**Yes**

Notes:

(A) Number of Ignition Sources in Compartment
(B) Total Number of Ignition Sources in Selected Plant Location
(C) Total Number of Ignition Sources/Compartments in Plant

*Ignition Source Frequency: FIF = WFL*WFLS*FF*
Compartment Fire Frequency: FL = SUM(FIF)

Figure 4- 1 Fire Induced Vulnerability Evaluation Overview

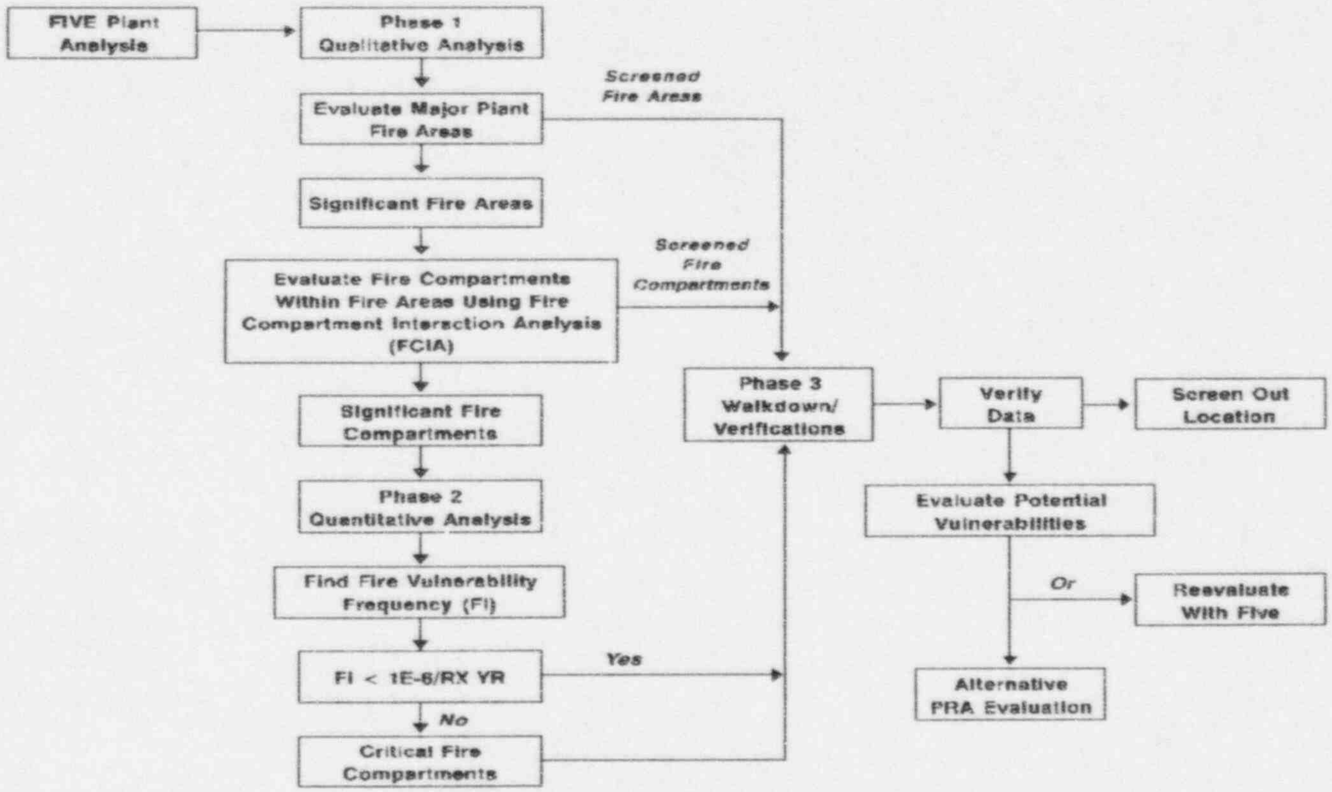
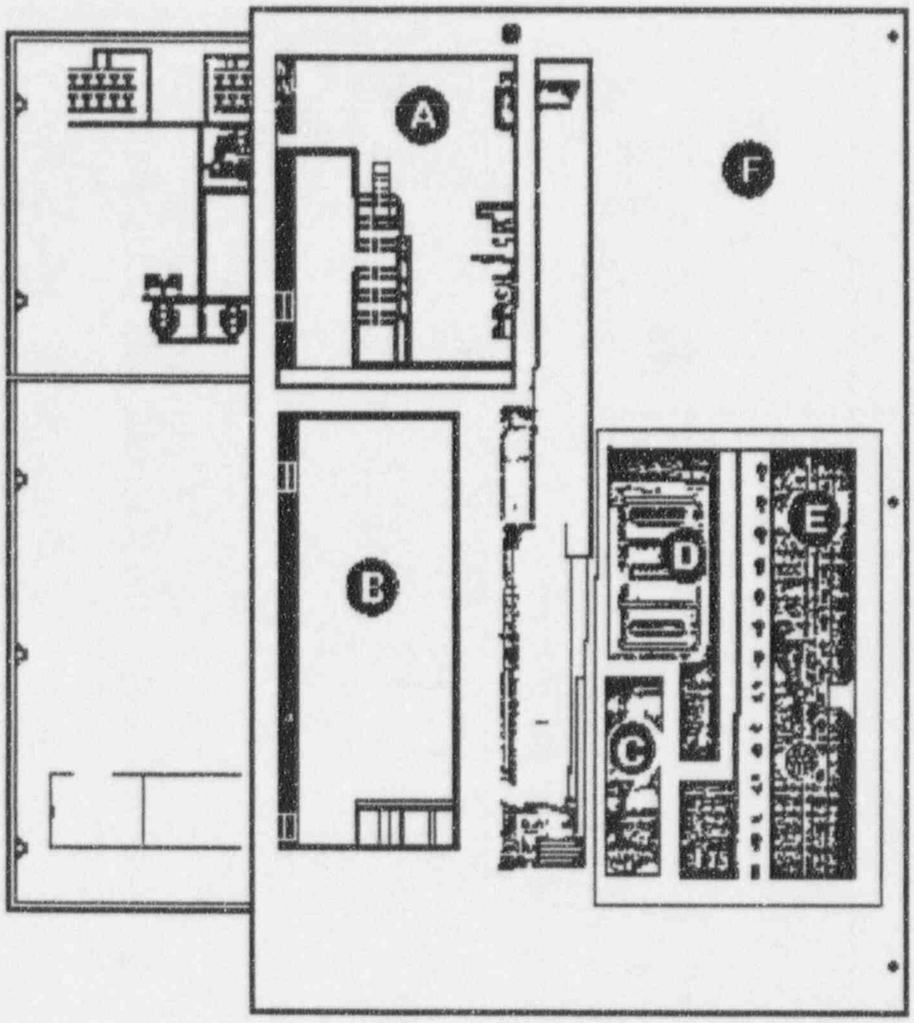


Figure 4- 2 Representative Construction Design of Fermi 2 Switchgear Bus Units (End View)



A. Instrument Compartment

B. Circuit Breaker Compartment

C. Current Transformer Compartment

D. Bus Compartment

E. Cable Compartment

F. Auxiliary Device Compartment

Figure 4- 3 General Depiction of Fermi 2 Switchgear Bus and Adjoining Transformer Cabinet

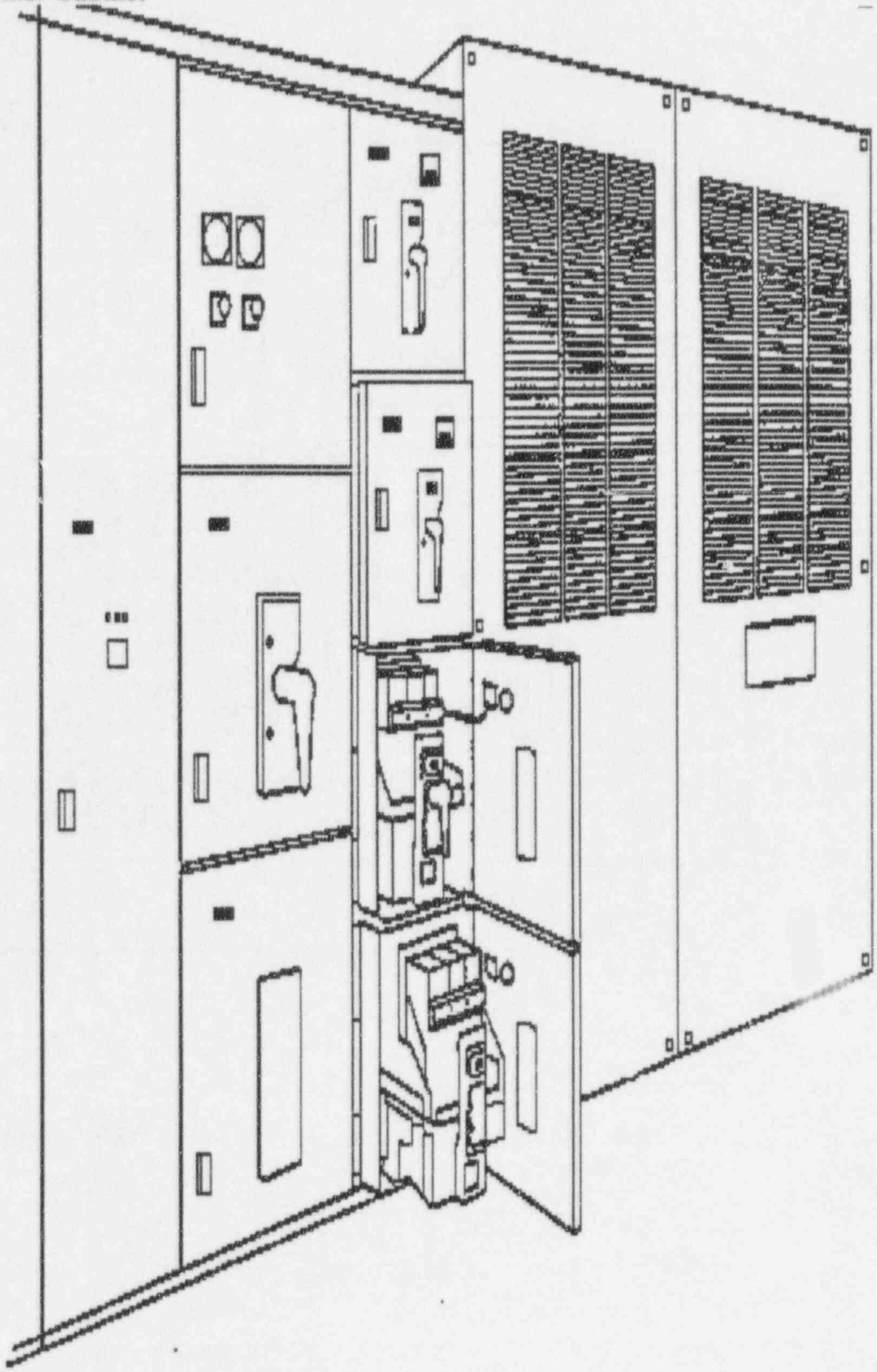
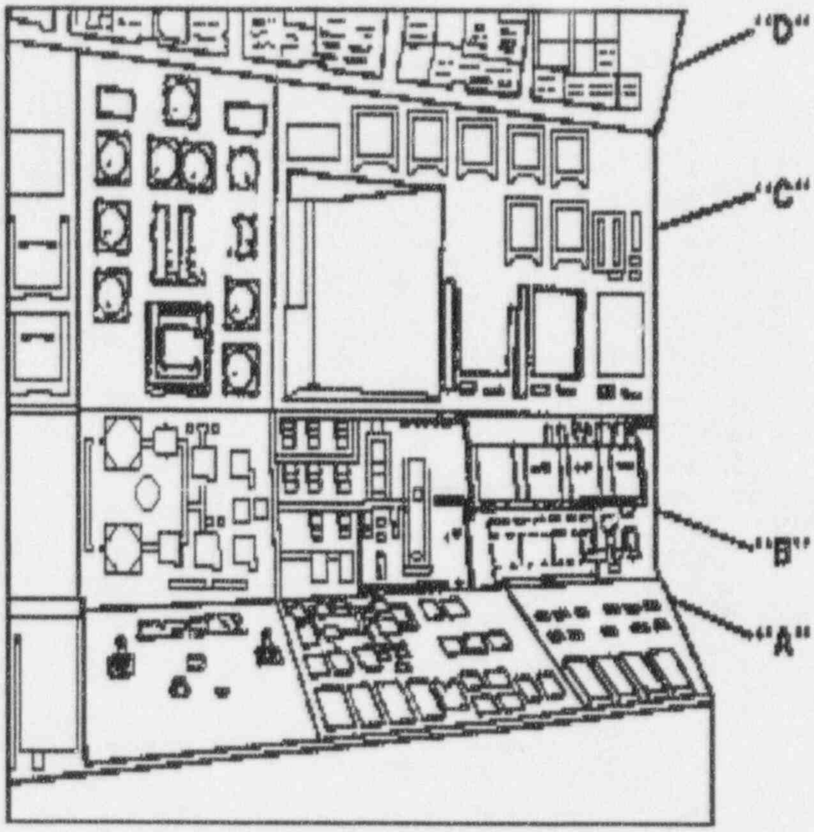


Figure 4- 4 General Layout of Control Room Main Control Board



SURFACE	LOCATED ON SURFACE
A	System Mimic Displays System Process Instrumentation System Component Controls • CMC Switches • Pushbuttons • Multi-Position Switches • Interlock Defeat Switches • Auto/Manual Controllers
B	
C	**Process Instrumentation** **Monitoring Instrumentation** • **Recorders** • **Indicators** • **Meters** • **Digital Displays** • **Core Mimic Display**
D	**Annunciator Windows** **Window Boxes**

SECTION 5

HIGH WINDS, FLOODS AND OTHER EVENTS ANALYSIS

As part of the Individual Plant Examination of External Events (IPEEE), Detroit Edison completed an evaluation to identify potential vulnerabilities at Fermi 2 due to high winds, floods and nearby facility accidents. This evaluation used the progressive screening methodology outlined in NRC NUREG-1407 **[5.1]**. The key elements of this methodology are depicted in Figure 5-1.

This methodology is implemented in essentially three steps:

- The first step requires review of the current event hazard for the site and comparison with the licensing basis of the plant. To do this, data was collected from external sources to adequately characterize the event hazard for Fermi 2. This was then compared to the Fermi 2 site design basis as documented in the Updated Final Safety Analysis Report (UFSAR) **[5.2]** and related design basis documents.

- Through this review and comparison, specific areas which showed significant change since the issuance of the Fermi 2 license were identified.

- Following the steps outlined above, the Fermi 2 design was reviewed to determine if it conformed to the requirements of the 1975 Standard Review Plan (SRP) **[5.3]**. Particular emphasis was provided in this review to those areas described above which exhibited significant change in the characterization of the event hazard since the issuance of the Fermi 2 operating license.

The sections below provide the results of the evaluation of Fermi 2 for effects of high winds, floods, and transportation and nearby facility accidents. This summary addresses the most significant aspects of the Fermi 2 design in comparison to SRP requirements. A more comprehensive assessment can be found in the "tier 2" report, "Individual Plant Examination of External Events (IPEEE) for High Winds, Floods, and Transportation and Nearby Facility accidents for Enrico Fermi Unit 2" **[5.4]**. Each section below contains three key elements required to summarize the results of the evaluation:

- Information on the development of the event hazard for Fermi 2 including the plant licensing basis and any changes since the plant was licensed;
- Identification of significant changes in the event hazard since Fermi 2 was licensed. This includes only those changes not previously reported per 10CFR50.71(e); and

- Summary of the review to determine the robustness of the Fermi 2 design relative to the NRC's 1975 Standard Review Plan (SRP).

All elements of the Fermi 2 design, relative to high winds, floods, and transportation and nearby facility accidents were evaluated and found conforming to the 1975 SRP criteria.

NUREG-1407 requires that the licensee assure that no other plant-unique external event, other than high winds, floods, and transportation and nearby facility accidents, poses a significant threat of severe accident to Fermi 2. To this end, those external events as defined in NRC NUREG/CR-2300 **[5.5]** were screened to identify any other events which could be potentially important accident initiators and may present severe core damage threat. The methodology and results of this screening are provided in Section 5.4.

5.1 HIGH WINDS

High or extreme winds present a potential threat to a nuclear power plant. The components of the general atmospheric circulation which give rise to extreme winds are extra tropical cyclones, tropical cyclones, and tornadoes. The intensity and occurrence frequency of winds which are generated by these components are a function of the climatic conditions of the geographic area in which the plant is situated.

Winds have a number of effects on structures within their path. They can apply effective external pressures to structures, they can create external/internal pressure differentials in closed structures, and they can generate missiles which are carried with potentially damaging kinetic energies. The winds associated with tornadoes are typically the most intense and highest in magnitude. Tornadoes can eject large damaging missiles with high kinetic energies. Tornadoes typically provide the controlling wind related loads which must be considered in the wind resistant design of nuclear power plant structures and the protection of components required to safely operate the plant.

The following sections provide a discussion of the climatic conditions in the area of the Fermi 2 site, the extreme wind storms which are associated with the area, the licensed extreme wind design basis for Fermi 2, and a comparison of the Fermi 2 licensed design basis with the requirements of the 1975 SRP.

5.1.1 Plant Design Basis

This section provides a comparison of the wind resistant design of Fermi 2 to the requirements of the SRP. Design and construction of Fermi 2 began several years prior to the NRC's issuance of the SRP; a construction permit was issued for Fermi 2 in September 1972 and the plant began commercial operation in 1988. Since the Fermi 2 construction permit preceded the issuance of the SRP, the plant design was not committed to the SRP. However, because of the extended term over which the plant was designed and constructed, many of the plant design features were compared to the SRP requirements prior to receipt of the Operating License. In some cases, the plant design was upgraded to comply with later regulatory requirements.

The comparison focuses on the following three principal elements of the SRP criteria for the extreme wind design of nuclear power plants:

- Definition of climatic conditions, average and extreme, which may affect the plant site. This includes determination of the 100-year return period "fastest mile of wind" and the design basis tornado characteristics (SRP Section 2.3.1);
- Evaluation of high wind loading (SRP Section 3.3.1); and
- Evaluation of tornadic wind loading including potential tornado-generated missiles (SRP Sections 3.3.2 and 3.5.1.4).

The following sections provide the details of the comparison of the Fermi 2 design with the SRP criteria for these elements.

5.1.1.1 Regional Climatology

The Fermi 2 site is located on the western shore of Lake Erie in the southeast lower climatic district of Michigan. The lake has a significant effect on the climatic extremes which are typical of the midwest. This effect is most pronounced during the cold winter months which are characterized by generally overcast skies. The areas along the lake tend to experience temperatures which are moderated relative to the extremes away from the lake. The climate is best characterized as alternating between semi-marine and continental **[5.6 and 5.7]**. Prevailing winds during the winter months are from the west. Periods of easterly winds moving across Lake Erie tend to temper the climate during the summer months.

The region around the Fermi 2 site sometimes experiences severe weather, generally associated with extratropical cyclonic air masses. The severe weather often takes the form of thunderstorms accompanied by high winds, particularly in the late spring and early summer.

Tornadoes sometimes accompany the storms in the region. The National Severe Storms Forecast Center (NSSFC) has compiled a database of tornado records over the past 50 years in an effort to characterize this intense storm and provide a basis for future risk assessments. This data was summarized by Ramsdell et.al. in NUREG/CR-4461 **[5.8]**. During the period from 1954 to 1983, 168 tornadoes were reported in a 13,853 square mile area around the Fermi 2 site. This is an occurrence rate of 4.0E-04 tornadoes per square mile per year around the Fermi 2 site.

Due to its inland location, Fermi 2 does not experience weather associated with tropical cyclonic air masses. As a result the area experiences little residual effect associated with hurricanes. The extreme wind design considerations which are appropriate for the Fermi 2 site are winds associated with severe weather, usually thunderstorms and tornadic winds.

Design Wind Velocity

The basis for the design wind velocity provided in the SRP is ANSI A58.1 "Minimum Design Loads for Buildings and Other Structures" **[5.9]**. This standard provides the extreme fastest mile wind speed for a 100-year return period. The variation in wind velocity with height above ground and factors to account for the fluctuating nature of wind velocities (gusts) are also provided.

Using the basic wind speed map provided in ANSI A58.1 for 50-year recurrence interval and a factor to conservatively adjust the wind speed to estimate the 100-year recurrence interval wind speed provides a fastest mile-of-wind at a height of 10 meters (33 feet) above the ground for the Fermi 2 site of approximately 80 miles per hour (mph).

Fermi 2 structures and components were designed using guidance provided in ASCE Paper 3269, "Wind Forces on Structures" **[5.10]**. This report provided reasonable and accepted data and methodology related to wind resistant design. The ASCE paper contains maps utilizing contours to show the fastest mile of wind over the contiguous United States. From this paper, a 100-year recurrence interval wind speed of 90 miles per hour was selected for the Fermi 2 design. The Fermi 2 design wind speed exceeds the velocity recommended by ANSI A58.1 and the fastest mile of wind recorded in the Detroit and the Toledo areas (72 mph with peak gust of 75 mph as reported by the National Climatic Data Center in the 1993 data summary for Toledo). Therefore, the Fermi 2 basic wind speed design velocity meets the SRP intent.

Tornado

In the 1975 SRP, the NRC provided specific criteria for establishing the parameters for the tornado design of nuclear power plants. These criteria included the requirements for tornado design provided in Regulatory Guides 1.117 **[5.11]** and 1.76 **[5.12]**. Additionally, the SRP provided guidance on the spectrum of potential missiles to be considered in the tornado design of the plant.

In Regulatory Guide 1.76, the NRC adopted the regionalization scheme proposed by Markee, et.al. in "Technical Basis for Interim Regional Tornado Criteria" **[5.13]**. Using this scheme, the Fermi 2 site falls into Tornado Intensity Region I. For each of the tornado intensity regions, Markee developed a definition for design basis tornado in terms of six parameters:

- Maximum wind speed;
- Maximum rotational wind speed;
- Tornado translational speed (maximum and minimum);
- Radius of maximum rotational speed;
- Pressure drop; and
- Rate of pressure drop.

The tornado resistant design of Fermi 2 was completed prior to the introduction of the aforementioned regionalization and the issuance of Regulatory Guide 1.76. The parameters of the design basis tornado were based on the state of tornadic wind knowledge at the time. A comparison of design basis tornado characteristics provided in Reg. Guide 1.76 and the design basis of Fermi 2 is provided in Table 5-1.

From Table 5-1, it is noted that there are differences in the definition of the design basis tornado used for Fermi 2 and that specified in the SRP. Specifically, Regulatory Guide 1.76 recommends 290 miles per hour tangential and 70 miles per hour translational wind velocities and a three pound per square inch pressure drop in one and one-half seconds for sites located in Tornado Intensity Region I.

Though there are differences, the Fermi 2 design meets the intent of the SRP. The design uses the same maximum tornadic wind velocity, 360 miles per hour, that the Regulatory Guide recommends. Unvented structures are designed for the tornadic wind pressure and the full three pound per square inch pressure drop. The venting of the reactor/auxiliary building fifth floor and the steam tunnel is accomplished via blow-away siding and blow-out panels; therefore, the design is independent of the rate of pressure drop. Therefore, the Fermi 2 tornado model meets the SRP intent.

5.1.1.2 Wind Loadings

SRP Section 3.3.1 provides criteria utilized to transform the design wind velocity, including variation with height and gust factors, into an effective pressure applied to structures and components. This transformation should account for the physical and geometric characteristics of the structure or component. The SRP criteria is consistent with the requirements provided in ANSI A58.1.

The wind resistant design of Fermi 2 was developed using guidance provided in ASCE Paper 3269. The ASCE paper provides a similar formulation for calculating effective pressures applied to structures and components.

Table 5-2 provides a comparison of the average design wind pressures using the methodology implemented by Fermi 2 and the criteria adopted in the SRP for design wind speeds of 80 and 90 miles per hour. See the discussion above for Fermi 2's design wind velocity.

From Table 5-2 it is evident that the methodology used by Fermi 2 to transform wind into effective external building pressure is conservative relative to the SRP criteria. Therefore, the Fermi 2 basic wind loading design meets the SRP intent.

5.1.1.3 Tornado Loadings

There are three primary elements contained in the SRP Section 3.3.2 criteria for tornado loadings and tornado resistant design:

- Definition of the characteristics of the design basis tornado;
- Methodology used for transformation of the design basis parameters into loads on structures; and

- Evaluation/review of structures and components not designed for tornado loads to assure that their collapse or failure will not degrade the function of safety-related structures or components.

The first element, design basis tornado characteristics, was discussed previously.

In Section 3.3.2 the SRP provides the requirements for developing and combining the three basic components of tornado loading: (1) effective pressures due to wind velocity, (2) differential pressures between the interior and exterior of the structure, and (3) impact forces resulting from tornado missiles. Additionally, the SRP provides requirements for the combination of these components.

Effective Wind Pressures

The criteria specified in the SRP for transforming the tornado wind velocity into an effective pressure applied to structures and components is consistent with the guidelines provided in ASCE Report 3269 or ANSI A58.1.

The Fermi 2 designs utilized criteria consistent with the SRP for the development of effective pressures due to tornado winds. The design transformed the maximum tornado wind velocity into effective pressures on structures and components using the methodology provided in ASCE 3269. The tornado wind velocity was considered constant with height and the gust factor was taken as unity. Pressure coefficients (shape factors) to account for the geometry of the structure and the pressure variation on the windward and leeward faces of the structure were also developed from the ASCE guidance.

Differential Pressures

The Fermi 2 Category I structures, with the exception of the reactor/auxiliary building above the fifth floor and the steam tunnel, are designed to resist the effects of the full three pound per square inch pressure differential. The differential is the result of a pressure drop due to wind vortexing at the core of the tornado. In unvented structures such as the reactor/auxiliary and RHR buildings, the pressure differential acts as an effective uniform pressure on the building interior tending to force the walls outward. This effect tends to act against effective tornadic wind pressure on the exterior windward side of the building and is additive to wind pressure on the leeward side of the building.

The reactor/auxiliary building above the fifth floor and the steam tunnel were designed with blow-away siding and blow-out panels, respectively. These designs result in fully vented structures which are not affected by the differential atmospheric pressures. These structures have been designed for the full effects of the windward and leeward tornadic wind pressures.

The Fermi 2 design has considered the differential pressure effects resulting from the tornadic winds in a manner consistent with the SRP requirements.

Tornado Missiles

Sections 3.3.2 and 3.5.1.4 of the SRP require that nuclear plants protect safety-related equipment against damage from missiles which might be generated by the design basis tornado. During the period of issuance of the SRP, additional research was in progress to provide definitive guidelines on the most likely and most damaging missiles which should be postulated. Ultimately, Revision 1 of the SRP specified that structures and components withstand at least missiles C and F (SRP 3.5.1.4 Rev. 0 Missile Spectrum). The criteria and procedures utilized for the design of Category I structures, shields, and barriers to withstand the effects of these missiles are provided in SRP Section 3.5.3.

A comparison of the missiles considered in the design of Fermi 2 and the minimum spectrum of missiles required by the SRP is provided in Table 5-3. The tornado missiles designated as (General) represent the spectrum of missiles which was agreed upon by the Atomic Energy Commission (AEC) during the Construction Permit review stage for Fermi 2. This was several years prior to the development of the SRP missile spectrum. Additional tornado missile evaluations were performed during the course of Fermi 2 completion. Specifically, in approximately 1975, at the request of the NRC, Detroit Edison assessed the degree of comparability of the Fermi 2 tornado missile protection with other in-process nuclear plant designs which addressed the SRP missiles C and F. Missiles C and F are the steel rod and the utility pole, respectively. Using the results of recent tests, similar to those summarized by Jankov, et. al., in "Proceedings of the Symposium on Tornadoes, Assessment of Knowledge and Implications for Man" **[5.14]**, Detroit Edison showed that the missile C and F velocities would not result in penetration or backside scabbing of the minimum 18-inch thick reinforced concrete slabs and walls used in the Category I building design.

Additional detailed studies were completed which provided overall tornado missile hazard estimates for essential plant features which were not explicitly missile protected **[5.15, 5.16, and 5.17]**. These features included the RHR mechanical draft cooling towers and unprotected reactor/auxiliary building wall openings such as the south wall equipment access door, HVAC intake enclosure, and removable precast panel. These studies considered the full spectrum of SRP missiles, the site specific quantities and locations of missiles, and advanced methods of tornado missile hazard modeling. These studies concluded that the vulnerability of these unprotected plant features to damage from tornado generated missiles was acceptably low.

Combination of Load Components

SRP Section 3.3.2 provides acceptable criteria for the combination of the tornado load components to develop the total tornado load which is then combined with other loads per SRP Sections 3.8.1, 3.8.4 and 3.8.5.

The Fermi 2 Category I structures have been designed to withstand the simultaneous effects of tornado wind velocity pressure, differential atmospheric pressure, and a single tornado generated missile. The wind velocity pressure distribution and atmospheric pressure distribution are combined and applied at various locations along the buildings. The local effect of a single tornado generated missile is then superimposed to assure that it will not penetrate the structure or cause back side concrete spalling.

The Fermi 2 design combination of load components conforms with the SRP requirements.

Failure of Other Structures and Components

The SRP requires that the effect on Category I structures and components of the failure of structures and components not designed to withstand tornado loading must be considered.

The Fermi 2 design considered the effect of the tornado induced failure of non-Category I structures on Category I structures. In general, the location of the non-Category I structures precludes significant interaction. Structures such as the natural draft cooling towers are located at least one cooling tower height from the nearest safety-related structure. The debris which would be generated from the tornado induced failure of non-Category I structures has been inherently considered in the tornado missile resistant design of the Category I structures and in the detailed missile hazard studies performed for the plant.

Additionally, Detroit Edison maintains a practice of evaluating, through the Safety Evaluation (SE) process, any additional tornado missile hazard presented by the addition of non-Category I facilities such as the permanent outage building or temporary trailers and materials placed on the site for outage support.

The effects of tornado-induced failures of structures or components not designed to resist tornadoes have been adequately considered in the design of the Fermi 2 Category I structures and satisfy the intent of the SRP.

5.1.2 Plant Walkdown

Walkdowns were conducted at the Fermi 2 site on March 22 and 23, 1995. To identify potential vulnerabilities to high winds and tornadoes, the walkdowns focused on the exterior of the plant power block structures, safety-related components outside of the power block structures, and other facilities, equipment and material situated around the plant site.

The elements of the Fermi 2 wind resistant design which were assessed are:

- General ruggedness of the Category I structures and exposed Category I components to resist tornadic winds;
- Materials (type and quantity) around the plant site which could become tornado missiles;
- Barriers and protection to prevent the entry of tornado-generated missiles into Category I structures;
- Potential for tornado missile impact on exposed Category I components;
- Effect of tornado missile impact on exposed Category I components; and
- Potential effects of wind/tornado damaged non-Category I structures and components on Category I structures and components.

The conclusions drawn from the walkdown are summarized in the following paragraphs. Overall, the field condition provided no unexpected findings.

The design of the Category I structures provides adequate tornadic wind resistance and missile protection since it utilizes reinforced concrete construction with minimum 18-inch thick walls and slabs. No parts or portions of these structures appeared to have any vulnerability to the design basis tornado maximum wind velocity.

There are no safety-related components which are exposed at Fermi 2. All components are adequately protected by reinforced concrete structures or barriers. The diesel exhaust pipes are a concern at some facilities since significant portions are sometimes exposed and could experience damage resulting from tornadic winds or missiles. The four diesel exhausts and intakes are well protected at Fermi 2 by reinforced concrete structures on the roof of the RHR complex. Only a small portion of each exhaust is exposed from the east, and that portion is inherently rugged since it consists of a short tail pipe supported by a stanchion.

Prior to the walkdown, the walkdown team reviewed the study which developed tornado missile hazard estimates considering the non-missile protected openings and penetrations in the reactor /auxiliary building **[5.15]**. The calculation conservatively evaluated the total cumulative area of all openings which were not specifically designed to provide missile protection. During the walkdown the size and location of openings and penetrations in the Category I structures were observed. The walkdown team did not identify openings which had not been evaluated in the study. Similarly, the RHR complex was observed to identify unprotected openings which might result in vulnerability to tornado missiles. The walkdown team did not identify any significant unprotected openings.

The team reviewed the overall site area during the walkdown to gain an appreciation for the quantity and type of materials which might be entrained in a tornado and later become damaging missiles. In general, the total population of missiles assumed in the aforementioned study appears conservative. Observations during the walkdowns indicated a potentially lower missile population. Therefore, the Fermi 2 tornado missile design envelopes the field condition.

During the walkdown, the team also reviewed the potential for unacceptable interactions with Category I structures resulting from the failure of non-Category I structures. Many of the non-Category I structures, such as the turbine building and the fifth floor of the reactor/auxiliary building are clad with light-gauge metal sheathing. This sheathing is not designed to withstand maximum tornadic winds and will partially or fully release. However, this sheathing, considered as a missile, is clearly bounded by the other design basis tornado missiles with significantly greater kinetic energies. With the sheathing released, the superstructures are sufficiently rugged to withstand the maximum wind pressure. No significant potential interactions were observed.

5.1.3 Conclusions and Recommendations

The evaluation of high winds focused on four principal elements of the SRP criteria:

- Definition of climatic conditions which affect the plant site;
- Evaluation of high wind loading;
- Evaluation of tornadic wind loading; and
- Evaluation of tornado-generated missiles.

Though the Fermi 2 design was not committed to the SRP, the design features are consistent with the criteria recommended by the SRP. The Fermi 2 wind resistant design is robust and has adequately considered the aspects of extreme wind hazard which may occur in the area. High or extreme winds do not pose a significant threat to safe operation of Fermi 2.

This evaluation of the Fermi 2 design for the effects of high winds also considered the lessons learned from Hurricane Andrew on Turkey Point Nuclear Generating Station documented in NRC Information Notice 93-53 **[5.54]** including Supplement 1. These lessons are related to adequacy of the following:

- Plant shutdown timing;
- Off-site communication after the disaster;
- Compensatory measures for equipment at facilities not designed for the event;
- Early preparations; and
- Impact of non-safety equipment on important equipment.

Events such as a tornado, which is the controlling extreme wind event for Fermi 2, do not allow the early preparation which hurricanes allow. The most critical defense against plant degradation resulting from a tornado strike is adequate tornado-resistant design. The design must include consideration of structures and barriers to protect all essential equipment from not only wind pressure effects but the effects of the failure of non-safety structures and components.

This review has concluded that the design of Fermi 2 is adequate to resist the effects of tornadoes which are postulated to occur in the area including the impact of non-safety equipment on safety-related equipment.

5.2 EXTERNAL FLOODS

Extreme floods (or high water level) present a potential threat to a nuclear power plant. High water levels at a plant site can be caused by a single source or a combination of sources: stream flooding, surges, seiches, tsunamis, dam failures, landslides, and ice melt. The water levels associated with storm surges are generally much higher than those associated with other sources for plant sites located near large bodies of water. Therefore, at coastal sites, storm surge may provide the controlling water level which must be used in the design of nuclear power plant structures and the protection of components required to safely operate the plant.

High water has a number of effects on structures within its path. High water can apply effective external pressures to structures (hydrostatic loads), create buoyant forces (uplift) on closed structures, and apply dynamic forces generated by wave activity. High water associated with flooding, rather than normal groundwater elevation, often provides the controlling loads which must be considered in the design of nuclear power plant structures and the protection of components required to safely operate the plant.

The following sections provide a discussion of the regional climatology near the Fermi 2 plant site, the licensed extreme flood design basis for the plant, and a comparison of the licensed design with the requirements of the 1975 Standard Review Plan (SRP).

5.2.1 Plant Design Basis

This section provides a comparison of the flood-resistant design of Fermi 2 to the requirements of the SRP. The comparison focuses on seven principal elements of the SRP criteria for the external flooding design of nuclear power plants:

- Description of flood history for the area/basin around the plant site. This includes specific events which must be considered when identifying the controlling flood for the plant site (e.g., stream flooding, surges, seiches, tsunamis, dam failures, landslides, ice) as described in SRP Section 2.4.2;
- Evaluation of probable maximum flood on streams and rivers (SRP Section 2.4.3);
- Evaluation of probable maximum wind tide and surge or seiche flooding (SRP Section 2.4.5);
- Determination of the need to protect plant structures and components for the effects of flooding (SRP Section 2.4.10);
- Evaluation of the effect of local and regional groundwater on the plant structures (SRP Section 2.4.13);
- Evaluation of protection provided for plant structures and components required to withstand the effects of flooding (SRP Section 3.4.1); and

- Review of the analysis procedures used in the design of safety-related structures required to withstand the static and dynamic effects of the design basis flood (SRP Section 3.4.2).

The following sections provide the details of the comparison of the Fermi 2 design to the SRP criteria for these elements.

5.2.1.1 Flood Design Considerations

Category I plant structures housing safety-related equipment consist of the reactor/auxiliary building and the residual heat removal (RHR) complex. The site is not susceptible to flooding caused by surface runoff because of the shoreline location and the distance of the site from major streams. Plant grade is raised approximately 11 feet above the surrounding area to further minimize the possibility of flooding. Flooding of the site is only conceivable as the result of an extremely severe storm with a storm generated rise in the level of Lake Erie. Protection of safety-related structures and equipment against this type of flooding and possible storm-generated waves is provided through the location, arrangement, and design of the structures with respect to the shoreline.

Section 2.4.2 of the Standard Review Plan identifies the following types of flood producing phenomena which must be considered in establishing the flood design bases for safety-related plant features:

- Stream Flooding;
- - probable maximum flood (PMF) with coincident wind-induced waves, considering dam failure potential due to inadequate capacity, inadequate flood-discharge capability or existing physical condition,
- - ice jams, both independently and coincident with a winter probable maximum storm,
- - tributary drainage area PMF potential, and
- - combinations of less severe river floods, coincident with surges and seiches.
- Surges;
- - probable maximum hurricane (PMH) at coastal sites,
- - PMH wind translated inland and resulting wave action coincident with runoff induced flood level,.
- - probable maximum wind-induced (non-hurricane) storm surges and waves, and
- - combinations of less severe surges, coincident with runoff floods.
- Seiches;
- - meteorologically induced in inland lakes (Great Lakes and harbors) and at coastal harbors and embayments,
- - seismically induced in inland lakes,
- - seismically induced by tsunamis (seismic sea waves) on coastal embayments, and
- - combinations of less severe surges and seiches, coincident with runoff floods.

- Tsunamis;
- - near field, or local, excitation, and
- - far field, or distant, excitation.
- Seismically induced dam failures, and maximum water level at site from;
- - failure of dam during safe shutdown earthquake (SSE) coincident with 25-year flood,
- - failure during operating basis earthquake (OBE) coincident with standard project flood (SPF),and
- - failure during other earthquakes, coincident with runoff, surge, or seiche floods where the coincidence is at least as likely as an SSE or OBE dam failure.
- Flooding caused by landslides;
- - flood waves; and
- - backwater effects due to stream blockage.
- Ice loadings from water bodies

These phenomena were evaluated during the original plant design. Many of the phenomena were ruled out as either not applicable or not capable of creating consequences as significant as other phenomena. There have been no significant changes in the site area or characteristics nor the regional climatology which change the applicability or relative significance of these flood producing phenomena.

The original design considered the following phenomena in determining the controlling event for the design basis flood level at Fermi 2. Consideration of these phenomena is compared to the SRP criteria for purposes of the IPEEE review:

- Local probable maximum precipitation (PMP) runoff on the plant site coincident with runoff from the two square mile area above the plant site, assuming blockage of plant drainage;
- Probable Maximum Flood (PMF) on Swan Creek coincident with the mean monthly maximum water level in Lake Erie; and
- Probable maximum wind tide coincident with the mean monthly maximum water level in Lake Erie.

Therefore, the Fermi 2 plant design adequately considers potential flood producing phenomena to determine the controlling event for the design basis and satisfies the intent of the SRP.

5.2.1.2 Effects of Local Intense Precipitation

Local intense precipitation or Probable Maximum Precipitation (PMP) procedures and criteria have changed since the original design of Fermi 2. These changes are the subject of NRC Generic Letter 89-22 **[5.18]**. The latest National Oceanic and Atmospheric Administration/National Weather Service (NOAA/NWS) publications provide PMP estimates for drainage areas as small as one square mile and for durations as short as five minutes.

The original design for the Fermi 2 plant considered the effects of local intense precipitation as defined by practices and procedures at that time. The local PMP was determined to be 10.2 inches of rainfall in the maximum one-hour period based on the U.S. Weather Bureau, Hydrometeorological Report No. 33 **[5.19]**. Plant design considered the effect of PMP on:

- Category I structures (roof ponding);
- Yard and site drainage (local ponding); and
- Site flooding due to runoff from an adjacent two-square mile drainage area west of the plant.

Based on the changes in procedures and criteria, revised estimates for short duration storm intensities were calculated using the guidance in NOAA/NWS HMR Reports 51, 52, and 53 **[5.20, 5.21 and 5.22]**. The maximum point rainfall (PMP) for a one-hour duration and a one-square mile area is taken directly from Figure 24 of HMR Report No. 52. For the Fermi 2 plant site this corresponds to an accumulation of 17.3 in. This is a substantial increase in the postulated short-duration (one-hour) rainfall. The effect of the increase in rainfall intensity on local site ponding and roof ponding was addressed as discussed in the following paragraphs.

Site Ponding

The following describe significant elements of the original plant design for site ponding due to local intense precipitation **[5.2]**:

- The plant site storm drainage system was not relied upon to protect Category I structures from local PMP flooding.
- Flooding due to a local PMP on the adjacent two-square mile drainage area west of the plant site was evaluated as part of the original plant design. The calculated peak discharge due to the local PMP was determined to be 25,000 cubic feet per second (cfs), which is 10,000 cfs greater than indicated by the PMF peak envelope curve for the Great Lakes Region. A flow of 31,500 cfs was passed through a hypothetical cross section to determine maximum water level at the plant site. The maximum water surface elevation determined in this conservative analysis was 582 feet, more than one foot below plant grade.
- Runoff from the plant site will flow overland under conditions of site gradient to lower elevations surrounding the site and then to Lake Erie. All door sills on safety-related structures are at least six inches above plant grade. To assure that runoff is not directed towards the openings of plant structures, no downspouts or scuppers are located near doorways.

- Some Category I yard structures (e.g., pipe tunnels and duct banks) are located below the site design flood elevation. These structures are reinforced concrete and are designed for continuous underwater service.

The new PMP was determined to have no effect on the capability of the Fermi 2 plant site and/or structures to protect safety-related components from the effects of flooding, as documented in DER Number 89-1284 **[5.23]**.

If the new PMP values are used, flood flows due to PMP runoff would be expected to increase significantly in the two square mile local area to the west of the plant and the immediate plant site area. The water level resulting from the increased peak flood flow was estimated to be 585.2 feet. This is very conservative since the cross section is maintained as water level increases. This water level is still less than the water level predicted for the Probable Maximum Meteorological Event (PMME) used as the site design basis.

Plant peripheral roads have crown elevations generally ranging from 582.5 to 583.5 feet. During local intense precipitation on and around the plant structures, run-off could be diverted over the area enclosed by the perimeter roads and rail tracks. However, based on the plant design, where road elevations are generally at or below plant grade, it is unlikely that water levels would exceed those established for run-off from the adjacent two square mile basin.

Local site flooding or ponding to the extent required to challenge the plant design is not credible. The controlling flood level for plant design remains the PMME. Therefore, the Fermi 2 plant design for local intense precipitation with respect to local site ponding satisfies the SRP intent (considering the new criteria and procedures for PMP).

Roof Ponding

The following items describe significant features of the original plant design for roof drainage and/or ponding due to local intense precipitation **[5.2]**:

- The reactor/auxiliary building roof is designed for an operating live load of 30 pounds per square foot (psf). The design basis (extreme load case) live load is 87 psf. This load is equivalent to 16.7 inches of water. Roof drains are designed for a rainfall of four in/hr.
- The roofs of the RHR complex are designed for a postulated maximum operating basis ice and snow load of 70 psf. This load is based on the simultaneous accumulation of the most severe postulated ice resulting from the mechanical draft cooling towers drift loss (21 psf) plus the seasonal snowpack (30 psf), and an additional ice load (19 psf). This load is equivalent to 13.5 inches of water. The

design basis live load is 276 psf, which is equivalent to 53.0 inches of water. This depth of water exceeds the roof parapet height.

The Fermi 2 safety-related structures were evaluated for increased ponding loads resulting from the new PMP rainfall intensities, as documented in DER Number 89-1284 **[5.23]**. The effect of the new PMP values on roof loads was assessed by estimating the potential maximum depth of accumulation of water on the roofs of the reactor building and the auxiliary building. The maximum water accumulations were estimated by accounting the inflow to the roof due to PMP, the outflow from the roof due to the discharge through the scuppers and conductors, and the depth of accumulation of rainwater on the roof during the periods when the rate of rainfall exceeds the rate of discharge through the roof drains.

The resulting maximum average depth of water was determined to be 14.4 inches on the reactor building roof and 27.32 inches on the auxiliary building roof. Since the reactor building roof slopes with a 15.5-inch total drop over its 124.75-foot length, the accumulated maximum water depth would vary from approximately 6.7 inches at the high end to 22.2 inches at the low end. The roof deck, purlins, and roof girders were investigated for the additional live loads and were determined to have sufficient margin.

The depth of accumulation on the auxiliary building roof would vary from 18.0 inches at the high end to 36.7 inches at the low point. The estimated accumulation at the roof low point is slightly greater than the height of the parapets. The resulting live load due to ponding is 190.8 psf. The auxiliary building roof slabs were investigated for this extreme loading condition and were shown to have adequate margin.

Potential in-leakage through roof joints such as air intakes or exhausts was addressed as part of the walkdown discussed in Section 5.2.3.

5.2.1.3 Probable Maximum Flood on Streams and Rivers

The Probable Maximum Flood (PMF) is an estimated flood that may be expected from the most severe combination of critical meteorological and hydrological conditions that are reasonably possible in the region. The PMF on Swan Creek was estimated as the maximum flood runoff resulting from a PMP occurring over its drainage basin of 109 square miles and was considered in the original design of the Fermi 2 plant. There are no other streams or rivers near the plant site that could significantly affect plant structures or equipment.

Since the procedures and criteria used to determine probable maximum precipitation have changed, the water level resulting from a flood on [illegible] Creek could be affected. To assure that the new criteria and procedures do not r[illegible] water levels in excess of the design basis, flooding on Swan Creek due to the new [illegible] intensities was evaluated as documented in DER 89-1284 **[5.23]**. The evaluation documented in DER 89-1284 concluded that the change in rainfall intensities could result in a higher flood flow rate from Swan Creek and a maximum water level between 583 and 584 feet. This water level is still less than the water level associated with the PMME of 586.9 feet. Therefore, the flood levels associated with the PMF flow on Swan Creek remain less critical than storm surge in Lake Erie.

Development of the original (design basis) PMF on Swan Creek considered all of the significant parameters required by the SRP and used recognized and accepted methods. Since the changes in criteria and procedures for determining PMP do not increase flooding on Swan Creek to a level in excess of the PMME (storm surge on Lake Erie), stream flooding does not control plant design.

5.2.1.4 Probable Maximum Surge and Seiche Flooding

The original design basis for surge and seiche flooding is described in detail in Section 2.4.5 of the Fermi 2 UFSAR **[5.2]**. Some of the more significant considerations are discussed briefly in the following paragraphs.

Probable Maximum Winds

Extensive studies were made regarding the effects of wind setup on Lake Erie. Data developed by G. W. Platzman, which relates lake level to various wind conditions, was used to establish the wind setup for the plant site. The Platzman model has been shown to consistently calculate peak longitudinal setup greater than the measured peak setup when using the wind stress and bottom friction coefficients proposed by Platzman. The conservatism of the model in predicting the longitudinal setup increases with increasing wind speeds.

To establish meteorological conditions appropriate for the plant site, the National Weather Service was commissioned to examine 25 years of wind records for eight stations in the vicinity of Lake Erie. Significant parameters used to calculate the probable maximum wind tide at the Fermi 2 site were obtained from the table of probable maximum wind estimates supplied by the Atomic Energy Commission. The following Probable Maximum Meteorological Event (PMME) data was used to calculate the wind tide:

- The peak ten-minute wind speed was 100 mph;
- The PMME winds were directed along the axis of Lake Erie; and
- The PMME had a translational velocity of 20 mph moving from east to west and a duration of 60 hours.

Surge and Seiche History

Data collected by the U.S. Lake Survey at gages near Monroe (from 1932 to 1939 and from 1952 to 1973), Gibraltar (from 1897 to 1973), and Toledo (from 1897 to 1973), indicate that the maximum wind tide at Monroe was 4.5 feet on January 30, 1939. Earlier data, covering the period 1886 to 1896, reported a maximum wind tide of 5.5 feet at Monroe. The description of the easterly gales that produced this wind tide suggests that they were more intense than those reported since 1897. Therefore, 5.5 feet was accepted as the maximum wind tide occurring since 1886.

Based upon data collected by the U.S. Lake Survey, the highest observed monthly mean water level in Lake Erie during the period of record from 1860 to 1973 was 4.9 feet above Low Water Datum. This level occurred during June 1973, at Monroe, Michigan. During 1973, the monthly mean water level varied between 3.0 and 4.9 feet above Low Water Datum.

Data for the period from 1970 through 1994 are consistent with the monthly mean water levels identified during the original design and construction phases for Fermi 2 **[5.24 and 5.25]**.

Surge and Seiche Sources

Maximum surge stillwater elevation was based on the Platzman wind setup model for Lake Erie. This is consistent with the SRP requirements for estimating maximum surge or seiche stillwater elevations for Great Lakes sites.

A maximum wind tide of 11.4 feet was calculated for the Fermi 2 site using the PMME wind speeds as input to the Platzman model. As an additional conservatism, a wind tide of 11.6 feet was used for design purposes.

A total stillwater elevation of 16.4 feet (elevation at the plant site of 586.9 feet) was selected as the design maximum water level. This maximum was based on the PMME defined by the Atomic Energy Commission resulting in a wind tide of 11.6 feet superimposed on a maximum monthly mean lake level of 4.8 feet. This storm surge would occur at the plant site approximately nine hours after the maximum wind reaches the shore.

Wave Action

Wave characteristics are dependent upon wind speed, wind duration, water depth, and fetch length. Waves were calculated coincident with the maximum storm surge to determine the maximum flood elevations at the plant site. The shallow water depths over the fetch approaching the Fermi 2 plant site preclude deep water wave activity. Only shallow water waves are generated during the PMME.

During the occurrence of the PMME, plant grade at elevation 583.0 feet is flooded for approximately 17 hours. Incident waves attacking the shoreline can be transmitted inland across the flooded plant grade. These transmitted wave heights depend on the available water depth above plant grade, the incident wave characteristics attacking the shoreline, the configuration of the shore barrier, and the location and configuration of other obstacles.

Shallow water wave generation was based on "Shore Protection Planning and Design," Technical Report No. 4 from the U. S. Army Coastal Engineering Research Center **[5.26]**. This is consistent with the SRP requirements for estimating wind generated waves and run-up. Using the inland depth of water caused by flooding of plant grade, the maximum wave heights for plant grade elevation 583.0 feet and 580.0 feet are 3.0 feet and 5.4 feet, respectively.

Resonance

Resonance can be a problem in enclosed bays or harbors when the natural period of oscillation of the bay is equal to the period of the incident waves. The Fermi 2 site is not located in an enclosed bay. The full exposure of the site to Lake Erie during PMME conditions, plus the flat slopes surrounding the site area, result in a natural period of oscillation of the flooded area that is much greater than that of the incident shallow water storm waves. Therefore, resonance will not occur at the site during the PMME conditions.

Wave Run-up

Wave run-up was based on "Shore Protection Planning and Design," Technical Report No. 4 from the U. S. Army Coastal Engineering Research Center **[5.26]**. This is consistent with the SRP requirements for estimating wind generated waves and run-up. Maximum run-up elevations on the exposed north faces of the reactor/auxiliary building and the RHR complex are 593.0 feet and 598.0 feet for the 3.0 and 5.4 foot waves, respectively. The maximum run-up elevation on the exposed south faces of the reactor/auxiliary building and the RHR complex, the exposed east face of the RHR complex, and the west face of the reactor/auxiliary building is 593.0 feet for the 3.0 foot wave. The east face of the reactor/auxiliary building is not exposed to waves and wave run-up. The west face of the RHR complex is landward of the storm direction and not subject to waves and wave run-up.

Comparison to SRP

Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants" **[5.27]** Appendix A adopts American National Standards Institute (ANSI) Standard N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites" **[5.28]** for determining high water levels. ANSI Standard N170-1976 has been revised and re-issued as ANSI/ANS-2.8-1992, "Determining Design Basis Flooding at Power Reactor Sites" **[5.29]**. For Probable Maximum Surge and Seiche Flooding (Section 7), ANSI/ANS-2.8-1992 sets basic input parameters for the Great Lakes Region that are consistent with those used in the Fermi 2 design:

- Maximum overwater wind speed of 100 mph;
- Constant translational speed;
- Wind speeds over water vary from 1.3 to 1.6 times the overland speed; and
- Storm center moves along a critical path oriented along the major axis of the water body.

In addition, the models used to develop the maximum surge with coincident wind generated waves and runup are consistent with those required by the Standard Review Plan and recommended in ANSI/ANS-2.8-1992. More recent information from the National Weather Service, U.S. Coast Guard, and U.S. Geological Survey indicates that the parameters used as input for the original plant design have not changed significantly in the past twenty years. Therefore, since the original design for surge and seiche flooding considered the parameters required by the 1975 SRP, and those parameters have not changed significantly, the Fermi 2 plant design satisfies the SRP intent.

5.2.1.5 Flooding Protection Requirements

SRP Section 2.4.10 requires that the locations and elevations of safety-related facilities and of structures and components required for protection of safety-related facilities be reviewed in comparison with the estimated static and dynamic effects of design basis flood conditions to determine whether flood effects need to be considered in plant design or emergency procedures. Because the postulated elevation of the design basis flood for Fermi 2 exceeded the ground elevation of safety-related structures, flood protection was required. Specific measures taken to protect structures and components at Fermi 2 are discussed in Section 5.2.1.7.

Since there is no change in the maximum water level resulting from the Probable Maximum Meteorological Event, the original determination for plant flood protection requirements is acceptable and the Fermi 2 design satisfies the SRP intent.

5.2.1.6 Ground Water

Groundwater can exert hydrodynamic loads on safety-related structures. SRP Section 2.4.13 requires local and regional groundwater to be evaluated to assess its effect on plant foundations.

Section 2.4.13 of the Fermi 2 UFSAR discusses ground water design considerations. The Fermi 2 design does not use a dewatering system to lower design basis ground water levels. The natural groundwater level at the site is approximately 575 feet. As a conservative value for computing normal subsurface hydrostatic loadings, the ground water level was assumed to be 576.0 feet for design.

Because the postulated water level during the PMME exceeds the natural groundwater level, the Category I plant structures are designed to withstand the hydrostatic loading associated with the maximum water level. All safety-related systems and components which are below the maximum water level are located within Seismic Category I structures.

5.2.1.7 Flood Protection

In accordance with Standard Review Plan requirements, safety-related facilities, as well as structures and components required for protection of safety-related facilities, must be designed for the estimated static and dynamic effects of design basis flood conditions. Specifically, the locations of safety-related systems and components that must be protected against flooding must be identified along with the structures that house the equipment to determine whether or not the equipment are subject to flooding and/or the structures relied upon to protect the equipment from flooding are adequately designed to withstand those effects.

At Fermi 2, all Category I components are protected from the adverse effects of the maximum flood by their location within reinforced concrete Category I structures. All Seismic Category I structures are designed against flooding to a minimum elevation of 588 feet, or 1.1 feet above the maximum stillwater elevation. Flood protection measures incorporated into the design include waterproofing the structures, designing the structure to withstand the hydrostatic and hydrodynamic forces associated with flooding, maximum usage of watertight seals and penetrations below the flood level, using exterior doors of watertight design below the maximum flood elevation, providing waterstops on all construction joints and water seal rings on all penetrations below the maximum flood level, and locating the Category I components within the reinforced concrete Category I structures.

Since there is no change in the maximum water level resulting from the Probable Maximum Meteorological Event, the flood protection measures incorporated into the original design are still applicable. Therefore, the Fermi 2 design satisfies the SRP.

5.2.1.8 Analysis Procedures

As required by the Standard Review Plan, the design of safety-related structures must assure that the static and dynamic effects of the design basis flood and highest ground water have been adequately transformed into effective loads on the structures. Specifically, the hydrostatic head associated with the design basis flood level (or maximum ground water, whichever controls) must be considered as a structural load on basement walls and foundation slabs. The effects of buoyancy on the structure must be considered. Lateral and overturning pressure on sidewalls and slabs must be considered.

Seismic Category I structures at the Fermi 2 plant would be partially submerged with the PMME water level at elevation 586.9 feet. Accordingly, the design considered the following load conditions:

- Hydrostatic pressures;
- Uplift pressure or buoyancy; and
- Static and hydrodynamic forces associated with wind generated waves.

The pressure induced by the maximum stillwater elevation was considered to be hydrostatic. A lateral pressure distribution on the structure walls below the flood line was considered in the design of the structures. From this, the uplift pressure on the basement slabs and flotation potential were calculated. This pressure was included in the load combinations considered in the design of the slabs.

Maximum wave pressures and forces on Category I structures were calculated for the significant wave heights of 3.0 feet or 5.4 feet for plant grade elevations of 583.0 feet or 580.0 feet, respectively. Wave pressures and thrusts were calculated for non-breaking, broken, and breaking wave conditions consistent with SRP requirements. The critical static pressure and thrust occur under the broken wave conditions, whereas the critical dynamic pressure and thrust occur under the breaking wave condition. All structures are designed to withstand these forces.

5.2.2 Plant Walkdown

A plant walkdown was conducted at the Fermi 2 site on March 22 and 23, 1995. The purpose of the plant walkdown was to assess the vulnerability of plant structures and equipment to external flooding. The plant walkdown included trips inside the protected area and inside plant structures. The walkdown also included a general survey of the plant site and surrounding area.

The objectives of the plant walkdown were to:

- Verify flood protection of structures and equipment;
- Perform a general assessment of the site and surrounding area topography to identify significant areas of runoff and restrictions or diversions to that runoff (i.e., potential for local site ponding); and
- Assess overall site drainage capabilities.

These objectives were achieved as discussed in the following paragraphs.

Flood Protection

The Fermi 2 plant site is considered an incorporated barrier design as defined in RG 1.102 **[5.30]**. That is, safety-related structures, systems, and components are protected from inundation and static and dynamic effects by engineered features in the structure/environment interface. As stated in the UFSAR, the design basis flood level is 586.9 feet and the ground floor of safety-related structures is 583.5 feet. Therefore, the structures and components must be protected from the effects of flooding by engineered features. Flood protection is provided by designing the structures to be watertight (e.g., watertight doors, penetrations) and to resist the loadings resulting from high water (e.g., hydrostatic pressure, wave forces).

During the plant walkdown, site topography was observed to be generally sloping away from safety-related structures. Therefore, local intense precipitation would quickly be directed away from plant structures toward the discharge canal, adjacent ditches and streams, to its ultimate destination in Lake Erie.

The walkdown included a detailed review of roof drainage for the reactor/auxiliary building and the RHR complex. The reactor building roof is a steel superstructure with decking. The roof slopes from west to east and drainage is provided on the east end of the roof. Drainage consists of four-inch diameter scuppers through the parapet walls. One objective of the walkdown was to identify potential sources of in-leakage under ponding conditions. No sources of in-leakage were noted with the exception of the doorway leading onto the roof. However, the doorway is elevated a minimum of 18 inches above the Reactor Building roof so in-leakage is not a concern.

The auxiliary building roof is reinforced concrete with additional support provided by structural steel under portions of the roof. The roof slopes from west to east and drainage is provided on the east end of the roof. Drainage consists of a series of six-inch diameter sumps. The sumps conduct runoff from the roof to the plant drainage system. Ponding was postulated to reach full height of the parapet at the roof low end. No sources of in-leakage were noted with the exception of the doorway leading onto the roof. The doorway is elevated a minimum of 18 inches above the auxiliary building roof and is located at the high end of the roof; therefore, in-leakage is not a concern.

The RHR complex roofs are reinforced concrete. Drainage consists of six-inch diameter scuppers through the parapet walls. Because the RHR complex includes mechanical draft cooling towers, a prim .y design consideration for the roofs of the complex is snow and ice build-up due to drift. The original design considered a build-up of snow and ice which, in its water equivalent, exceeds the height of the parapet walls. Therefore, ponding on the roofs during PMP conditions is not a structural concern. The walkdown considered that water may pond to a depth slightly greater than the parapet height for the purpose of identifying sources of in-leakage. No sources of significant in-leakage were noted. Air intakes and vents were designed such that their low points exceed the parapet height.

The walkdown also included a review of the exterior of the building structures to confirm flood protection. At Fermi 2 the design basis maximum stillwater elevation is 586.9 feet. The structures are protected from flooding to a minimum elevation of 588 feet. The walkdown included a survey of the exterior surfaces of Category I structures to verify that penetrations, doorways, and other openings at or below the flood level are watertight. Results of the walkdown confirmed that Category I structures are protected to a minimum elevation of 588 feet. The walkdown included verification of installation of watertight doors at locations required to protect safety-related equipment below the flood level and that the doors are normally maintained in a closed position. As additional assurance of flood protection, these doors are identified in the station abnormal operating procedures **[5.31]** and require verification that they are properly secured closed during flood conditio ns.

Four vent p' oes penetrate the east wall of the RHR complex below the design basis flood elevation. F 'om the walkdown it was unclear whether or not these vent pipes could be a source of in-leakage. Subsequent review of plant design documents indicated that the vent pipes have normally closed butterfly valves to provide isolation and protection against flooding.

Site Topography

General site topography was observed from the roofs of safety-related structures and by driving on site access roads and local roadways. There are lagoons and wetlands to the north and south of the plant site. The western edge of the plant site is also largely comprised of wetlands and lagoons, but does represent a potential source of runoff to the site area, albeit minimal. The eastern edge of the plant site is formed by the shore of Lake Erie. The plant site area is very flat with plant grade sloping away from Category I structures.

Runoff from surrounding areas would be intercepted by the wetlands and lagoons and directed toward its ultimate destination in Lake Erie. Runoff from the plant site would be directed via plant grade and/or the plant drainage system to the wetlands, lagoons, and Lake Erie. As a result, local site ponding of a magnitude equivalent to the design basis flood is not credible.

Site Drainage

Site drainage facilities (ditches, culverts, sewers) were assessed during the plant walkdown. It was concluded that the facilities have adequate capacity for the most common storms observed at the plant site and, in general, provide adequate drainage for the ten-year and 100-year storms. Clearly, the drainage system would be overcome by precipitation associated with the PMP. However, even in the event the drainage system is overcome, the natural drainage for the site and surrounding areas is away from plant structures. Local site ponding to a depth required to challenge the plant design basis as controlled by the design basis flood elevation of 586.9 feet is not credible. The crest of plant roads and the railroad spur are typically at or below plant grade. Only limited areas of the roadways are above plant grade. Therefore, site runoff would not be confined around the Category I facilities.

5.2.3 Conclusions and Recommendations

The evaluation of the Fermi 2 flood-resistant design focused on seven principal elements of the SRP criteria:

- Description of flood history for the area/basin around the plant site. This includes specific events which must be considered when identifying the controlling flood for the plant site (e.g., stream flooding, surges, seiches, tsunamis, dam failures, landslides, and ice);
- Evaluation of probable maximum flood on streams and rivers;
- Evaluation of probable maximum wind tide and surge or seiche flooding (i.e., Probable Maximum Meteorological Event);
- Determination of the need to protect plant structures and components for the effects of flooding;
- Evaluation of the effect of local and regional groundwater on the plant structures;
- Evaluation of protection provided for plant structures and components required to withstand the effects of flooding; and
- Review of the analysis procedures used in the design of safety-related structures required to withstand the static and dynamic effects of the design basis flood.

The review demonstrates that the design of Fermi 2 satisfies the intent of the 1975 Standard Review Plan criteria for protection against external flooding.

5.3 TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

Nearby industrial, transportation, and military facilities can present a potential threat to the safe operation of a nuclear power plant. Consequences of transportation accidents or accidents at nearby industrial or military facilities can involve direct collision, pressure loading, missile impact, fire, vapor cloud detonation, and/or drifting of toxic fumes into the control room leading to potential degradation of plant facilities and equipment or incapacitation of plant operators.

The severity of events and the potential vulnerability of Fermi 2 to such accidents are evaluated in this section. Transportation routes and industrial or military facilities within a five-mile radius of the plant site are considered. The U.S. Nuclear Regulatory Commission 1975 Standard Review Plan (SRP) **[5.3]** and Regulatory Guide (RG) 1.70 **[5.32]** provide guidance for evaluating such events and their impact on the plant.

The following sections provide a discussion of the location of facilities in the area of the site, the potential threat to the plant, and a comparison of the Fermi 2 licensed design basis with the requirements of the 1975 SRP.

5.3.1 Plant Design Basis

This section provides a comparison of the Fermi 2 design to SRP requirements with respect to nearby industrial, transportation, and military facilities accidents. The comparison focuses on two principal elements of the SRP criteria:

- Identification of potential hazards in the site vicinity (SRP Sections 2.2.1 and 2.2.2); and
- Evaluation of potential accidents (SRP Section 2.2.3).

The following sections provide the details of the comparison of the Fermi 2 design with the SRP criteria for each element.

5.3.1.1 Site Location and Description

The Fermi 2 plant is located on an 1120-acre site in Frenchtown Township, Monroe County, Michigan. The site is situated on the western shore of Lake Erie approximately 5.5 miles northeast of the Monroe City limits. The nearest large cities are Detroit, Michigan, approximately 30 miles to the northeast, and Toledo, Ohio, approximately 25 miles to the southwest.

Transportation Routes

Several transportation routes are located near Fermi 2. The nearest public roads are Toll Road on the northwestern edge of the site boundary approximately 0.65 mile from the powerblock and Pointe Aux Peaux Road to the south approximately 0.6 mile from the powerblock. These roads carry predominately local traffic. North Dixie Highway passes approximately 1.5 miles west of the plant and also carries predominately local traffic.

The nearest major highway is Interstate 75 which is approximately 4.1 miles northwest of the plant at its closest point. U.S. Highway 24 lies west of Interstate 75 and is approximately 5.8 miles from the plant at its closest point. Fermi 2 was in the design phase in 1971 when annual average 24-hour traffic volume on Interstate 75 and U.S. Highway 24 was 27,300 and 9,200 vehicles, respectively. The most recent data available from the Michigan Department of Transportation (1993) **[5.33]** shows that the traffic volume on these routes, at their nearest point to the Fermi 2 site, has increased to 38,000 and 11,000 vehicles, respectively. Interstate 75 experiences heavy commercial traffic since it is a major access route to industries in the Detroit area.

The nearest railway line to the Fermi 2 site is operated by the CN North America/Grand Trunk Railroad and passes approximately 3.5 miles from the plant at its closest point. This line, which is called the Shore Line Subdivision, is low density in terms of the transportation of hazardous materials. In 1994 approximately 30 tank cars of liquid (unspecified type) were trafficked on this line **[5.34]**. This was the only potentially hazardous material identified, and this material does not present a significant risk factor based on the yearly rail traffic provisions of Regulatory Guide 1.78 **[5.44]**.

The Grand Trunk line has a spur which provides rail access to the plant. Inadvertent or unauthorized rail access to the site is prevented by a secured derailment device adjacent to the first site security post on Enrico Fermi Drive.

Conrail operates two rail lines within a five-mile radius of the plant. The closer is immediately parallel to the Grand Trunk line and is approximately 3.5 miles from the plant at its closest point. The second line is roughly parallel to the first and approximately 3.8 miles from the plant at its closest point.

The Grand Trunk and Conrail rail lines were previously identified in the Fermi 2 design basis but at the time were operated by the Detroit and Toledo Shore Line and Penn Central Railroads, respectively. There are no other rail lines which are operated within a five-mile radius of the Fermi 2 site.

The nearest shipping lanes to the Fermi 2 site are the West Outer Channel and the East Outer Channel in Lake Erie which connect to the Detroit River navigation channel. The closest lane, the West Outer Channel, is approximately five miles east of the plant. The majority of ship traffic uses the East Outer Channel which is significantly wider and deeper. This channel is approximately seven miles east of the site. Both channels are well marked. Large cargo ships could not inadvertently stray from the channels and reach the site since Lake Erie becomes very shallow to the west of the channels. The average lake depth is as little as 14 feet one mile east of the Fermi 2 site.

Nearby Facilities

There are no military facilities within ten miles of the plant.

Additional industry has moved into Monroe County since the Fermi 2 plant was originally sited. The Fermi 2 design basis did not identify any industrial facilities within a five-mile radius of the site which stored hazardous materials in significant amounts. Through discussion with the Monroe County Emergency Management Division (EMD), five industries which store or use hazardous materials are now located within the five-mile radius **[5.35]**. This information was obtained through review of the emergency and hazardous chemical inventory reports filed by the county.

The Meijer Newport Distribution Facility is located south of Swan Creek Road between Interstate 75 and the Conrail rail line. The facility is a large distribution center for grocery products and is approximately four miles from the Fermi 2 site. The facility maintains significant quantities of anhydrous ammonia, used in refrigeration. Average daily inventory reported was approximately 22,000 pounds.

The Frenchtown Township Water Treatment facility is located at North Dixie Highway and Point Aux Peaux Road approximately two miles southwest of the Fermi 2 site. The facility maintains a number of chemicals in small quantities **[5.53]** (but no gaseous chlorine) used for water purification.

The Ameritech (Michigan Bell) substation facility is located on North Dixie Highway approximately 2.5 miles southwest of the Fermi 2 site. The facility utilizes banks of storage batteries and, therefore, maintains sulfuric acid.

The aforementioned facilities all began operation since Fermi 2 was constructed and, therefore, were not discussed in the design basis.

The Rockwood Stone, Inc. Quarry was originally identified in the Fermi 2 design basis and is located approximately three miles northeast of the plant. Access to the quarry is along Reaume Road. The quarry is still operating and continues the use of explosives, primarily ammonium nitrate fuel oil (ANFO), in its process. However, the quarry no longer maintains explosives on the site overnight **[5.36]**. Only the required quantities are transported to the site on the day of the shots. Any unused explosives are removed from the site at the end of the day.

A new quarry recently began operation on the same square mile section of land on which the Rockwood Quarry is operated. The Thompson McCully Quarry is located due west of the Rockwood Quarry and is bounded by Reaume Road on the north and Sunlight Road on the east. The size of the new operation is comparable to the Rockwood operation. Potential quarry hazard evaluation is described in Section 5.3.1.2 below.

In mid-1990 the Edw. C. Levy Co. was investigating the siting of a new quarry near Toll Road, less than one mile from the Fermi 2 site. The proposed process at the quarry required the use of explosives, similar to the aforementioned quarry operations. Detroit Edison evaluated the hazard due to the use of explosives (25,000 pounds) at the proposed site and determined it was acceptable **[5.37 and 5.38]**. The evaluation was submitted to the NRC for review and a Safety Evaluation was issued **[5.52]**. The quarry investigation was subsequently dropped by the Levy Company. There are no plans to continue the investigation or to operate a quarry at the Toll Road location in the foreseeable future **[5.39]**.

There are no significant gas or oil products pipelines within the five-mile radius of the Fermi site. Sun Refining and Marketing, Panhandle Eastern, and Marathon companies operate pipelines in the vicinity, but which are well outside of the radius. Most are several miles west of Interstate 75. There are smaller natural gas distribution lines (four- to six-inch diameter) within 1.5 to two miles of the plant. These lines do not present a hazard to the site.

There is one small airfield, Marshal Field, within a five-mile radius of the Fermi 2 site. This airfield has a grass strip located approximately two miles west of the plant, immediately adjacent to North Dixie Highway. This airfield does not appear active at this time. There are a number of small airports outside of the five-mile radius including Carl's southwest of South Rockwood and Wickenheiser south of Carleton which see infrequent private aircraft traffic and do not present a hazard to Fermi 2. Both Marshal Field and Carl's airports have been addressed by the Fermi 2 design basis.

The closest airports having commercial facilities are Monroe Custer Airport approximately nine miles southwest of the plant, and the Detroit Grosse Ile Airport, approximately 11 miles northeast of the plant. The air traffic, in terms of number of annual departures and arrivals for the Custer and Grosse Ile airports, is 25,000 and 65,000 aircraft, respectively **[5.40]**.

There are two major airports within 25 miles of the Fermi 2 site. The Detroit Metropolitan Airport is approximately 20 miles northwest of the site and experiences approximately 480,000 aircraft departures and arrivals annually **[5.40]**. The Willow Run Airport is approximately 24 miles northwest of the plant site and experiences approximately 159,000 aircraft departures and arrivals annually **[5.40]**. This traffic has increased substantially from that documented in the Fermi 2 design basis.

There are a number of low altitude and high altitude Federal airways that pass over the area around the Fermi 2 site. The airways closest to the site are summarized in Table 5-4 which provides the airway designation, the airway width as given in the Federal Aviation Regulations (FAR) **[5.41]**, the distance from the airway centerline to the site, and the calculated distance from the airway edge to the site. Current Federal Aviation Administration information indicates that the three low-altitude airways are utilized by approximately 200 aircraft per day and the six high altitude jet airways are utilized by approximately 360 aircraft per day **[5.40]**.

There are no military training routes which pass over a five-mile radius around the Fermi 2 site.

5.3.1.2 Evaluation of Potential Accidents

The accident categories below are addressed in the SRP and have been evaluated using the guidance provided by Regulatory Guide 1.70.

Explosions

No large industrial or military facilities exist within five miles of Fermi 2. Explosions in facilities located further than five miles pose negligible hazard to the power plant.

Explosions as a result of transportation accidents involving heavy trucks or trains could be potentially damaging because of the proximity of the interstate highway and the railways. Additionally, the two quarries actively use explosives in their operations. Each quarry may have as much as 80,000 pounds of explosives on site to accomplish the day's shots.

Regulatory Guide 1.70 requires evaluation of accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels in facilities processing, transporting or storing such material. The effect of the explosions can potentially expose the nuclear power plant structures to blast over-pressure, dynamic pressure, blast-induced ground motion, or blast-generated missiles. Regulatory Guide 1.91 **[5.42]**, based on experimental data, demonstrated that the consequences of blast over-pressure on plant structures envelopes the other potential events. Explosions with the potential to produce an over-pressure on the order of magnitude of one psi or greater must be considered in the plant design basis. To assess the significance of a blast, the Regulatory Guide provided a relationship to conservatively estimate a safe blast distance, R, also known as stand-off distance.

The safe blast distance for the maximum estimated highway and railroad track hazardous cargo weight was calculated using the RG 1.91 relationship. The quarry hazard was evaluated conservatively assuming the simultaneous detonation of the maximum amount of explosives present on a given day. In all cases the required safe standoff distance was found to be significantly less than the actual distance from the hazard to the plant.

There is one potential significant explosion hazard cn site. Detroit Edison recently installed a system to inject hydrogen gas into the feedwater system and oxygen gas into the off-gas system. The system is currently in the testing stage.

This system, called the hydrogen water chemistry system, was installed to slow the rate of intergranular stress corrosion cracking (IGSCC) in the reactor piping and crack growth in the reactor internals. A 20,000 gallon liquid hydrogen storage tank is located west of the cooling towers, approximately 1100 feet northwest of the nearest Category I structure. Since rupture of the tank and subsequent release of hydrogen could result in a significant explosion, the siting of the tank considered the safe stand-off distance to assure that the Category I structures are not adversely affected by blast over-pressure. A summary of this evaluation was provided in a letter transmitted to Nuclear Mutual Limitea in March, 1995 **[5.43]**. This summary stated that the minimum safe stand-off distance for the hydrogen tank is approximately 800 feet which is less than the distance of approximately 1100 feet. Therefore, the installation of the hydrogen tank does not present an explosion hazard.

Additionally, the summary also described the transportation of hydrogen to the site to refill the storage tank. The evaluation concluded that the quantities of hydrogen transported will be significantly less than the total capacity of the storage tank and that the trucks will not pass closer to Category I structures than the aforementioned 1100 feet. Therefore, the transportation of hydrogen to the site does not present an explosion hazard. Additional Detail is provided in Detroit Edison Safety Evaluation 95-0024, Rev.1 **[5.55]**.

Flammable Vapor Clouds (delayed ignition)

There is no industry in the vicinity of the plant which can produce a flammable vapor cloud in significant amounts.

Although explosive materials can be shipped via the roadways and railroad, the nearest transportation routes are far enough from the plant site that delayed ignition of a vapor cloud can be ruled out as a potential hazard.

Toxic Chemicals

Regulatory Guide 1.78 **[5.44]** provides general design criteria to be considered in assessing the capability of the control room to withstand all postulated hazardous chemical releases on-site or in the surrounding area.

The Regulatory Guide anticipates toxic chemical spills may render the control room uninhabitable when:

- Large amounts of known toxic chemicals are spilled from stationary or mobile sources such as industrial facilities or transporters within five miles of the plant;
- Poisonous gases, as a result of a large toxic chemical spill or prolonged small leaks, under favorable wind conditions are expected to travel faster and penetrate the control room in high concentrations exceeding the maximum allowed exposure limits; and
- No detection system for known toxic chemicals (with possible poisonous fumes) is installed on-site for advance warning and automatic actuation of air isolation systems in the control room.

Table C-1 of the Regulatory Guide **[5.44]** provides some hazardous chemicals which have frequent industrial uses and could potentially be involved in accidental releases. The table provides basic human toxicity limits for each of the chemicals. This table along with other authoritative references for chemicals not presented in the table can be used to provide an assessment of the hazard to Fermi 2 associated with the transportation and storage of chemicals in the area around the site. This information can also be used to assess chemicals stored on the site.

Table C-2 of the Regulatory Guide **[5.44]** provides weights of hazardous chemicals, based on a 50 mg/mm^3 toxicity limit, and safe distances from the control room. From review of the Fermi 2 control room habitability position provided in Amendment 33 to the Final Safety Analysis Report (FSAR) **[5.47]**, the control room is best characterized as Type B.

Using Table C-2 the safe quantity/distance relationship for some of the more commonly stored or transported chemicals was evaluated. This evaluation demonstrated that significant quantities of the most commonly used hazardous chemicals, such as anhydrous ammonia, present insignificant risk to the Fermi 2 site when transported on the Interstate highway or railways 3.5 to 4.1 miles west of the plant.

Additional appreciation of the level of risk associated with accidental release of hazardous materials transported on the railways is gained through review of statistical data compiled by Nayak for the U.S. Department of Transportation **[5.45]**. From Nayak, the frequency of a railway accident involving a railcar of any type is 0.9 per 10,000,000 car miles. Four percent of all cars carry hazardous materials. Of the cars carrying hazardous materials involved in accidents, 16 percent resulted in release of at least a portion of their hazardous cargo. This translates into a probability of hazardous material release resulting from a rail car accident of approximately 5.8E-10 per car mile.

The hazard associated with this event could be further diminished for the Fermi 2 site by considering the occurrence frequency of prevailing winds which would carry the release plume toward the site.

The quantity of hazardous chemicals stored at the nearby industrial facilities can also be evaluated on the basis of Table C-2. It is clear that the Meijer Facility, which maintains approximately 22,000 pounds of anhydrous ammonia at an approximate distance of four miles from the Fermi 2 site, does not present a significant hazard to the site. Likewise, the water treatment facility and telephone substation facility store small quantities of hazardous material and are sufficiently far from the site.

Some toxic materials required for operation of the plant are stored on the Fermi 2 site. The materials, quantities, and locations are discussed in Fermi 2 Administrative Procedure NPP-EN1-01 **[5.46]**. The most toxic material in significant quantity is 4000 gallons of sulfuric acid. This tank is located east of the auxiliary boiler house near the lake. Though this chemical has very high toxicity, it also has low volatility. Low volatility combined with its location away from the control room and shielded by the turbine building assure that sulfuric acid does not present a significant risk to the plant.

Based on the separation distances between the transportation and industrial facilities and the site, accidental releases of toxic materials do not present a significant hazard to safe operation of Fermi 2.

Fires

There are no industrial or military facilities in the vicinity of the plant site that pose a fire hazard. Forest or brush fires do not pose any danger because the site has been cleared.

Based on the distance of the main transportation routes from Fermi 2, the potential fire hazard from a transportation accident in the vicinity of the power plant can be ruled out. Furthermore, the Fermi 2 control room can be isolated by placing the ventilation system in the recirculation mode upon initial smoke detection.

Aircraft Hazards

The aircraft hazard at a nuclear power plant is composed of three elements:

- The hazard associated with the proximity to an airport with significant departure and arrival traffic;
- The hazard associated with the proximity to military aircraft training routes; and
- The hazard associated with federal airways, holding patterns, or approach procedures which pass over the site.

The SRP considers that the occurrence frequency of an aircraft accident at a site which could result in unacceptable radiological consequences is less than approximately 1.0E-07 per year if the following requirements are met:

- The plant-to-airport distance, D, is between five and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2 ;
- The plant is at least five statute miles from the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 100 flights per year, or where activities (such as practice bombing) may create an unusual stress situation; and
- The plant is at least two statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern.

There are no military training flights which pass within five miles from the Fermi 2 site. However, based on the proximity of Fermi 2 to the Detroit Metropolitan Airport, with annual operations of approximately 480,000, and the number of federal airways which pass over the site, it is clear that the aircraft accident hazard cannot be screened by the above criteria.

The aircraft hazard was previously evaluated and documented in the Fermi 2 UFSAR Section 2.2. At that time the frequency of occurrence of a commercial aircraft crash at the site was estimated as 8.9E-08 per year and for a private aircraft about 8.9E-06 per year. The Category I reactor/auxiliary building was subsequently evaluated to demonstrate that the structure would withstand the impact of a small private aircraft without sustaining significant damage.

The following paragraphs summarize the review of the aircraft hazard at Fermi 2 using current air traffic data.

There are three low altitude and six high altitude federal airways which pass over the Fermi 2 site. These airways are described in Table 5-4. The FAA provided current traffic for these airways: 200 flights per day for the low altitude airways and 360 flights per day for the high altitude jet airways **[5.40]**. This information was used to calculate the probability of an in-flight crash using the hazard estimate methodology presented in the SRP. Two hazard estimate cases have been developed. The first case represents a conservative lower bound occurrence frequency and assumes that the total traffic occurs only on the airways which overlap the Fermi site. This case results in an estimated aircraft accident occurrence frequency of 1.7E-07 per year. The second case represents a more realistic estimate assuming that the total air traffic is evenly distributed over the low altitude airways and the high altitude airways. The estimated aircraft site accident occurrence frequency for this case is 7.1E-08 per year.

5.3.2 Plant Walkdowns

A plant walkdown was conducted at Fermi 2 on March 22 and March 23, 1995. The purpose of the plant walkdown was to assess the vulnerability of plant structures and equipment to nearby industrial, transportation, and military facility accidents and to confirm the location of nearby facilities and transportation routes. The plant walkdown included trips inside the protected area and plant structures and a general area survey via automobile.

In general, the objectives of the plant walkdown were to:

- Identify significant nearby industrial, transportation, and military facilities within a five-mile radius of the plant site; and
- Identify significant storage facilities within a five-mile radius of the plant site.

These objectives were fulfilled as discussed in the following paragraphs.

Nearby Industrial, Transportation, and Military Facilities

Prior to performing the plant walkdown, agencies such as the Monroe County Emergency Management Division, the Michigan Department of Transportation, the U.S. Army Corps of Engineers, the Grand Trunk and Conrail Railroads, and local quarry operators were contacted to identify facilities and transportation routes near the plant site. The objective of the walkdown was to assure the initial investigation was accurate and complete.

Several hours were spent driving the highways and local roads within five miles of the plant site. All of the major transportation routes were driven to locate the industries previously identified and to determine if any industries which may use hazardous chemicals existed in the vicinity. No additional industries were identified. The Grand Trunk and Conrail railways were also observed as part of the drive. No military facilities were observed within five miles of the plant site.

The walkdown observations confirmed the results of the initial investigation.

On-Site Storage Facilities

The walkdown of the Fermi site reviewed the location, quantity, and type for the hazardous materials which are listed in Fermi 2 Administrative Procedure NPP-EN1-01. The walkdown observations confirmed the results of the initial investigation.

5.3.3 Conclusions

As a result of the review documented in Sections 5.3.1 and 5.3.2, the Fermi 2 design has been shown to satisfy the requirements of the 1975 Standard Review Plan . This review focused on two principal elements of the SRP criteria:

- Identification of potential hazards in the site vicinity; and
- Evaluation of potential accidents.

All aspects of the Fermi 2 design with respect to nearby transportation, industrial, and military facility accidents satisfy the SRP criteria.

5.4 OTHER EVENTS

Section 2.0 of NUREG-1407 **[5.1]** identifies specific events evaluated for inclusion in the IPEEE program. Based on the evaluations conducted, the following five events were identified in NUREG-1407 for consideration by all licensees in the IPEEE: seismic events, internal fires, high winds and tornadoes, external floods, and transportation and nearby facility accidents. However, NUREG-1407 also requires that each individual licensee confirm that no plant unique external events known to the licensee with potential severe accident vulnerability are being excluded from the IPEEE. As part of the response to the IPEEE, Fermi 2 has performed a comprehensive screening of external events to assure that no unique events were excluded from the evaluation.

5.4.1 Screening Methodology

The methodology used to ensure that all significant external events relevant to Fermi 2 were evaluated is described below:

- The first step was to develop a complete listing of Fermi 2 external events based on the Updated Final Safety Analysis Report (UFSAR). This list of events provided the basis for the Fermi 2 external events evaluation. However, to confirm that no plant unique external events with potential severe accident vulnerability are excluded from the IPEEE review, other external events were reviewed based on the recommendations of NUREG/CR-2300 **[5.5]**.

- Second, each event was evaluated on the basis of an interim screening approach. The intent of this screening approach was to eliminate from further study those events with negligible contribution to the overall plant risk. The screening criteria were adopted from NUREG/CR-2300 Section 10 (specifically Section 10.3.1). The screening criteria are summarized as follows:

  Criterion 1: Low Frequency
  The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and will not result in worse consequences than these events. For example, meteorite impact as an external event can be eliminated on the basis of low frequency of occurrence.

  Criterion 2: Design Basis
  The event is of equal or lower damage potential than the events for which the plant has been designed. For example, Fermi 2 has been designed for a Design Basis Tornado with 300-mph wind velocity and tornado missiles; therefore, consideration of hail as a missile source is not necessary.

  Criterion 3: Relevance

The event cannot occur close enough to the plant to affect it. Fermi 2 is located in an area of insufficient seismic activity to generate tsunami activity on Lake Erie. Thus, a tsunami as an external event is eliminated from consideration.

Criterion 4: Inclusion
The event is included in the definition of another event. For example, release of toxic gases is included in the effects of nearby industrial, transportation, and military facility accidents.

Criterion 5: Speed
The event is slow in developing (e.g., drought) and there is sufficient time to eliminate the source of the threat or to provide an adequate response.

- Third, those events which are not screened using the above criteria are categorized under Criterion 6. This categorization indicates that the event is evaluated as part of the IPEEE in accordance with NUREG-1407.

Application of the above screening criteria resulted in the selection of a limited number of significant events for IPEEE consideration.

5.4.2 Results of Event Screening

The interim screening approach was used to screen each of the events in Table 5-5. The purpose of this screening was to confirm that there are no plant unique events which must be considered in addition to the aforementioned external events addressed by NUREG-1407 or to identify additional external events which require specific evaluation for the Fermi 2 plant site.

Based on the results of the interim screening, no unique events were identified for inclusion in the Fermi 2 IPEEE program. The basic events identified in NUREG-1407 for inclusion in the IPEEE program are evaluated in the Fermi 2 program. These basic events consider additional events based on similarity of subject matter (i.e., inclusion) as indicated below:

External Flooding - This accident category includes intense precipitation, storm surge, waves, and groundwater (insofar as in-leakage may occur).

High Winds and Tornadoes - This accident category includes missiles generated by natural phenomena (insofar as they may be induced by high winds and tornadoes).

Transportation and Nearby Facility Accidents - This accident category includes aircraft impact, fog, pipeline accidents, release of chemicals from storage on-site, toxic gas (i.e., exposure to hazardous chemical release), missiles generated by events near the site, explosions, and flammable vapor clouds.

These events were previously addressed in Sections 5.1, 5.2, and 5.3. All remaining events listed in Table 5-5 were screened by application of the interim screening criteria. The specific criteria used to screen each event is identified.

5.5 PLANT EXPERIENCE

Two actual external events have occurred at Fermi 2 leading to equipment malfunction. It was deemed appropriate to discuss them as part of this external event examination.

5.5.1 Loss of GSW Due to Low Lake Level

As noted in Table 5-5, low lake water level can be excluded from further consideration as an external event due to its inclusion in the design basis. The major impact of such an event, loss of general service water (GSW), is considered in the design basis of Fermi 2. However, since the event actually occurred once in 1989, and since loss of GSW does challenge other safety systems and would likely force a plant shutdown were it to occur at power, this event will be briefly discussed.

The event occurred on January 8, 1989, while the plant was in cold shutdown. GSW pumps were shut down over a period of about two hours due to lack of sufficient water depth in the intake canal. At the time there was a westward wind (compass direction about 250°) of about 25 mph. Recovery was initially established by cross-tying to the circulating water reservoir **[5.48]**.

Only small temperature rises were observed in the turbine and reactor building closed cooling water (TBCCW and RBCCW) systems. Compensatory fire watches were established to meet the Technical Specification one hour action statement. As stated in Section 2.2.3.1 of the Fermi 2 UFSAR, had the plant been at power when GSW was lost, there is about a 12 hour supply of water for at power conditions in the circulating water reservoir with sufficient reserve to accomplish normal shutdown. Moreover, for shutdown conditions, the ultimate heat sink is provided by the RHR reservoir in conformance with Regulatory Guide 1.27.

Investigation after the incident indicated silt buildup in the GSW inlet canal such that elevation of the canal bottom was approximately 568 ft. (above sea level). This level was approached as a result of the strong west wind leading to the observed loss of GSW.

Even though loss of GSW is within the design basis, it is obviously an undesirable event for the reasons stated above. Following the 1989 event, several corrective actions were taken to help prevent recurrence and to minimize its impact should it reoccur. These are listed below **[5.48]**:

- The low level alarm set-point was set at a higher elevation to trigger earlier operator response (such as transfer of GSW suction to the circulating water reservoir).

- The abnormal operating procedure "Loss of GSW System" was revised to incorporate a new section for loss of GSW pump suction including direction to transfer to the circulating water reservoir **[5.49]**.

- The excess silt observed following the event was removed by dredging the intake canal.

- An annual intake silt inspection has been included in the Fermi 2 Performance Scheduling and Tracking system **[5.50]**. If canal depth is not adequate, dredging is to be performed.

5.5.2 Loss of Diesel Generator Cooling Water Due to Ice Formation

A very recent event (February 6, 1996) occurred where there was a potential for a common cause failure of diesel generator cooling function due to ice formation in a diesel generator service water pump column. This event is the subject of Licensing Event Report No. 96-001 **[5.51]** and is still in the review and evaluation stage.

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Table 5-1 Comparison of Reg. Guide 1.76 and the Fermi 2 Design Basis Tornado

Tornado	Rotational Speed (mph)	Transitional Speed (mph)		Maximum Wind Speed (mph) (for design)	Radius of Maximum Rotational Speed (ft)	Pressure Drop (30)	Rate of Pressure Drop (psi/sec)
		Maximum	Minimum				
Fermi 2 Design Basis	300	60	N/A[1]	360	227	3.0	N/A
Reg. Guide 1.76 (Region I)	290	70	5	360	150	3.0	2.0

[1] This parameter has not been specified.

Table 5-2 Design Wind Pressure

Design	Design Wind Speed	Total Design Wind Pressure (Combined Windward and Leeward)			
		Height Above Ground (z)			
		<50 ft	50 ft ≤ z < 100 ft	100 ft ≤ z < 150 ft	150 ft ≤ z < 200 ft
Fermi 2 (1)	90 mph	35.5 psf	54.3 psf	66.1 psf	74.5 psf
SRP Criteria (2)	80 mph	29.1 psf	34.1 psf	38.0 psf	39.7 psf
SRP Criteria (2)	90 mph	36.9 psf	43.2 psf	48.1 psf	50.3 psf

Notes:

(1) From UFSAR Table 3.3-1. Utilizes 90-mph basic wind speed, 1.1 gust factor, and velocity distribution with height typically associated with the Atlantic and Gulf coastal regions.

(2) Using $C_p = 1.3$, K_z and G_h are per Table 6 and 8, respectively, of ANSI A58.1 for Exposure C.

Table 5-3 Fermi 2 Design Basis Tornado Missiles For Category I Structures And SRP Required Missiles

Design	Missile	Dimension	Weight (lbs.)	Velocity (Fraction of Max. Wind Velocity)	Height (ft. above ground)
Fermi 2 (General)	Wooden Plank	4" x 12" x 12' long	160	0.71	Any
Fermi 2 (General)	Automobile	25 sq. ft. contact area	4000	0.14	0 to 25
Fermi 2 (RHR Cooling Towers)	Utility Pole	13.5" dia x 35" long	1490	0.4	0 to 30
Fermi 2 (General)	Steel Rod	1" dia x 3' long	8	0.6	Any
SRP Missile (C)	Steel Rod	1" dia. x 3' long	8	0.6	Any
SRP Missile (F)	Utility Pole	13.5" dia. x 35' long	1490	0.4	0 to 30

Table 5-4 Federal Airways In The Vicinity Of The Fermi Site

Airway	Width (Nautical Miles)	Approximate Centerline Distance to Site (Nautical Miles)	Approximate Distance of Airway Edge to Site (Nautical Miles)
Low Altitude			
V10-188	8	3	0
V26-133	8	4	0
V493	8	10	6
High Altitude			
J34	8	10	6
J43	8	20	16
J146	8	30	26
J190-J584	8	0	0
J554	8	10	6
J586	8	20	16

Table 5-5 Results of External Event Screening

Screening Criterion	External Events Screened
Criterion 1: Low Frequency	Meteorite, Turbine-Generated Missiles
Criterion 2: Design Basis	Coastal Erosion, Forest Fire, Frost, Hail, High Summer Temperature/Low Winter Temperature, Ice Cover, Landslide, Lightning (Severe Weather Phenomenon), Low Water Considerations (River Water Level), Seiche, Snow
Criterion 3: Relevance	Avalanche, Dam Failure, Hurricanes, River and Channel Diversion, Sandstorm, Tsunami, Volcanic Activity
Criterion 4: Inclusion	Aircraft Impact, Fog, High River Stage, Pipeline Accidents (Gas, etc.), Intense Precipitation, Release of Chemicals from Storage On-site, Storm Surge, Toxic Gas (Exposure to Hazardous Chemical Release), Waves, Missiles Generated by Natural Phenomena, Missiles Generated by Events Near the Site, Probable Maximum Flood on Streams and Rivers, Explosions, Flammable Vapor Clouds, Groundwater
Criterion 5: Speed	Drought
Criterion 6: Requires Review	External Flooding; High Winds and Tornadoes; Nearby Industrial, Transportation, and Military Facility Accidents

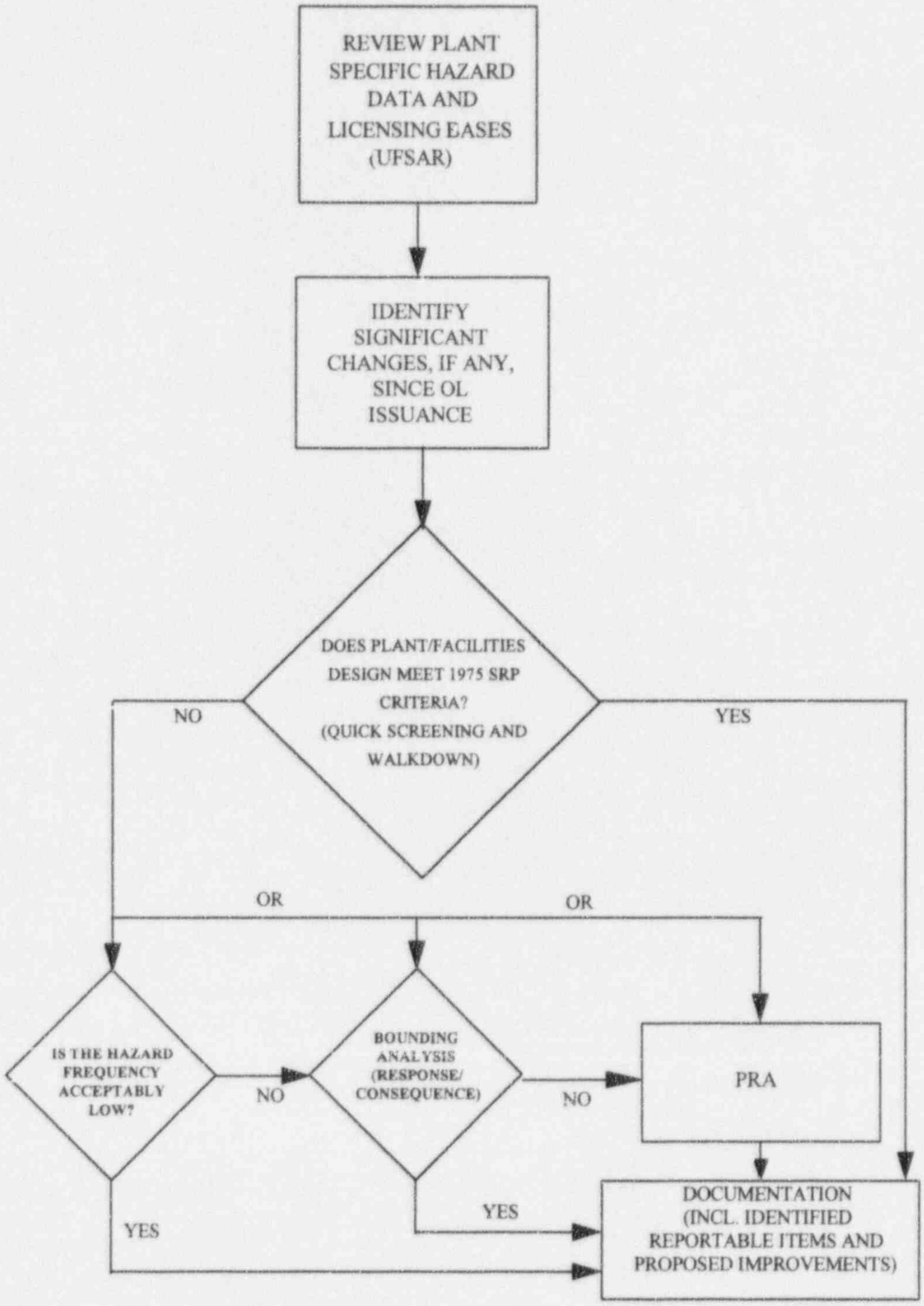

Figure 5-1 NUREG-1407 Progressive Screening Approach

SECTION 6

LICENSEE PARTICIPATION AND INDEPENDENT REVIEW

6.0 INTRODUCTION

This section describes the Fermi 2 staff involvement in the IPEEE process and the nature of the independent review. This participation structure was utilized to maximize the benefit of the IPEEE and assure the quality of the product.

6.1 IPEEE PROGRAM ORGANIZATION

The Fermi 2 IPEEE effort has been a joint utility-consultant effort managed by the Fermi 2 Nuclear Organization. Unlike the IPE effort, the major portion of the technical effort did not reside in the Risk Analysis group since the PSA approach was not used as the evaluation tool for the fire and seismic examination. Thus, while the overall IPEEE Project Manager is in the PSA group, two major task managers were used to head the activities relating to the fire and to the seismic and other external event examinations.

Project Direction

Name	Discipline	Organization	IPEEE Task
Earl Page	Risk Analyst	Licensing/Risk Analysis	Project Manager
A. I. Hassoun	Civil Engineer	Plant Support Eng./ Mechanical-Civil	Task Manager, Seismic
			Task Manager, "Other" Events
M. McDonough	Fire Protection Engineer	Plant Support Eng./ Electrical	Task Manager, Fire

Utility and associated contractor participation in the IPEEE activity is best described by individually discussing the three major activity areas: seismic, fire, and "other" external events (high winds, floods, and transportation and nearby facility accidents).

Seismic Events:

The major part of the seismic examination was carried out by Detroit Edison personnel of the Fermi 2 seismic review team (SRT) supplemented by a few contractor personnel, under the direction of the Seismic Task Manager.

Two of the Fermi 2 personnel were involved in the system analysis; others participated in the walkdowns and seismic evaluation. The contractor personnel supported both the walkdowns and selected portions of the seismic evaluation. See Section 3.1.4.2 for more detail on SRT qualifications and assignments. In addition, Dr. R. P. Kennedy, a senior consultant, was involved in several areas of the seismic evaluation, including a scoping walkdown, consultation on the generation of new review level earthquake (RLE) response spectra, and providing the special study on reactor internals.

The overall methodology employed for the seismic evaluation, general conclusions and insights were presented to selected middle Fermi 2 managers in January, 1996, and to senior Fermi 2 management in February, 1996.

Personnel involvement in the independent review process is discussed in Section 6.2.

Fire Events:

The overall fire examination was carried out under the general direction of the Fire Task Manager. Compilation of 10CFR50, Appendix R documentation and extensive cabling routing, and equipment location tasks required for the FIVE methodology were performed by Fermi 2 personnel including the Task Manager. The majority of the FIVE methodology evaluation was performed by contractors with ERIN Engineering evaluating the Auxiliary Building and VECTRA the remainder of the plant. Both utilized a single software pack ige that automated the EPRI FIVE methodology. The extensive PSA analysis required to evaluate the availability of non-fire damaged equipment was conducted by Mr. Dennis Jondle, a Fermi 2 PSA engineer, with support by PLG personnel brought on-site. The FIVE confirmatory walkdowns were conducted by ERIN Engineering and VECTRA personnel guided by the Fermi 2 Fire Task Manager. Supplemental walkdowns principally conducted to verify transient combustible and equipment locations in the reactor building were conducted by Fermi 2 personnel. Treatment of the Sandia Fire Risk Scoping Study Issues were addressed by an onsite contractor under the direction of the Task Manager. Treatment of the related generic safety issues was performed largely by on-site contractor personnel under the direction of the Fire Task Manager.

The overall methodology employed for the fire evaluation, general conclusions, and insights were presented to selected Fermi 2 middle managers in January 1996, and to senior management in February, 1996.

Personnel involvement in the independent review process is discussed in Section 6.2.

"Other" External Events:

Evaluation of high winds, floods, and other events was conducted by VECTRA Technologies, under the direction of the Other Events Task Manager. Supplemental review of a 1989 low level lake event leading to loss of General Service Water (GSW) was provided by Fermi 2 personnel.

The overall methodology employed for the "other" events, general conclusions, and insights were presented to senior management in February, 1996.

Personnel involvement in the independent review process is discussed in Section 6.2.

6.2 INDEPENDENT REVIEW - PARTICIPATION, SCOPE, COMMENTS, AND RESOLUTION

The IPEEE activity included several tiers of independent review by both in-house and contractor personnel and consultants. This review activity, including participants, scope, comments, and comment resolution, is best described individually for each of the three major classes of external events: seismic, fire, and "other" (high winds, floods, transportation and nearby facility accidents). The following subsections describe this review activity.

6.2.1 Seismic Review

- A major seismic peer review was conducted by Dr. John Stevenson of Stevenson & Associates. A sampling approach was used by selecting 16 screening and evaluation worksheets (SEWS) representing a variety of mechanical and electrical components at different plant locations. The evaluations and the relevant assumptions were reviewed; following the analytical review, a confirmatory walkdown was made covering the same components. In addition, general observations were made. Results of this review are documented in Dr. Stevenson's summary letter **[6.1].** All comments have been satisfactorily dispositioned. **[6.2].**

  These peer review comments covered the seismic evaluation process as well as observations made during the plant walkdown. A sampling of the technical comments and their disposition is given below:

  Comment: It was recommended that specified minimum concrete strength be used in the calculations unless justification for higher concrete strength can be provided.
  Disposition: The use of "higher than nominal" concrete strength has been justified. The justification was based on actual results of concrete sample cylinder testing of Fermi 2 concrete pours at different ages.

  Comment: If shock isolators are cast iron and they bottom out as indicated in the evaluation, what assurance is there that they will not rupture?
  Disposition: An evaluation demonstrated that the isolater elements, upon impact, would be subject to a compression stress much lower than the allowable stress for the cast iron material used; therefore, it was concluded that the isolater would not fail.

  Comment: Recommend using the tensile and shear capacity of expansion anchor bolts evaluation as given in the "SQUG-GIP," Appendix C. If more aggressive factors of safety are used, justification should be provided.
  Disposition: For the general anchorage evaluation performed in DC-5634, a factor of safety equal to 2.0 was used for anchor shear capacity as recommended in

EPRI NP-6041, Table O-2 for Conservative Deterministic Failure Margin (CDMF) evaluations. For tensile capacity, a factor of safety equal to 3.0 was generally used. This factor is conservative since it corresponds to a single bolt anchorage with "hairline crack unlikely" in Table O-2. The recommended factor of safety for the same crack condition with two or more bolts is 2.8. This latter factor was only used in a few exceptional cases. The Seismic Qualification Utility Group (SGUG), General Implementation Procedure (GIP), Appendix C expansion anchor allowable loads were also used in some special evaluations, since they are generally more conservative than the ones in EPRI NP-6041. However, for the IPEEE program, the EPRI document provides the main guidelines and evaluation criteria; therefore, it was used at Fermi 2.

Comment: It should be confirmed that the voltage regulator is installed in the plant in the same manner as during seismic testing.
Disposition: A note has been added to the pertinent SEWS addressing the similarity between field mounting and seismic test mounting configurations.

Comment: During review of the switchgear room cooling units it was noted that, according to the American Institute of Steel Construction, evaluation of bolt bending need not be considered unless the gap (shims) is greater than five times the bolt diameter.
Disposition: Although evaluation of bolt bending may not be required, it is conservatively retained.

Comment: It was noted that the HPCI stop and control valves were evaluated for an acceleration of 1.5g. Confirm that 1.5g envelops the spectral loading.
Disposition: The applicable spectral accelerations (base mat of building) are well below 1.5g.

Comment: There is a large air dryer orange tank located near safe shutdown components on the second floor of the reactor building that appears to have marginal anchorage. There may be an interaction concern due to the nearby safe shutdown components.
Disposition: This item has been added to the list of anomalies, and a Technical Service Request initiated to address the anchorage.

- Selection of the two safe shutdown paths was reviewed and endorsed by the IPEEE Project Manager. The basis of this endorsement was consistency with the functional requirements and success criteria utilized in the PRA for the Fermi 2 IPE.

- Mr. Paul Hayes of MPR Associates performed an independent assessment of the preliminary Fermi 2 Success Path Logic Diagram and the associated Safe Shutdown Equipment List (SSEL). Detroit Edison resolved the resulting review comments **[6.11]** by incorporating them in the process of finalizing the SSEL or by providing the appropriate disposition **[6.12]**.

Typical among the MPR comments are (1) recommendation to check the Success Path Logic Diagram against the Appendix R safe shutdown analysis report, (2) a confirmation be made that offsite power is not needed for the electrically powered equipment on the SSEL, and (3) that the IPE dependency tables be used in identifying SSEL dependencies or support systems. These recommendations were all implemented in the final preparation efforts for the SSEL.

- Mr. Jess Betlack of MPR participated with members of the SRT and others to perform an assessment of the IPEEE seismic relay evaluation plan and conducted sample reviews of equipment on the preliminary SSEL **[6.13]**. The conclusion was reached that the proposed approach for identifying low ruggedness relays complied with the intent of the program.

- Dr. Robert P. Kennedy, senior seismic consultant, provided an independent review of the seismic capacity (HCLPF) calculations for the masonry and shield walls. Dr. Kennedy endorsed the final results after his comments were incorporated in the analysis.

- Mr. Steve Reichle of VECTRA Technologies performed an independent review of the seismic containment performance evaluation plan **[6.14]**. The review concluded that the process was essentially consistent with the NUREG-1407 guidance and with the approach used by other nuclear facilities.

  A principal recommendation from Mr. Reichle's review was to include the torus-to-drywell vacuum breakers on the SSEL. This recommendation was incorporated in the final SSEL.

- The safe shutdown equipment list was reviewed to assure that the success paths systems are consistent with plant procedures and the components and instrumentation selected are sufficient for successful operation of the systems contained in the safe shutdown paths. The review was conducted by a Shift Technical Advisor (STA) who consulted licensed operators and other STAs as needed. Details of the Operations review efforts and comments are documented in Section 3.1.2.2.5. A major comment dealt with the need to add several residual heat removal system valves to the SSEL that had mistakenly been identified as passive valves. All comments have been dispositioned and concurred with by the reviewer **[6.4, 6.5, 6.10]**. It should be noted that Plant Support Engineering, I&C, augmented the Operations review to assure that adequate support was available for the instrumentation that was selected. In some cases this led to the addition of I&C components and the substitution of instruments when the initial selection would have required offsite power. Additional discussion of this review is given in Section 3.1.2.2.5.

- Mr. Charbel Abou-Jaoude of VECTRA Technologies conducted a review of the Fermi 2 IPEEE seismic evaluation program when the program was about 60% complete. The review was intended to be an intermediate partial peer review. The review included a walkdown of selected plant areas. This review concluded that the Fermi 2 seismic walkdowns that had been conducted to date were performed in a very thorough and competent manner. The review also noted that Fermi 2 has seismically rugged structures, systems, and equipment compared to other plants **[6.15]**.

6.2.2 Fire Review

- As described in Section 6.1, the major portion of the FIVE methodology evaluation was performed by ERIN Engineering and VECTRA Technologies, under the direction of the Fire Task Manager. An independent review of these two contractor evaluations was conducted by Mr. Richard Anderson of Plant Support Engineering, Electrical. The review focused on accurate equipment location and assuring the correct fire impact on electrical support to systems.

  More than 100 comments were generated and documented **[6.6, 6.7]**. Some comments dealt with clarity of the documentation. Several comments alluded to equipment location errors, particularly in the RHR complex as a result of redefining fire compartments in that building. A significant comment identified additional failures as a result of a postulated specific panel fire in the control room that increased the impact of that fire scenario. Comments were directly communicated to the contractor, dispositioned, and incorporated in this evaluation as appropriate. A summary of the comment disposition for the ERIN Engineering evaluation is contained in reference **[6.8]**. A summary of comment disposition for the VECTRA evaluation is contained in reference **[6.9]**.

6.2.3 Other External Events Review

- The analysis of high winds, floods, and transportation and nearby facility accidents performed by VECTRA Technologies was independently reviewed by Mr. Albert Burg of the Mechanical and Civil group in Plant Support Engineering. Comments were dispositioned during a meeting of the reviewer and the Fermi 2 "Other" Events Task Manager with the initial evaluators from VECTRA. The IPEEE report draft prepared by VECTRA was reviewed by the same Detroit Edison reviewer and by the IPEEE Project Manager. Among the comments made was the addition of some discussion on the 1989 loss of GSW due to low lake level and the acknowledgment of a recent partial loss of diesel service water during a period of low ambient temperatures.

6.3 REFERENCES

6.1 John D. Stevenson, Letter to S. Hassoun, 96C1910A Hassoun 2.13, Stevenson & Associates, February 19, 1996.

6.2 A. I. Hassoun Letter to K. E. Howard, "Resolution of the Seismic IPEEE Peer Review Comments," TMPE-96-0103, March 8, 1996.

6.3 Dale Hoskins, Memo to Dennis Jondle, "Preliminary Operations Report on the Seismic Margin Assessment," May 24, 1995.

6.4 Dennis Jondle, Memo to Dale Hoskins, "Disposition of Recommendations Obtained from the Operation's Review of the Seismic IPEEE SSEL," TMLR-95-0020, July 17, 1995.

6.5 D. P. Ockerman, Memo to J. G. Walker, "Operations' Review of the Safe Shutdown Equipment List (SSEL)," NPOP-95-0038, July 17, 1995.

6.6 R. C. Anderson, Memo to M. B. McDonough, "IPEEE Peer Review Comments - ERIN Draft 'FIVE' Report," TMPE-96-0067, February 19, 1996.

6.7 R. C. Anderson, Memo to M. B. McDonough, "IPEEE Peer Review Comments - VECTRA Draft 'FIVE' Report," TMPE-96-0066, February 19, 1996.

6.8 V. M. Anderson, Letter to Mark McDonough, "IPEEE NRC Submittal Documentation of Fermi 2 Auxiliary Building FIVE Analysis (2nd Revision)," C1429501-2487/2, January 11, 1996.

6.9 J. N. Amason, Letter to R. C. Anderson, "Response to Comments Originating from Peer Reviews Dated December 15, 1995, January 3, 1996, and January 4, 1996 for VECTRA Draft Reports 0102-00063.R01 Rev. A and 0102-00063.R02 Rev. A," CN:0102-00063.079, January 11, 1996.

6.10 A. I. Hassoun, Memo to D. R. Hoskins, "Operations Review of Seismic IPEEE SSEL," TMPE-95-0448, September 28, 1995.

6.11 "Fermi 2 IPEEE Safe Shutdown Equipment List Review," MPR Associates report, December 31, 1992. (Edison File No. P1-15432)

6.12 D. D. Jondle, Memo to A. I. Hassoun, "Seismic IPEEE - Disposition of MPR Comments on Draft SSEL," TMLR-93-0007, April 19. 1993.

6.13 Jess Betlack, Letter to Earl Page, "Fermi 2 IPEEE Relay Evaluation," December 11, 1992.

6.14 Steve Reichle, Letter to Sam Hassoun, "Containment Evaluation Review for Fermi Seismic IPEEE," 0102-00058.000, November 18, 1994.

6.15 Charbel M. Abou-Jaoude, Letter to A. I. Hassoun, "IPEEE Seismic Reviews;" 0102-00058.000-00002 LTR 01, April 24, 1995.

SECTION 7

UNIQUE SAFETY FEATURES AND PLANT IMPROVEMENTS

7.0 INTRODUCTION

This section describes safety features unique to Fermi 2 and plant improvements that have been implemented or are to be considered as a consequence of the findings made and insights gained through the IPEEE process.

7.1 FERMI 2 UNIQUE SAFETY FEATURES

The NSSS design of Fermi 2 is a BWR 4 with an inerted Mark I containment. However, the plant has several unique features which influence its safety performance relative to other NRC reference plants. This subsection identifies these features and, where possible, identifies the estimated magnitude of their impact (positive or negative) on the overall core damage frequency as cited in the Fermi 2 IPE Report.

7.1.1 Plant Unique Initiating Events

The most significant plant unique feature related to initiating events involves the divisional offsite power supply system. Unlike many other plants, Fermi 2 has an offsite power distribution system which is entirely independent and divisional. Division 1 is supplied by three 120 kV offsite lines through a switchyard located at the decommissioned Fermi 1 site. Division 2 is supplied by two 345 kV offsite lines connected to a switchyard located adjacent to Fermi 2. During power operations the two divisions are totally independent, and the only commonality between the two divisions is the right-of-way leaving the Fermi site.

This offsite distribution system has two counteracting impacts with respect to risk. On one hand, the design decreases the likelihood of the plant losing all offsite power. On the other hand, each division is susceptible to one or more single failures which in some cases are difficult to restore. For example, in some plants the failure of a single transformer may lead to loss of AC power to all safety buses. However, this transformer is usually backed up by an automatic or manual transfer from a primary to a reserve feed. If the transfer is successful it allows rapid restoration of offsite power, sometime even without causing a plant trip. At Fermi 2, failure of a single transformer leads to loss of AC power to only a single division; the other division is unaffected. Thus, while some other plants can experience an interruption of offsite power due to failure of a single transformer, Fermi 2 cannot. The only significant contributors to loss of all offsite power at Fermi 2 are grid related or severe weather related events impacting either the right-of-way or a large fraction of the Detroit Edison grid. However, in Fermi's case, for divisional power losses there is often not a backup transformer to provide an alternate feed of offsite power; therefore, recovery of offsite power to that division is impaired.

This separate switchyard configuration results in different contributors to core damage frequency than some past BWR PRAs have found. For example, since the frequency of loss of all offsite power is lower than other sites, the total contribution from station blackout events is reduced. On the other hand, since the divisional events lead to a loss of condenser and some BOP components, the contribution to sequences involving loss of makeup and loss of heat removal is somewhat increased.

7.1.2 Plant Unique Systems

Fermi 2 has several plant unique system design features including the standby feedwater system, availability of an on-site blackstart combustion turbine generator, and four emergency diesel generators (EDGs) with some intra-divisional cross-tie capability. These features all contribute to reducing the total core damage frequency for Fermi 2.

Standby Feedwater System - The standby feedwater system (SBFW) at Fermi 2 is a non-safety related, motor-driven, two-train system which can provide up to 600 gpm per train of high pressure coolant makeup to the RPV from the condensate storage tank. For most transients resulting in shutdown, one of two trains is adequate for coolant inventory requirements. The system is manually initiated by the operators from the control room and is the preferred means of coolant makeup in the event of loss of normal feedwater. A sensitivity case performed using the IPE model found that the presence of the standby feedwater system results in a core damage frequency reduction of more than a factor of 5.

On-Site Blackstart CTG - Fermi 2 has four on-site combustion turbine generators (CTGs) which can provide power to the Division 1 electrical buses (18.8 Mw per generator). One of these generators (CTG 11-1) has blackstart capability and is credited in Fermi's station blackout coping analysis as an alternate AC power source. In the event of loss of all offsite power, tests have found that the operators can start CTG 11-1 and connect it to safety related buses within 20 minutes. Once started, CTG 11-1 has enough capacity to pick up all BOP and safety related loads powered from Division 1 supplies. If desired, the other units could then be started. A sensitivity case performed using the IPE model found that the presence of CTG 11-1 results in a core damage frequency reduction of roughly 37 percent.

Four EDGs with Intra-divisional Cross-ties - The Fermi 2 electrical distribution system consists of four safety related 4 kV buses, each with its own emergency diesel generator (EDG). Power distribution is divided into two redundant divisions with two EDGs per division. Each division can provide power to necessary shutdown cooling and control power systems to ensure long term operation and control. In addition, if an EDG fails, the 480V buses in the same division can be cross connected to provide power to necessary low voltage plant loads, and by taking actions such as interlock defeats, the 4160V buses can also be cross connected.

Additional AC Bus Cross-tie Capability - Many AC power buses have proceduralized cross-tie capability to another bus, either within the division or cross-divisional. For example, most BOP buses including circulating water (CW), the general service water (GSW) and standby feedwater (SBFW) AC power busses can be cross-connected to provide power from either offsite source of BOP power. This increases the likelihood of restoring the main condenser following a long term loss of either source of offsite power. In addition, the 4 kV ESF buses can be cross-connected through a maintenance crosstie to provide power from one offsite power feed to the other division. However, there are administrative controls that preclude this cross-tie during normal power operations.

The IPE model found that the presence of the three interdivisional bus cross-ties (GSW, SBFW, maintenance) plus the condenser restoration capability is responsible for a core damage frequency reduction of about one-third.

7.1.3 Plant Unique Sequences and Operator Actions

Fermi 2 utilizes Emergency Operating Procedures (EOPs) based on Revision 4 of the BWROG Emergency Procedure Guidelines (EPGs). The only unique operator actions called for in the EOPs involve the use of a plant unique system as part of the EOPs (i.e., Standby Feedwater). The human reliability analysis of these plant unique actions performed for the IPE did not identify any particular procedural weaknesses.

The core damage sequences quantified in the Fermi 2 IPE did not identify any unique plant susceptibilities other than those associated with loss of divisional off-site power as discussed in Section 7.1.1.

One unique result from the Fermi 2 IPE is the low frequency of station blackout core damage sequences. Some past BWR PRAs have found station blackout to be a large or even dominant contributor to core damage frequency. This is not the case for Fermi 2. Station blackout contributes only about 2 percent to the total core damage frequency. This is primarily due to two factors:

- Four standby EDGs
- CTG 11-1 as alternate AC power supply for Division 1.

Each of these Fermi 2 features leads to a reduction in the likelihood of sustained loss of all AC power.

An additional feature somewhat unique to Fermi 2 is the use of preplanned, prepared EOP packets for implementing EOP actions. Each EOP action requiring plant operators to perform a unique or unusual action such as installing electrical jumpers is described in an attachment to the EOPs. As appropriate, each of these attachments has a packet located in a locked file cabinet in the shift supervisors office which contains all equipment necessary to perform the action.

These EOP packets are controlled by Operations and audited regularly to ensure they are complete and consistent with the current revision of the EOPs **[7.1]**.

7.2 PLANT IMPROVEMENTS

This section describes improvements to be made to the plant as a result of the IPEEE. These are discussed below for each major external event grouping. Where appropriate, a brief description of the particular insight gained that led to the planned improvement is also included.

7.2.1 Seismic Events

As a result of the seismic evaluations discussed in Section 3, the SRT found that in general plant components are securely mounted and in compliance with design configuration drawings. The SRT found relatively few conditions which were not in conformance with plant drawings and which merited corrective action through Work Requests. Such corrective actions to restore the affected plant components to their original configurations are not considered plant improvements. Only those modifications or proposed activities that extend beyond the design configuration or current practice are included below as plant improvements. These improvements are discussed below:

- Several adjacent panels containing relays are not bolted together. These panels are located in the relay room, switchgear rooms, and RHR Division 2 switchgear rooms. Banging of these panels during a seismic event may cause contact chatter in sensitive relays mounted in the panels. Provisions for fastening these panels together have been designed and are scheduled for implementation by the end of the 1996 fall refueling outage (RFO5). Additional discussion is provided in Section 3.

- Four low-ruggedness relays used in the emergency diesel generator voltage sensing circuits are to be replaced upon selection of a suitable replacement.

- The anchorage for a large non-safety related air dryer tank on the second floor of the reactor building is not robust. Since there are safe shutdown components nearby, the need for additional seismic restraint is to be evaluated.

- A weakness in the seismic load path was identified for two large CCHVAC instrumentation panels on the fifth floor of the auxiliary building. This items is to be evaluated for resolution.

- A large fraction of the mounting hardware deficiencies found were believed to be associated with maintenance activities rather than original installations. Additional training will be incorporated in the continuing maintenance training program to increase the awareness level and emphasize the importance of mounting hardware installation and restoration during and after maintenance activities. Training is planned for completion by the second quarter of 1996.

- Operations training does not include a sustained loss of offsite power and CTG 11-1 scenario as may result from a severe seismic event. Current simulator training assumes CTG 11-1 is restorable within the first 30 to 60 minutes after a loss of offsite power. Also, during a severe seismic event, it is expected that many spurious alarms could be received in the control room due to low seismic ruggedness relay chatter. Although this may not have a direct effect on safe plant shutdown, it may cause some confusion in the control room. These two features will be included in the seismic simulator training event to be incorporated into the operator training program by the end of 1996.

7.2.2 Fire Events

One of the six unscreened fire compartments is the second floor of the reactor building. While its core damage frequency is on the borderline for screening, it was noted that the dominant risk contributors were assumed fires in the dedicated shutdown related cabinets. Loss of these cabinets isolate the affected equipment from the main control room thereby causing loss of the equipment function. While this loss potential is adequately covered by current operator training, additional Fire Brigade drills in the vicinity of these cabinets are planned to increase the awareness of the brigade members of the need to quickly isolate and extinguish such cabinet fires.

7.2.3 Other External Events

There were no plant improvements identified by this IPEEE for protection against high winds and tornadoes, external floods, and transportation and nearby facility accidents. Lessons learned from a 1989 incident involving the loss of GSW water intake had previously resulted in several preventive and mitigative measures to help prevent and/or mitigate recurrence. More details are provided in Section 5.5.1.

7.3 REFERENCES

7.1 "Performance of Audits," Operations Conduct Manual, Chapter 3 - Policies and Practices, Revision 1, January 29, 1996 (See Operations Audit Forms 18, 19, 20, 21).

SECTION 8

SUMMARY AND CONCLUSIONS
(Including Proposed Resolution of USIs and GIs)

8.0 INTRODUCTION

This section briefly summaries the conclusions of this IPEEE and includes the overall result of the evaluation with regard to degree of protection against external events, corrective actions taken or planned, and insights gained together with any associated plant improvements. This summary is presented separately for each of the major external event categories.

Also included is a discussion of the intended disposition of the related ongoing programs described in NUREG-1407 and Supplement 4 to Generic Letter 88-20.

8.1 SEISMIC EVENT SUMMARY

With the completion of the plant modifications and corrective maintenance activities discussed below, all outliers identified during the seismic evaluation and walkdowns are shown to have adequate capability to withstand the prescribed Review Level Earthquake without degradation of the components or pertinent systems. As a result, this study has demonstrated, by using the above-described methodology, that the plant seismic HCLPF at Fermi 2 is equal to or greater than 0.3g. While no significant seismic vulnerabilities were identified, there were several observations made and insights gained that led to corrective action and planned plant improvements.

As a result of the seismic evaluation, the seismic review team (SRT) found that plant equipment is securely mounted and in compliance with the design configuration drawings. However, for some components, minor deviations were noted which mostly involved missing or damaged mounting hardware. These deviations were addressed by initiating maintenance work requests to correct the anomalies. A summary of these work requests is given in Table 3-7. Most of them have been completed; the remainder will be completed by the end of the fall 1996 refueling outage (RFO5). These work requests are being tracked through the Deviation and Corrective Action program by DER 94-0644 **[8.1]**.

Several plant improvements were identified. Four are modest hardware changes. Two involve additional training. These plant improvements are summarized below:

- Several adjacent panels containing relays are not bolted together. These panels are located in the relay room, switchgear rooms, and RHR Division 2 switchgear rooms. Banging of these panels during a seismic event may cause contact chatter in sensitive relays mounted in the panels. Provisions for fastening these panels together have been made and are scheduled for implementation by the end of the 1996 Fall refueling outage (RFO5). The design is documented in an approved Engineering Design Package (EDP-27108) **[8.2]** and tracked through DER 94-0644 **[8.1]**.

- Four low-ruggedness relays used in the emergency diesel generator voltage sensing circuits are to be replaced upon selection of a suitable replacement. This planned change is documented in Technical Service Request (TSR 27566) **[8.3]** and tracked through DER 95-0104 **[8.9]**.

- The anchorage for a large non-safety related air dryer tank on the second floor of the reactor building is not robust. Since there are safe shutdown path components in the vicinity, installation of additional seismic restraints will be evaluated. This potential change is documented in TSR 28195 **[8.4]** and tracked through DER 94-0644 **[8.1]**.

- A weakness in the seismic load path was identified for two large CCHVAC instrumentation panels on the fifth floor of the auxiliary building. DER 96-0289 **[8.11]** was initiated to treat the resolution of this issue and the implementation of any necessary improvements.

- A large fraction of the mounting hardware deficiencies found were believed to be associated with maintenance activities rather than original installations. Therefore, additional training will be incorporated in the continuing maintenance training program to increase the awareness level and emphasize the importance of mounting hardware installation and restoration during and after maintenance activities. Training is planned for completion by the second quarter of 1996. This training activity is being initiated through a Training Work Request **[8.5]**.

- Operations training does not include a loss of offsite power and permanent loss of CTG 11 Unit 1 (CTG 11-1) scenario as may result from a severe seismic event. Current simulator training assumes CTG 11-1 is restorable within the first 30 to 60 minutes after a loss of offsite power. Also, during a severe seismic event, it is expected that many spurious alarms could be received in the control room due to low seismic ruggedness relay chatter. Although this may not have a direct effect on safe plant shutdown, it may cause some confusion in the control room. These two features will be included in the seismic simulator training event to be incorporated into the operator training program by the end of 1996. This training activity is being initiated through a Training Work Request **[8.6]**.

8.2 FIRE EVENT SUMMARY

The progressive screening process employed in the FIVE methodology led to six fire compartments that did not meet the screening criterion of less than 1.0E-6/yr core damage frequency (CDF). Since the screening criterion was only modestly exceeded (largest computed CDF was 4.5E-06/yr) and in view of the recognized conservatisms in the FIVE methodology as applied by Fermi 2, this result is considered to represent an acceptably low risk to fire induced damage and thus presents no vulnerabilities.

Five of the unscreened six compartments are control center compartments including the relay and control rooms, the switchgear rooms, and Division 1 portion of the miscellaneous room, which is a finding consistent with other plants. The sixth compartment is the second floor of the reactor building. This latter unscreened compartment leads to the single fire insight in that the dominating contributors are cabinets used for dedicated shutdown and whose loss would isolate the affected equipment from the main control room thereby causing loss of the equipment function. While this loss potential is adequately covered by current operator training, additional Fire Brigade drills in the vicinity of these cabinets are planned to increase the awareness of the brigade members to the need to quickly isolate and extinguish such cabinet fires. This training activity is being initiated through a Training Work Request **[8.12]**.

8.3 OTHER EVENTS SUMMARY (High Winds, Floods, And Transportation And Nearby Facility Accidents)

The site review and design comparison relative to the 1975 Standard Review Plan revealed no vulnerabilities or insights relative to these other external events. This review included a screening process that assured there were no additional external events relevant to the Fermi 2 site. The recently observed potential for a common cause failure of diesel generator cooling function due to ice formation is the subject of LER 96-001 **[8.10]** and is currently under evaluation.

8.4 PROPOSED RESOLUTION OF UNRESOLVED AND GENERIC SAFETY ISSUES

There are two basic categories of related programs dealing with generic safety issues that are not yet resolved: (1) those that are subsumed in the IPEEE and (2) those that are related by topic to the IPEEE and that could potentially be resolved through the IPEEE effort. Those issues that apply to Fermi 2 in both categories are briefly summarized with a statement of intent for those cases where resolution is intended.

8.4.1 Issues Subsumed in the IPEEE

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

  The information given in the USI-A45 evaluation presented in Sections 3.2 and 4.6.1 and the insights presented in the IPE Report **[8.8]**, Section 3.4.3, are considered a sufficient basis to resolve USI A-45 for Fermi 2.

- The Eastern U.S. Seismicity Issue

  The seismic portion of this IPEEE is considered a sufficient basis to resolve this issue for Fermi 2. See Section 3.2.

8.4.2 Other Related Issues

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

  The seismic portion of this IPEEE coupled with the justification provided in Section 3.2 is considered a sufficient basis to resolve USI A-17 for Fermi 2.

- USI A-40, "Seismic Design Criteria, A Short Term Program"

  This issue is not considered relevant to Fermi 2 for the reasons given in Section 3.2.

- GI-57, "Effects of Fire Protection System Actuation on Safety Related Equipment"

  The fire portion of this IPEEE as discussed in Section 4.6.2 is considered a sufficient basis to resolve GI-57 for Fermi 2.

It should be noted that GI-131 and USI A-46 are not applicable to Fermi. Note also that the Sandia "Fire Risk Scoping Study" (NUREG/CR-5088) issues were addressed in the fire portion of this IPEEE as requested in Generic Letter 88-20, Supplement 4.

8.5 REFERENCES

8.1 "Deviations Found During Seismic IPEEE Walkdowns," DER 94-0644, November 1, 1994.

8.2 "Seismic Restraint of Panels in the Relay room, Switchgear Rooms and RHR Div. 2 Switchgear Rooms," EDP-27108, November 1995.

8.3 "Replace R30 EDG at Voltage Relay Westinghouse SV-1 with Westinghouse SSV-T," TSR-27566, May 2, 1995.

8.4 "Restraint of ILRT Air Dryer Tank on Second Floor Reactor Building," TSR-28195, February 28, 1996.

8.5 "Include Lessons Learned Emphasizing the Importance of Mounting Hardware Installation," WR 96-0161, February 14, 1996.

8.6 "Develop Simulator Scenario to Support IPEEE Analysis to be Used for Evaluating Operator Response," WR 96-0129, February 7, 1996.

8.7 Deleted

8.8 Fermi 2 Individual Plant Examination (Internal Events), Detroit Edison, August, 1992.

8.9 "Spare Relay for Stock Code 480-2622 Did Not Pass Seismic Test," DER 95-0104, February 7, 1995.

8.10 "Emergency Diesel Generator Cooling Water Function Potentially Lost Due to Ice Formation in the Pump Column," LER 96-001, Event Date February 6, 1996, Report Date March 6, 1996.

8.11 "Seismic Mounting Weakness in Panels H21P296A & B," DER 96-0289, March 18, 1996.

8.12 Training Work Request (Programs), "Develop Fire Drill Scenario for Second Flr. Reactor Bldg. by Dedicated Shutdown Panel H21-P625," WR 96-0271, March 20, 1996.